B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM) ~~(= Tave > 200°F)~~ Strike-out ~~(= Tave > 200°F)~~

BASES (CP 3.1.1.6-1) (DC 3.1-ED)

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Rod Control ~~Rod~~ System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Rod Control ~~Rod~~ System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The ~~soluble boron~~ Chemical and Volume Control System can control the soluble boron concentration to compensate for fuel depletion during operation and all xenon burnout reactivity changes and can maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured, assuming that core reactivity is within design limit of LCO 3.1.2, by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.~~7~~6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration

APPLICABLE SAFETY ANALYSIS

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis ~~(Ref. 2)~~ establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

~~analysis~~ FSAR (Refs. 2 and 3). In addition to the limiting MSLB transient, the SDM requirement is also used in the analyses of the following events:

APPLICABLE SAFETY ANALYSIS (continued)

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition;

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

d. Rod ejection.

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 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 with RCS T_{avg} equal to 547°F. 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 is also used in the analyses of the following events:~~

~~a. Inadvertent boron dilution;~~

~~b. An uncontrolled rod withdrawal from subcritical or low power condition; and~~

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

~~d. Rod ejection.~~

~~Each of these events is discussed below.~~ Strike-out

Q3.1.6-1

APPLICABLE SAFETY ANALYSIS (continued)

In the boron dilution analysis, 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.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod 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 ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

~~will~~ redline. (B3.1.G-1)

The startup of an inactive RCP in MODES 1 or 2 is precluded. In MODE 3, the startup of an inactive RCP can 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 start is less than half the minimum required

(Continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

~~NRC Policy Statement~~ DC 3.1-ED

SDM satisfies Criterion 2 of ~~the~~ 10CFR50.36(c)(2)(ii). 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 assumption.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

DC 3.1-ED

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be ~~applicable~~ sufficient. The required SDM is specified in the COLR.

APPLICABILITY

In MODE 2 with k_{eff} < 1.0 and in MODES 3, ~~and~~ 4, and 5 the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. ~~[In MODE 5, SDM is addressed by LCO 3.1.2, "SHUTDOWN MARGIN (SDM) — T_{avg} ≤ 200°F."]~~ In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.~~65~~, "Shutdown Bank Insertion Limits," and LCO 3.1.~~76~~, "Control Bank Insertion Limits" ← redline

Q3.1.6-1

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the borated water source should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

BASES

~~In determining the boration flow rate, the time in core life must be considered. Forinstance, the most dificult time in core life to increase the RCS boron concentration is at the begining of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% Δk/k must be recovered and a boration flow rate of [ ] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approcimately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of the RCS~~

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.~~5~~6 and LCO 3.1.~~6~~7 are met. In the event that a rod is known to be untrippable, however, SDM

Q3.1.6-1

~~parameters will increase the SDM by 1% Δk/k. These boration parameters of [ ] gpm and [ ] ppm represent typical values and are provided for the purpose of offering a specific example.~~

(Continued)

APPLICABLE SAFETY ANALYSIS (continued)

against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide (continued) an accurate representation of the core reactivity. DC 3.1-ED

Design calculations are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion as well as providing inputs to the safety analysis.

The comparison between measured and predicted initial core reactivity provides a validation of the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron redline concentration to the measured value ~~is typically performed after~~ when deemed necessary shall be performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle. Q3.1.G-1

Core reactivity satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii). DC 3.1-ED

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily altered once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of ± 1% Δk/k has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

(Continued)

DC 3.1-ED

APPLICABLE SAFETY ANALYSIS (Ref. 2 (Continued))

The consequences of accidents that cause core overheating must be evaluated when the is positive. Such accidents include the rod withdrawal transient from either zero or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. An EOC measurement is conducted at conditions when the RCS boron concentration reaches a boron redline concentration equivalent to 300 ppm at an equilibrium, all rods out, RTP condition. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions. @3.1.6-1

redline

The most negative MTC value, equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR accident analyses to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) adding margin to this value to account for the largest difference in MTC observed between an EOC, all rods withdrawn, RATED THERMAL POWER condition and an envelope of those most adverse conditions of moderator temperature and pressure, rods inserted to their insertion limits, axial power skewing, and xenon concentration that can occur in normal operation within Technical Specification limits and lead to a significantly more negative EOC MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR accident analyses into the limiting EOC MTC value. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by adding an allowance for burnup and soluble boron concentration changes to the limiting EOC MTC value.

(Continued)

BASES

strike out ~~more positive~~ P3.1.6-1 redline

~~b~~3. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is less negative than the 60 ppm Surveillance limit, the EOC limit will not be exceeded because of the gradual manner in which MTC changes with core burnup. P DC3.1-ED

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.

2. FSAR, Chapter 15.

3. WCAP-9273-A , "Westinghouse Reload Safety Evaluation Methodology," July 1985.

~~4. FSAR, Chapter [15].~~

DC3.1-ED

BASES

~~with a center-to-center distance of 3.75 inches, which is six steps~~ (P3.1.6-1)

BACKGROUND position (continued)

group all receive the same signal to move and should, therefore, all be at the same indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod. ~~(± 7.5 steps)~~ (P3.1.6-1)

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data system fails, the DRPI will go on half accuracy. ~~with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 6 steps (±3.75 inches), and the maximum uncertainty is ±12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.~~ The DRPI system is capable of monitoring rod position within at least ± 12 steps with either full accuracy or half accuracy

APPLICABLE SAFETY ANALYSIS

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing rod inoperability or misalignment are that:

a. There be no violations of:

1. Specified acceptable fuel design limits, or

2. Reactor Coolant System (RCS) pressure boundary integrity: and

b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control or shutdown rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

(Continued)

BASES

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are typically fully inserted and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)~~—T_{avg} > 200°F~~," for SDM in MODES 2 with K_{eff}<1.0, 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable ~~(i.e., untrippable)~~, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM.

With an inoperable rod(s), This ACTION provides for verification of SDM, this is most simply accomplished by verifying rod insertion limits are met. Additionally, actions could include calculation of the current SDM and boration to meet limits specified in the COLR or proceed

(Continued)

ACTION (continued)

to MODE 3. These actions are consistent with those specified in LCO 3.1.5 and LCO 3.1.6.

A rod is considered trippable if it was demonstrated OPERABLE during the last performance of SR 3.1.4.2 and met the rod drop time criteria during the last performance of SR 3.1.4.3.

In this situation, SDM verification must ~~include the worth~~ account for the absence of the negative reactivity of the untrippable rod(s), as well as the rod of maximum worth.

redline redline Q3.1-15

A.2

If the ~~untrippable~~ inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

redline

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1

Q 3.1.G-1

When a rod becomes misaligned, it can usually be moved and is still trippable (i.e., OPERABLE). If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

redline

An alternative to realigning a single misaligned RCCA to the group demand position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO ~~3.1.6~~ 3.1.5, "Shutdown Bank Insertion Limits," and LCO ~~3.1.7~~ 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

(Continued)

BASES

~~D.1~~ C.1 redline

Q3.1.G-1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

~~C.1.1 and C.1.2~~ D.1.1 and D.1.2 redline

Q3.1.G-1

More than one ~~control~~ rod becoming misaligned from its group ~~average demand~~ position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to

(Continued)

DC3.1-ED

BASES

ACTIONS and (continued)

complete the action. This allows the operator sufficient time to align the required valves start the boric acid pumps. Boration will continue until the required SDM is restored.

~~Additionally, the requirements of LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," apply if the misaligned rods are not within the required insertion limits.~~

~~C.2~~ D.2 redline

@3.1.6-1

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant system.

SURVEILLANCE REQUIREMENTS

SR ~~3.1.5.1~~ 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. ~~If the rod position deviation monitopr is inoperable, a Frequency of 4 hours accomplishes the same goal.~~ The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

(Continued)

SR ~~3.1.5.2~~ 3.1.4.2

Verifying each rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each rod would result in radial or axial power tilts, or oscillations. Exercising each individual rod every 92 days provides confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR ~~3.1.5.1~~ 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between or during required performances of SR ~~3.1.5.2~~ 3.1.4.2 (determination of rod OPERABILITY by movement), if a rod(s) is discovered to be immovable, but remains trippable, the rod(s) is considered to be OPERABLE. At any time, if a rod(s) is immovable, a determination of the trippability (OPERABILITY) of the rod(s) must be made, and appropriate action taken.

redline Q3.1.G-1 Q3.1.G-1 control and aligned Q3.1-15

SR ~~3.1.5.3~~ 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions.

Control Q3.1.G-1

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B ~~3.1.6~~ 3.1.5 Shutdown Bank Insertion Limits

DC 3.1-ED

BASES

redline

Q3.1.6-1

ACTIONS (continued)

~~for an extended period of time.~~ Additionally, the requirements of LCO ~~3.1.5~~ 3.1.4, "Rod Group Alignment Limits," apply if one or more shutdown rods are not within the required alignment limits.

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR ~~3.1.6.1~~ 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
2. 10 CFR 50.46.
3. FSAR, Chapter 15, Section 15.4.3.2.4.

B 3.1 REACTIVITY CONTROL SYSTEMS

B ~~3.1.7~~ 3.1.6 Control Bank Insertion Limits

BASES

~~are normally controlled~~ (Q3.1.G-1)

~~Figure B 3.1.7-1~~The COLR also indicates how the control banks are moved in an overlap pattern. Overlap is the distance traveled together by two control banks. ~~The predetermined position of control bank C, at which control bank D will begin to move with bank C on a withdrawal, will be at 118 steps for a fully withdrawn position of 231 steps. The fully withdrawn position is defined in the COLR.~~ The control banks are used for precise reactivity control of the reactor. The positions of the control banks can be controlled manually, or automatically by the Rod Control System. They are capable of ~~altering~~ adding reactivity very quickly (compared to borating or diluting). (DC 3.1-ED) redline

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO ~~3.1.5~~ 3.1.4, "Rod Group Alignment Limits," LCO ~~3.1.6~~ 3.1.5, "Shutdown Bank Insertion Limits," LCO ~~3.1.7~~ 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained assuming LCO 3.1.2, "Core Reactivity" is met for core reactivity.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the

(Continued)

(DC 3.1-ED)

BASES

SURVEILLANCE REQUIREMENT

SR 3.1.~~7.26~~.2

Verification of the control bank insertion limits at a Frequency of 12 hours is sufficient toensure OPERABILITY and to detect control banks that may be approaching the insertion (continued)limits since, normally, very little rod motion occurs in 12 hours. ~~If the insertion limit monitor becomes inoperable, verification of the control bank position at a frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.~~

SR 3.1.~~7.36~~.3

When control banks are maintained within their insertion limits as checked by SR ~~3.1.7.2~~ 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. The verification of compliance with the sequence and overlap limits specified in the COLR consists of an observation that the static rod positions of those control banks not fully withdrawn from the core are within the limits specified in the COLR. Bank sequence and overlap must also be maintained during rod movement, implicit within the LCO. A Frequency of 12 hours is consistent with the insertion limit check above in SR ~~3.1.7.2~~ 3.1.6.2.

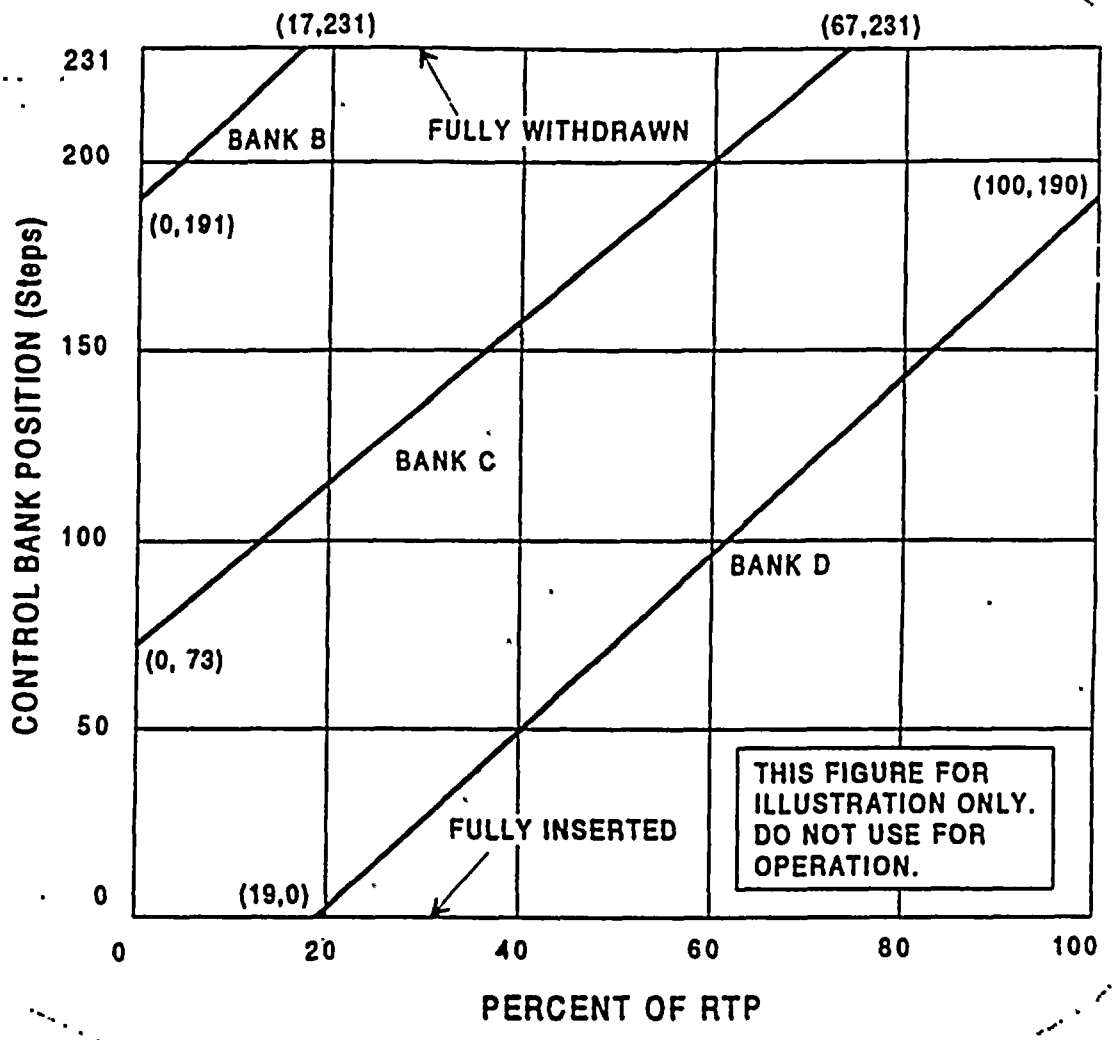
REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.
2. 10 CFR 50.46.
3. FSAR, Chapter 4, Section 4.3.2.4.
4. FSAR, Chapter 4, Section 4.3.2.4.
5. ~~FSAR,~~ WCAP-9273-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

Show Figure B3.1.7-1 as struck-out

Q3.1.6-1

6 Delete



Q 3.1.G-1

~~Figure B 3.1.7-1 (page 1 of 1)~~
~~Control Bank Insertion vs. Percent RTP~~

BASES

APPLICABLE SAFETY ANALYSIS

Control and shutdown rod position accuracy is essential during power operation. Powerpeaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the bank sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO ~~3.1.6~~ 3.1.5, "Shutdown Bank Insertion Limits," and LCO ~~3.1.7~~ 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO ~~3.1.5~~ 3.1.4, "Rod Group Alignment Limits"). ~~Control~~ Rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The ~~control~~ rod position indicator channels satisfy Criterion 2 of 10CFR50.36(c)(2)(ii). The ~~control~~ rod position indicators monitor ~~control~~ rod position, which is an initial condition of the accident.

LCO

LCO ~~3.1.8~~ 3.1.7 specifies that ~~the~~ one DRPI System and ~~the~~ one Bank Demand Position Indication System be OPERABLE for each ~~control~~ rod. For the ~~contol~~ rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following: Q3.1.6-1

a. The DRPI System on either full accuracy or half accuracy indicates within 12 steps of the group step counter demand position as required by LCO ~~3.1.5~~ 3.1.4, "Rod Group Alignment Limits"; and

~~b. For the DRPI system, there are no failed cells; and~~

b. ~~c.~~ The Bank Demand Indication System has been ~~calibrated either~~ reset in the fully inserted position, fully withdrawn position or to the DRPI System. Q3.1.6-1

The 12 step agreement limit between the Bank Demand Position Indication System and the DRPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of ~~control~~ rod bank position.

A deviation of less than the allowable limit, given in LCO ~~3.1.5~~ 3.1.4, in position indication for a single rod, ensures high confidence that the position uncertainty of the corresponding ~~control~~ rod group is within the assumed values used in the analysis (that specified ~~control~~ rod group insertion limits).

(Continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-2 **APPLICABILITY:** DC, CP

REQUEST:

3.1.1 Shutdown Margin (SDM) (Comanche Peak and Diablo Canyon)
DOC 01-06-A
CTS 3/4.1.1 Applicability
ITS 3.1.1 Applicability

Comment: According to the Conversion Comparison Table, "MODE 2 with Keff < 1.0" and "MODE 5" are added to the Applicability section of TS 3.1.1 for Comanche Peak and Diablo Canyon. All of the FLOG ITS Sections 3.1.1 have these applicability requirements included in the ITS and not in the CTS. Provide a discussion for Comanche Peak and Diablo Canyon explaining/justifying these changes.

FLOG RESPONSE: In the CTS, shutdown margin is controlled via LCO 3.1.1.1 for MODES 1, 2, 3 and 4, and LCO 3.1.1.2 for MODE 5. Rod insertion limits are controlled by LCO 3.1.3.5 for shutdown rods, MODES 1 and 2, and LCO 3.1.3.6 for control rods, MODES 1 and 2. When the reactor is critical (MODE 1 and MODE 2 with Keff ≥ 1.0), shutdown margin is assured, via the CTS, by assuring that rod insertion limits are met (see SR 4.1.1.1.16.) When the reactor is not critical (MODE 2 with Keff < 1.0, MODE 3, MODE 4 and MODE 5), shutdown margin is assured by assessing boron concentration, temperature, etc. (see SR 4.1.1.1.1e and 4.1.1.2). The CTS requirements have been clarified and reorganized such that the rod insertion limits and shutdown margin requirements for MODES 1 and 2 with Keff ≥ 1.0 from CTS 3.1.1.1, 3.1.3.5 and 3.1.3.6 are converted to ITS 3.1.5 and 3.1.6. The shutdown margin requirements for MODE 2 with Keff < 1.0, MODE 3, MODE 4, and MODE 5 from CTS 3.1.1.1 and 3.1.1.2 are converted to ITS 3.1.1. This reorganization does not change requirements but presents them in a more logical manner, and therefore is a purely administrative change.

DOC 1-06-A will be revised to add the following:

"In the ITS format, the SHUTDOWN MARGIN in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ is controlled through compliance with control rod insertion limits (ITS LCO 3.1.5 and LCO 3.1.6). For those modes or conditions in which compliance with control rod insertions limits is not required, the SHUTDOWN MARGIN is verified in the more traditional manner by consideration of such factors as Reactor Coolant System boron concentration, coolant temperature, xenon and samarium concentrations, etc. Thus, the applicability of CTS 3.1.1.1 is modified by this change to be applicable to MODE 2 with $k_{eff} < 1.0$ as well as the current MODES 3 and 4. This change is considered to be administrative in nature, because, when the reactor was critical (MODE 1 and MODE 2 with $k_{ff} \geq 1.0$), the SHUTDOWN MARGIN was determined, in accordance with SR 4.1.1.1.1.b, by verifying compliance with the control rod insertion limits.

In addition, the SHUTDOWN MARGIN requirement, surveillances, and actions are the same for operation in MODE 5 as for operation in MODES 3 and 4. Therefore, the specifications have been combined to include MODE 5 with MODES 3 and 4. The change is considered to be

administrative in nature, because there is no change in the LCO, ACTIONS, or SURVEILLANCE REQUIREMENTS. See also Change 2-01-A."

ATTACHED PAGES:

Encl. 3A 2

CHANGE NUMBER	NSHC	DESCRIPTION
01-05	LG	The list of specific items to be considered in the performance of an SDM verification would be deleted. These items are listed in the ITS Bases. This change is of the type that removes unnecessary procedural details from the specifications while leaving the overall safety requirement intact.
01-06	A	This change revises the SDM limiting condition of operation (LCO) Applicability to MODE 2 with k_{eff}<1.0, MODE 3, and MODE 4. This change also creates a new core reactivity LCO based on ITS 3.1.3 This is consistent with NUREG-1431. Insert 3.1.2
01-07	LS16	The term "immediately" is changed to "15 minutes" which is more specific. The term "immediately" simply specifies a prompt ACTION. The term "completion time of 15 minutes" is meant to clearly state a completed ACTION. The requirements of this ACTION are met only if boron is already being injected at 15 minutes. This time period provides adequate time for the operator to align and start the required systems. This is consistent with NUREG-1431. (Q3.1-2)
01-08	A	The technical contents of this SR (verification of SDM through compliance with rod insertion limits) in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ have been incorporated into LCO 3.1.6 of the ITS.
01-09	A	The SR for verification of the estimated critical Condition during the approach to criticality is moved to ITS SR 3.1.6.1.
01-10	M	CTS SR 4.1.1.1.1.e is modified by this change to be applicable to MODE 2 with k_{eff}<1.0 as well as current MODES 3 and 4. This is consistent with NUREG-1431. Insert (Q3.1-3)
02-01	A	In the conversion process, this LCO will be combined with the SDM LCO applicable for T_{avg} > 200°F, in accordance with Traveler TSTF-136. Traveler TSTF-9, Rev. 1, relocated values for SDM to the COLR which removed the only difference between ITS LCO 3.1.1 and ITS LCO 3.1.2. Differences above and below 200°F will be addressed in the COLR.
03-01	A	The footnote referring to CTS special test exceptions would be deleted. This is acceptable because the requirements for special test exceptions are provided in separate LCOs. Therefore, a separate reference in the footnote is redundant.
03-02	LS3	ACTION Statement A.1 would be revised to require achieving MODE 2 with k_{eff} < 1.0 instead of achieving HOT STANDBY if the beginning of life (BOL) moderator temperature coefficient (MTC) limit is exceeded and revised rod withdrawal limits cannot be established. This change corrects a discrepancy between the BOL Applicability and the ACTION, while ensuring that the plant is taken to a Condition in which the LCO is not applicable. Revising the CTS, even to correct an inconsistency, represents a relaxation in ACTION Statement A.1.

DOC 01-06-A

In the ITS format, the SHUTDOWN MARGIN in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ is controlled through compliance with control rod insertion limits (ITS LCO 3.1.5 and LCO 3.1.6). For those modes or conditions in which compliance with control rod insertions limits is not required, the SHUTDOWN MARGIN is verified in the more traditional manner by consideration of such factors as Reactor Coolant System boron concentration, coolant temperature, xenon and samarium concentrations, etc. Thus, the applicability of CTS 3.1.1.1 is modified by this change to be applicable to MODE 2 with $k_{eff} < 1.0$ as well as the current MODES 3 and 4. This change is considered to be administrative in, because, when the reactor was critical (Mode 1 and 2 with $k_{ff} \geq 1.0$), the SHUTDOWN MARGIN was determined, in accordance with SR 4.1.1.1.1.b, by verifying compliance with the control rod insertion limits.

In addition, the SHUTDOWN MARGIN requirement, surveillances, and actions are the same for operation in MODE 5 as for operation in MODEs 3 and 4. Therefore, the specifications have been combined to include MODE 5 with MODEs 3 and 4. The change is considered to be administrative in nature, because there is no change in the LCO, ACTIONS, or SURVEILLANCE REQUIREMENTS. See also Change 2-01-A.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-3 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

3.1.1 Shutdown Margin (SDM) (All FLOG Plants)
DOC 01-10-M
CTS SR 4.1.1.1.1
ITS SR 3.1.1.1

Comment: The justification for modifying applicability of SR 3.1.1.1 is inadequate; it only refers to consistency with NUREG-1431. Also, it is not apparent why this change is not applicable to Wolf Creek and Callaway.

FLOG RESPONSE: For DCPP and CPSES, DOC 01-10-M is revised to add the following: "In the ITS format, the SHUTDOWN MARGIN in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ is controlled through compliance with control rod insertion limits. For those modes or conditions in which compliance with control rod insertion limits is not required, the SHUTDOWN MARGIN is verified in the more traditional manner by consideration of such factors as Reactor Coolant System boron concentration, coolant temperature, xenon and samarium concentrations, etc. Thus, the applicability of CTS SR 4.1.1.1.1.e is modified by this change to be applicable to MODE 2 with $k_{eff} < 1.0$ as well as the current MODES 3 and 4. This change is more restrictive, in that CTS 4.1.1.1.1.b addresses MODES 1 and 2 with $k_{eff} \geq 1.0$, and CTS 4.1.1.1.1.e addresses MODES 3 and 4. MODE 2 with $k_{eff} < 1.0$ is not specifically addressed in the CTS. See also revised Change 01-06-A, which provides a broad discussion of how the applicabilities for CTS 3.1.1.1, 3.1.1.2, 3.1.3.5, and 3.1.3.6 have been revised."

The Wolf Creek and Callaway Technical Specifications were modified by Amendment 89 and 103 respectively, to contain MODE 3, 4, and 5 Specifications for "Shutdown Margin" and a separate MODE 1 and 2 Specification for "Core Reactivity." This eliminated the need for individual MODE applications under the Surveillance Section. Wolf Creek and Callaway used DOC 01-02-M to make the MODE 2 with $K_{eff}<1.0$ change to both the LCO and the SR. This makes DOC 01-10-M not applicable to Wolf Creek and Callaway (see Enclosure 3B).

ATTACHED PAGES:

Encl. 3A 2

CHANGE NUMBER	NSHC	DESCRIPTION
01-05	LG	The list of specific items to be considered in the performance of an SDM verification would be deleted. These items are listed in the ITS Bases. This change is of the type that removes unnecessary procedural details from the specifications while leaving the overall safety requirement intact.
01-06	A	This change revises the SDM limiting condition of operation (LCO) Applicability to MODE 2 with k_{eff}<1.0, MODE 3, and MODE 4. This change also creates a new core reactivity LCO based on ITS 3.1.3 ~~3.1.2~~ This is consistent with NUREG-1431. Insert
01-07	LS16	The term "immediately" is changed to "15 minutes" which is more specific. The term "immediately" simply specifies a prompt ACTION. The term "completion time of 15 minutes" is meant to clearly state a completed ACTION. The requirements of this ACTION are met only if boron is already being injected at 15 minutes. This time period provides adequate time for the operator to align and start the required systems. This is consistent with NUREG-1431.
01-08	A	The technical contents of this SR (verification of SDM through compliance with rod insertion limits) in MODE 1 and MODE 2 with k_{eff}≥1.0 have been incorporated into LCO 3.1.6 of the ITS.
01-09	A	The SR for verification of the estimated critical Condition during the approach to criticality is moved to ITS SR 3.1.6.1.
01-10	M	CTS SR 4.1.1.1.1.e is modified by this change to be applicable to MODE 2 with k_{eff}<1.0 as well as current MODES 3 and 4. This is consistent with NUREG-1431. Insert
02-01	A	In the conversion process, this LCO will be combined with the SDM LCO applicable for T_{avg} > 200°F, in accordance with Traveler TSTF-136. Traveler TSTF-9, Rev. 1, relocated values for SDM to the COLR which removed the only difference between ITS LCO 3.1.1 and ITS LCO 3.1.2. Differences above and below 200°F will be addressed in the COLR.
03-01	A	The footnote referring to CTS special test exceptions would be deleted. This is acceptable because the requirements for special test exceptions are provided in separate LCOs. Therefore, a separate reference in the footnote is redundant.
03-02	LS3	ACTION Statement A.1 would be revised to require achieving MODE 2 with k_{eff} < 1.0 instead of achieving HOT STANDBY if the beginning of life (BOL) moderator temperature coefficient (MTC) limit is exceeded and revised rod withdrawal limits cannot be established. This change corrects a discrepancy between the BOL Applicability and the ACTION, while ensuring that the plant is taken to a Condition in which the LCO is not applicable. Revising the CTS, even to correct an inconsistency, represents a relaxation in ACTION Statement A.1.

DOC 01-10-M

In the ITS format, the SHUTDOWN MARGIN in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ is controlled through compliance with control rod insertion limits. For those modes or conditions in which compliance with control rod insertions limits is not required, the SHUTDOWN MARGIN is verified in the more traditional manner by consideration of such factors as Reactor Coolant System boron concentration, coolant temperature, xenon and samarium concentrations, etc. Thus, the applicability of CTS SR 4.1.1.1.1.e is modified by this change to be applicable to MODE 2 with $k_{eff} < 1.0$ as well as the current MODES 3 and 4. This change is more restrictive, in that CTS 4.1.1.1.1.b addresses MODES 1 and 2 with $k_{eff} \geq 1.0$, and CTS 4.1.1.1.1.e addresses MODES 3 and 4. MODE 2 with $k_{eff} < 1.0$ is not specifically addressed in the CTS. See also revised Change 01-06-A, which provides a broad discussion of how the applicability for CTS 3.1.1.1, 3.1.1.2, 3.1.3.5 and 3.1.3.6 has been revised."

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-4 **APPLICABILITY:** DC, CP

REQUEST:

3.1.2 Core Reactivity (Comanche Peak & Diablo Canyon)
DOC 05-06-A
JFD 3.1-2
CTS SR 4.1.1.1.2
ITS SR 3.1.2.1

Comment: The note to the core reactivity SR in the STS states that ".. predicted reactivity values **may** be adjusted (normalized) ...", while the note in the ITS states, ".. predicted reactivity values **shall** be adjusted (normalized) ...". The ITS use of the word "shall" is based upon the CTS use of the word. The Bases supporting this SR adds a parenthetical phrase stating "...normalization (adjustment, only if necessary)...", indicating that the STS wording is preferable. Using the word "shall" implies that an adjustment must always be may, regardless of the necessity. Adopt the STS wording to the SR 3.1.2.1 Note.

FLOG RESPONSE: Consistent with the STS wording, the ITS and CTS have been revised to use the word "may" instead of "shall." DOC 05-06-A is used as the justification for the CTS change.

ATTACHED PAGES:

Encl. 2	3/4 1-2, 3/4 1-8
Encl. 3A	5
Encl. 3B	4
Encl. 5A	3.1-3
Encl. 6A	1
Encl. 6B	1

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 2*, 3 or 4, at least once per 24 hours ~~by consideration of the following factors:~~ 01-10-M

~~1) Reactor Coolant System boron concentration,~~

01-05-LG

~~2) Control rod position,~~

~~3) Reactor Coolant System average temperature,~~

~~4) Fuel burnup based on gross thermal energy generation,~~

~~5) Xenon concentration, and~~

~~6) Samarium concentration.~~

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k once prior to entering MODE 1 after each refueling and at least once per 31 Effective Full Power Days (EFPD) after burnup > 60 EFPD. ~~This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above.~~ The predicted reactivity values ~~shall~~ be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. If the reactivity balance is not within limits, within ~~72 hours~~ evaluate the Safety Analyses and establish appropriate operating restrictions and/or surveillance requirements, or be in at least MODE 3 within the next 6 hours.

05-01-M
05-03-LG
05-02-LS7
05-05-LS17
7 days
TR 3.1-003
05-06-A
may
Q3.1-4
Q3.1-4

01-10-M

* With K_{eff} ≥ 1.0

L

DC 3.1-ED

SURVEILLANCE	FREQUENCY
(new) ----------NOTE---------- The predicted reactivity values ~~shall~~ may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. ----------	01-06-A @3.1-4 05-03-LG
Verify measured core reactivity is within ± 1% Δk/k of predicted values.	05-04-A 05-01-M 05-02-LS7 Once prior to entering MODE 1 after each refueling AND -----NOTE------ Only required after 60 EFPD ---------- 31 EFPD thereafter

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
05-04	A	CTS SR requires a SDM verification prior to operation above 5 percent power after each refueling with the control rod banks at maximum insertion limits. SDM in MODES 1 and 2 is determined by shutdown and control rods maintained at their insertion limits. The relevant requirements regarding the adequacy of the SDM with rods at their insertion limits is determined through compliance with ITS 3.1.2, which requires a reactivity balance prior to entering MODE 1.after each refueling; and ITS SR 3.1.6.1, which requires a verification of control bank position within insertion limits within 4 hours prior to criticality. Therefore, the requirements of this SR would be performed by other specifications in the ITS. [ ]
05-05	LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown. This is consistent with NUREG-1431, as modified by TSTF-142 TR 3.1-003
05-06	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Insert Q3.1-4
05-07	LS24	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). TR 3.1-003
06-01	R	The CTS 3.1.2.1, "Boration Flow Path Shutdown," and associated SR 4.1.2.1 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-6
07-01	R	The CTS 3.1.2.2, "Boration Flow Path Operating," and associated SR 4.1.2.2 are relocated outside of the TS. This is consistent with NUREG-1431. Q3.1-7 Not Applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-01	R	The CTS 3.1.2.3, "Charging Pump Shutdown," and associated SRs 4.1.2.3.1 and 4.1.2.3.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-8
08-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS19	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-04	A	Insert Q3.1-9
09-01	R	The CTS 3.1.2.4, "Charging Pump Operating," and associated SRs 4.1.2.4.1 and 4.1.2.4.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-10
10-01	R	The CTS 3.1.2.5, "Borated Water Source Shutdown," and associated CTS SR 4.1.2.5 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-11
11-01	R	The CTS 3.1.2.6, "Borated Water Source Operating," and associated CTS SR 4.1.2.6 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-12

05-06-A

CTS SR 4.1.1.5.1 requires that the predicted reactivity values "shall" be adjusted (normalized) at 60 EFPD after refueling. ITS SR 3.1.2.1 states the normalization requirement as "may" be adjusted. This is to recognize that normalization is not necessary if predicted and measured core activity are within acceptable tolerance. The scheduling of the normalization of predicted and measured core reactivity continues to be required at 60 EFPD. Therefore, this change reflects clarification of existing intent and is considered administrative.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-05 LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown.	Yes	Yes	No, already in CTS.	No, already in CTS.
05-06 A	CTS SR 4.1.1.5.1 requires that the predicted reactivity values "shall" be adjusted (normalized) at 60 EFPD after refueling. ITS SR 3.1.2.1 states the normalization requirement as "may" be adjusted: This is to recognize that normalization is not necessary if predicted and measured core reactivity are within acceptance tolerance. The scheduling of predicted and measured core reactivity continues to be required at 60 EFPD. Therefore, this change reflects clarification of existing intent and is considered administrative.	~~No, maintaining CTS wording.~~ Yes	~~No, maintaining CTS wording.~~ Yes	Yes Q3.1-4	Yes
06-01 R	Relocates "Boration Flow Path - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 1. - relocated to CN ECG	Yes	No, see Amendment 89. Q3.1-6	No, see Amendment 103.
07-01 R	Relocates "Boration Flow Path - Operating" TS to licensee controlled document.	~~Yes, see LAR 95 07 dated 10/4/95, DCL 95-222.~~ No, see Amendments 120/118	Yes	No, see Amendment 89. Q3.1-7	No, see Amendment 103.
07-02 A	Moves limitation on charging pumps in MODE 4 to ITS SR 3.4.12.2.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-01 R	Relocates "Charging Pumps - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 3. - relocated to CN ECG	Yes	No, see Amendment 89. Q3.1-8	No, see Amendment 103.
08-02 M	Moves charging pump SR when below 350°F to ITS SR 3.4.12.2 and decreases surveillance frequency to 12 hours from 31 days.	No, already in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-03 LS19	Deletes the method of verificating that charging pumps are incapable of injecting into the RCS.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
05-07 LS24	Insert	No, see CN 05-05-LS17	No, See CN 05-05-LS17	Yes	Yes TR 3.1-003
08-04 A	Insert	Yes	No, see CN 8-02-M	No, see Amendment 89	No, see Amendment 103 Q3.1-9

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR ~~3.1.3.1~~ 3.1.2.1 ----------NOTE---------- unstrike The predicted reactivity values ~~may~~ shall be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. ----------	~~3.1-4~~ ~~3.1-2~~
Verify measured core reactivity is within ± 1% Δk/k of predicted values.	Once prior to entering MODE 1 after each refueling <u>AND</u> -----NOTE------ Only required after 60 EFPD ---------- 31 EFPD thereafter

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.1

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[ ]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER	JUSTIFICATION
3.1-1	In accordance with TSTF-9, Rev. 1, this change would relocate the specified limit for SDM from ITS to the COLR. This change occurs in several specifications including the specification for SDM and those specifications with ACTIONS that require verifying SDM within limits.
3.1-2	Q3.1-4 The Note for SR 3.1.2.1 indicates that predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 EFPD after each refueling. However, both the Bases for Specification 3.1.3 and the CTS requirements in Specification 3.1.1.5 state that the normalization shall be done prior to exceeding a fuel burnup of 60 EFPD after each refueling. Not used.
3.1-3	~~Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).~~ Not Used
3.1-4	Q3.1-23 SR 3.1.4.2 of NUREG-1431, Rev. 1 would be deleted. In accordance with TSTF-13, Rev. 1, the intent of this SR is only to determine the next frequency for SR 3.1.4.3. TR3.1-006 Performance of SR 3.1.4.2 is not necessary to assure that the LCO is met; SR 3.1.4.3 fulfills that purpose. Therefore, SR 3.1.4.2 may be deleted. In addition, the note in the frequency column of SR 3.1.4.2 would be moved to become Note 1 in the surveillance column of SR 3.1.4.3. This is for clarification purposes. As discussed in CN 3.1-9, section renumbering results in SR 3.1.4.3 of NUREG-1431, Rev. 1 becoming ITS SR 3.1.3.2.
3.1-5	Per CTS [3.1.3.1], the words "with all" have been removed from ITS LCO 3.1.4. This is a clarification that ensures the proper interpretation of the LCO. The change makes it clear that only one channel of DRPI is necessary to meet the alignment accuracy requirement of the LCO. With the word "all" in the statement, it may be possible for those unfamiliar with the DRPI design to interpret the LCO as applying to all channels of DRPI.
3.1-6	LCO 3.1.4 would be split into two separate statements to clarify that the alignment limit is separate from OPERABILITY of the control rod. The Condition A wording is broadened from "untrippable" to "inoperable" to ensure the Condition encompasses all causes of inoperability. Previous wording was ambiguous for rods that, for instance, had slow drop times but were still trippable. These slow rods are inoperable rods, and the change clarifies the appropriate ACTIONS. The Bases are changed to reflect the changes to the LCO and Condition A. These changes are based on TSTF-107.
3.1-7	This change to the ISTS would incorporate, into LCO 3.1.7, an ACTION statement that was previously approved as part of the Callaway and Wolf Creek licensing basis as revised in Enclosure 2. The ACTION statement would permit continued POWER OPERATION for up to 24 hours with more than one DRPI channel per rod group inoperable. The ACTION statement specifies additional Required ACTIONS beyond those applicable to the Condition of 1 DRPI channel per group inoperable. The Bases for this change also would be incorporated into the Bases for the plant ITS. These changes are consistent with Traveler ~~WOG-73, Rev. 1~~ TSTF-234. The note under the ACTIONS is changed to be consistent with the new Required Actions. TR3.1-006

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.1-1	In accordance with industry Traveler TSTF-9, Rev. 1,this change would relocate the specified limits for SDM from several TS to the COLR.	Yes	Yes	Yes	Yes
3.1-2	Changes the note to SR 3.1.2.1, which deals with verifying core reactivity within limits, to state that the normalization of predicted reactivity values to correspond to measured values shall be done prior to exceeding a fuel burnup of 60 EFPD after each refueling.	~~Yes~~ NA Not Used	~~Yes~~ NA	~~No, maintaining ITS wording.~~ NA	~~No, maintaining ITS wording.~~ NA Q3.1-4
3.1-3	The Wolf Creek ITS LCO 3.1.6 Required Action C.1 is revised from "Be in MODE 3." to "Be in MODE 2 with k_{eff} < 1.0." Not Used	~~No~~ NA	~~No~~ NA	~~Yes~~ NA	~~No~~ NA Q3.1-23
3.1-4	In accordance with industry Traveler TSTF-13 Rev. 1, SR 3.1.4.2, which requires verifying MTC within the 300 ppm boron limit, is deleted and the note in that SR is moved to the SR that requires the lower MTC limit to be verified. The deleted SR is not a requirement separate from the lower MTC verification SR, but is essentially a clarification of when the SR for the lower MTC limit should be performed.	Yes TR3.1-006	Yes	Yes	Yes
3.1-5	Per CTS [3.1.3.1], the words "with all" are removed from the LCO for control rod alignment limits. This ensures that the number of channels of DRPI required to be OPERABLE will not be misconstrued.	Yes	Yes	Yes	Yes
3.1-6	In accordance with TSTF - 107, the change provides additional clarification that the alignment limits in the LCO are separate from the OPERABILITY of a control rod.	Yes	Yes	Yes	Yes
3.1-7	An ACTION statement that was previously approved as part of the current licensing basis of Callaway and Wolf Creek would be added to ITS 3.1.7, as revised in Enclosure 2. The ACTION statement would permit operation for up to 24 hours with more than one digital rod position indicator per group inoperable.	Yes	Yes	Yes	Yes

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-6 **APPLICABILITY:** DC, CP

REQUEST:

CTS 3/4.1.2 Boration Systems (Comanche Peak and Diablo Canyon)
DOC 06-01-R

Comment: The Discussion of Change (DOC) needs to specify where the CTS specification is being relocated. Correct the DOC.

FLOG RESPONSE: DOC 06-01-R is revised and Technical Specification Screening Form for CTS 3.1.2.1 prepared to provide additional justification for the relocation. This justification shows that the boration system is not assumed to operate to mitigate any accident. The maintenance of SDM provides all required reactivity margin. Since the system does not mitigate an accident, there is no installed instrumentation which is used to detect or indicate a significant degradation of the RCS boundary. This system is also not associated with any variable, design feature, or operating limit that is the initial condition of an event which challenges a fission product boundary. The boration system is not a part of, nor does it support a system requiring that support, to function as part of the success path to mitigate a design bases accident.

The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specifications branch reviewers during telephone calls on June 25 and June 30, 1998.

ATTACHED PAGES:

Encl. 3A	5
Encl. 3B	4

CHANGE NUMBER	NSHC	DESCRIPTION
05-04	A	CTS SR requires a SDM verification prior to operation above 5 percent power after each refueling with the control rod banks at maximum insertion limits. SDM in MODES 1 and 2 is determined by shutdown and control rods maintained at their insertion limits. The relevant requirements regarding the adequacy of the SDM with rods at their insertion limits is determined through compliance with ITS 3.1.2, which requires a reactivity balance prior to entering MODE 1.after each refueling; and ITS SR 3.1.6.1, which requires a verification of control bank position within insertion limits within 4 hours prior to criticality. Therefore, the requirements of this SR would be performed by other specifications in the ITS. [ ]
05-05	LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown. This is consistent with NUREG-1431, as modified by TSTF-142 TR 3.1-003
05-06	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Insert Q3.1-4
05-07	LS24	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). TR 3.1-003
06-01	R	The CTS 3.1.2.1, "Boration Flow Path Shutdown," and associated SR 4.1.2.1 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-6
07-01	R	The CTS 3.1.2.2, "Boration Flow Path Operating," and associated SR 4.1.2.2 are relocated outside of the TS. This is consistent with NUREG-1431. Q3.1-7 Not Applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-01	R	The CTS 3.1.2.3, "Charging Pump Shutdown," and associated SRs 4.1.2.3.1 and 4.1.2.3.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-8
08-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS19	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-04	A	Insert Q3.1-9
09-01	R	The CTS 3.1.2.4, "Charging Pump Operating," and associated SRs 4.1.2.4.1 and 4.1.2.4.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-10
10-01	R	The CTS 3.1.2.5, "Borated Water Source Shutdown," and associated CTS SR 4.1.2.5 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-11
11-01	R	The CTS 3.1.2.6, "Borated Water Source Operating," and associated CTS SR 4.1.2.6 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-12

The boration subsystem of the chemical and volume control system (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help control the boron concentration to maintain shutdown margin (SDM). To accomplish this functional requirement, the boration systems TS require a source of borated water, one or more flow paths to inject this borated water into the reactor coolant system (RCS), and appropriate charging pumps to provide the necessary charging head.

This proposed TS revision relocates requirements, which do not meet the TS criteria in 10CFR50.36 (c) (2) (ii), to documents with established control programs. This regulation addresses the scope and purpose of the TS. In doing so, It sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. Relocation of these requirements allow the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS. An evaluation of the applicability of these criteria to this specification is provided in Attachment 21.

To ensure an appropriate level of control, these requirements will be relocated to 1) documents that are subject to the provisions of 10CFR50.59, 2) other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in FSAR which is controlled by 10CFR50.54a), or3) to programs that are controlled via the Administrative Controls section of the improved TS. The identification of the specific licensee controlled document containing this requirement is provided in Enclosure 3B of the conversion submittal. This format for specifying the location of relocated requirements was found to be acceptable to the NRC technical specification branch reviewers during telephone calls on June 25 and June 30, 1998.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure the limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-05 LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown.	Yes	Yes	No, already in CTS.	No, already in CTS.
05-06 A	CTS SR 4.1.1.5.1 requires that the predicted reactivity values "shall" be adjusted (normalized) at 60 EFPD after refueling. ITS SR 3.1.2.1 states the normalization requirement as "may" be adjusted: This is to recognize that normalization is not necessary if predicted and measured core reactivity are within acceptance tolerance. The scheduling of predicted and measured core reactivity continues to be required at 60 EFPD. Therefore, this change reflects clarification of existing intent and is considered administrative.	~~No, maintaining CTS wording.~~ Yes	~~No, maintaining CTS wording.~~ Yes	Yes Q3.1-4	Yes
06-01 R	Relocates "Boration Flow Path - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 1. - relocated to CN ECG	Yes	No, see Amendment 89. Q3.1-6	No, see Amendment 103.
07-01 R	Relocates "Boration Flow Path - Operating" TS to licensee controlled document.	~~Yes, see LAR 95 07 dated 10/4/95, DCL 95-222.~~ No, see Amendments 120/118	Yes	No, see Amendment 89. Q3.1-7	No, see Amendment 103.
07-02 A	Moves limitation on charging pumps in MODE 4 to ITS SR 3.4.12.2.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-01 R	Relocates "Charging Pumps - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 3. - relocated to CN ECG	Yes	No, see Amendment 89. Q3.1-8	No, see Amendment 103.
08-02 M	Moves charging pump SR when below 350°F to ITS SR 3.4.12.2 and decreases surveillance frequency to 12 hours from 31 days.	No, already in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-03 LS19	Deletes the method of verificating that charging pumps are incapable of injecting into the RCS.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
05-07 LS24	Insert	No, see CN 05-05-LS17	No, see CN 05-05-LS17	Yes	Yes TR 3.1-003
08-04 A	Insert	Yes	No, see CN 8-02-M	No, see Amendment 89	No, see Amendment 103 Q3.1-9

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-7 **APPLICABILITY:** DC, CP

REQUEST:

CTS 3.1.2.2 Flow Parth - Operating (Comanche Peak & Diablo Canyon)
DOC 07-01-R

Comment: The DOC needs to specify where the CTS specification is being relocated. Correct the DOC. A relocated screening form is not provided for this relocated specification.

FLOG RESPONSE: For CPSES, DOC 07-01-R has also been revised to provide additional justification for the relocation.

The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specifications branch reviewers during telephone calls on June 25 and June 30, 1998.

For Diablo Canyon, License Amendment (LA) 120/118 (dated 2/3/98) eliminates this RAI. LA 120/118 relocated ten TSs in accordance with 10 CFR 50.36. Thus, CTS 3.1.2.2 no longer exists and DOC 07-01-R is not applicable to DCPP. CTS 3.1.2.2 was relocated to ECG 8.4, which is part of an NRC approved program controlled by 10 CFR 50.59, and discussed in FSAR Chapter 16.

ATTACHED PAGES:

Encl. 2	3/4 1-8
Encl. 3A	5
Encl. 3B	4

Q3.1-7

07-01-R

~~REACTIVITY CONTROL SYSTEMS~~

~~FLOW PATHS - OPERATING~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:~~

~~a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System (RCS), and~~

~~b. The flow path from the refueling water storage tank via a charging pump to the RCS.~~

~~APPLICABILITY: MODES 1, 2, 3 and 4#.~~

~~ACTION:~~

~~a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1%, ΔK/k at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.~~

~~b. With the flow path from the refueling water storage tank inoperable, restore the flow path to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:~~

~~a. At least once per 7 days by verifying that the temperature of the flow path from the boric acid tanks is greater than or equal to 65°F.~~

~~b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.~~

~~c. At least once per 18 months by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal, and~~

~~d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a delivers at least 30 gpm to the RCS.~~

~~#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F.~~

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
05-04	A	CTS SR requires a SDM verification prior to operation above 5 percent power after each refueling with the control rod banks at maximum insertion limits. SDM in MODES 1 and 2 is determined by shutdown and control rods maintained at their insertion limits. The relevant requirements regarding the adequacy of the SDM with rods at their insertion limits is determined through compliance with ITS 3.1.2, which requires a reactivity balance prior to entering MODE 1.after each refueling; and ITS SR 3.1.6.1, which requires a verification of control bank position within insertion limits within 4 hours prior to criticality. Therefore, the requirements of this SR would be performed by other specifications in the ITS. [ ]
05-05	LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown. This is consistent with NUREG-1431. as modified by TSTF-142 TR 3.1-003
05-06	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Insert Q3.1-4
05-07	LS24	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). TR 3.1-003
06-01	R	The CTS 3.1.2.1, "Boration Flow Path Shutdown," and associated SR 4.1.2.1 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-6
07-01	R	The CTS 3.1.2.2, "Boration Flow Path Operating," and associated SR 4.1.2.2 are relocated outside of the TS. This is consistent with NUREG-1431. Q3.1-7 Not Applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-01	R	The CTS 3.1.2.3, "Charging Pump Shutdown," and associated SRs 4.1.2.3.1 and 4.1.2.3.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-8
08-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS19	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-04	A	Insert Q3.1-9
09-01	R	The CTS 3.1.2.4, "Charging Pump Operating," and associated SRs 4.1.2.4.1 and 4.1.2.4.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-10
10-01	R	The CTS 3.1.2.5, "Borated Water Source Shutdown," and associated CTS SR 4.1.2.5 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-11
11-01	R	The CTS 3.1.2.6, "Borated Water Source Operating," and associated CTS SR 4.1.2.6 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-12

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-05 LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown.	Yes	Yes	No, already in CTS.	No, already in CTS.
05-06 A	CTS SR 4.1.1.5.1 requires that the predicted reactivity values "shall" be adjusted (normalized) at 60 EFPD after refueling. ITS SR 3.1.2.1 states the normalization requirement as "may" be adjusted: This is to recognize that normalization is not necessary if predicted and measured core reactivity are within acceptance tolerance. The scheduling of predicted and measured core reactivity continues to be required at 60 EFPD. Therefore, this change reflects clarification of existing intent and is considered administrative.	~~No, maintaining CTS wording.~~ Yes	~~No, maintaining CTS wording.~~ Yes	Yes Q3.1-4	Yes
06-01 R	Relocates "Boration Flow Path - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 1, relocated to CN ECG	Yes	No, see Amendment 89. Q3.1-6	No, see Amendment 103.
07-01 R	Relocates "Boration Flow Path - Operating" TS to licensee controlled document.	~~Yes, see LAR 95 07 dated 10/4/95, DCL 95-222.~~ No, see Amendments 120/118	Yes	No, see Amendment 89. Q3.1-7	No, see Amendment 103.
07-02 A	Moves limitation on charging pumps in MODE 4 to ITS SR 3.4.12.2.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-01 R	Relocates "Charging Pumps - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 3, relocated to CN ECG	Yes	No, see Amendment 89. Q3.1-8	No, see Amendment 103.
08-02 M	Moves charging pump SR when below 350°F to ITS SR 3.4.12.2 and decreases surveillance frequency to 12 hours from 31 days.	No, already in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-03 LS19	Deletes the method of verificating that charging pumps are incapable of injecting into the RCS.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
05-07 LS24	Insert	No, see CN 05-05-LS17	No, see CN 05-05-LS17	Yes	Yes TR 3.1-003
08-04 A	Insert	Yes	No, see CN 8-02-M	No, see Amendment 89	No, see Amendment 103 Q3.1-9

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-8 **APPLICABILITY:** DC, CP

REQUEST:

CTS 3.1.2.3 Charging Pump - Shutdown (Comanche Peak & Diablo Canyon)
DOC 08-01-R

Comment: The DOC needs to specify where the CTS specification is being relocated. Correct the DOC.

FLOG RESPONSE: DOC 08-01-R is revised and Technical Specification Screening Form for CTS 3.1.2.3 prepared to provide additional justification for the relocation. This justification shows that the charging pump system is not assumed to operate to mitigate any accident. The response for a boron dilution event would be to secure appropriate valves prior to loss of SDM. Since the system does not mitigate an accident, there is no installed instrumentation which is used to detect or indicate a significant degradation of the RCS boundary. This system is also not associated with any variable, design feature, or operating limit that is the initial condition of an event which challenges a fission product boundary. The boration system is not a part of, nor does it support a system requiring that support, to function as part of the success path to mitigate a design bases accident.

The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specifications branch reviewers during telephone calls on June 25 and June 30, 1998.

ATTACHED PAGES:

Encl. 3A	5
Encl. 3B	4

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
05-04	A	CTS SR requires a SDM verification prior to operation above 5 percent power after each refueling with the control rod banks at maximum insertion limits. SDM in MODES 1 and 2 is determined by shutdown and control rods maintained at their insertion limits. The relevant requirements regarding the adequacy of the SDM with rods at their insertion limits is determined through compliance with ITS 3.1.2, which requires a reactivity balance prior to entering MODE 1.after each refueling; and ITS SR 3.1.6.1, which requires a verification of control bank position within insertion limits within 4 hours prior to criticality. Therefore, the requirements of this SR would be performed by other specifications in the ITS. [ ]
05-05	LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown. This is consistent with NUREG-1431, as modified by TSTF-142 TR 3.1-003
05-06	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Insert Q3.1-4
05-07	LS24	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). TR 3.1-003
06-01	R	The CTS 3.1.2.1, "Boration Flow Path Shutdown," and associated SR 4.1.2.1 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-6
07-01	R	The CTS 3.1.2.2, "Boration Flow Path Operating," and associated SR 4.1.2.2 are relocated outside of the TS. This is consistent with NUREG-1431. Q3.1-7 Not Applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-01	R	The CTS 3.1.2.3, "Charging Pump Shutdown," and associated SRs 4.1.2.3.1 and 4.1.2.3.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-8
08-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS19	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-04	A	Insert Q3.1-9
09-01	R	The CTS 3.1.2.4, "Charging Pump Operating," and associated SRs 4.1.2.4.1 and 4.1.2.4.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-10
10-01	R	The CTS 3.1.2.5, "Borated Water Source Shutdown," and associated CTS SR 4.1.2.5 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-11
11-01	R	The CTS 3.1.2.6, "Borated Water Source Operating," and associated CTS SR 4.1.2.6 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-12

The charging pumps are components within the Boration System. The function of the Boration System is to ensure that negative reactivity control is available during each mode of facility operation.

This proposed TS revision relocates requirements, which do not meet the TS criteria in 10CFR50.36 (c) (2) (ii), to documents with established control programs. This regulation addresses the scope and purpose of the TS. In doing so, It sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. Relocation of these requirements allow the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS. An elevation of the applicability of these criteria to this specification is provided in Attachment 21.

To ensure an appropriate level of control, these requirements will be relocated to 1) documents that are subject to the provisions of 10CFR50.59, 2) other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in FSAR which is controlled by 10CFR50.54a), or3) to programs that are controlled via the Administrative Controls section of the improved TS. The identification of the specific licensee controlled document containing this requirement is provided in Enclosure 3B of the conversion submittal. This format for specifying the location of relocated requirements was found to be acceptable to the NRC technical specification branch reviewers during telephone calls on June 25 and June 30, 1998.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure the limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-05 LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown.	Yes	Yes	No, already in CTS.	No, already in CTS.
05-06 A	CTS SR 4.1.1.5.1 requires that the predicted reactivity values "shall" be adjusted (normalized) at 60 EFPD after refueling. ITS SR 3.1.2.1 states the normalization requirement as "may" be adjusted: This is to recognize that normalization is not necessary if predicted and measured core reactivity are within acceptance tolerance. The scheduling of predicted and measured core reactivity continues to be required at 60 EFPD. Therefore, this change reflects clarification of existing intent and is considered administrative.	~~No, maintaining CTS wording.~~ Yes	~~No, maintaining CTS wording.~~ Yes	Yes Q3.1-4	Yes
06-01 R	Relocates "Boration Flow Path - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 1, relocated to CN ECG	Yes	No, see Amendment 89. Q3.1-6	No, see Amendment 103.
07-01 R	Relocates "Boration Flow Path - Operating" TS to licensee controlled document.	~~Yes, see LAR 95 07 dated 10/4/95, DCL 95-222.~~ No, see Amendments 120/118	Yes	No, see Amendment 89. Q3.1-7	No, see Amendment 103.
07-02 A	Moves limitation on charging pumps in MODE 4 to ITS SR 3.4.12.2.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-01 R	Relocates "Charging Pumps - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 3, relocated to CN ECG	Yes	No, see Amendment 89. Q3.1-8	No, see Amendment 103.
08-02 M	Moves charging pump SR when below 350°F to ITS SR 3.4.12.2 and decreases surveillance frequency to 12 hours from 31 days.	No, already in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-03 LS19	Deletes the method of verificating that charging pumps are incapable of injecting into the RCS.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
05-07 LS24	Insert	No, see CN 05-05-LS17	No, see CN 05-05-LS17	Yes	Yes TR 3.1-003
08-04 A	Insert	Yes	No, see CN 8-02-M	No, see Amendment 89	No, see Amendment 103 Q3.1-9

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-9 **APPLICABILITY:** DC

REQUEST:

DCPP SR (on charging pump operability verification) (Diablo Canyon)
DOC 08-02-M

Comment: Conversion Comparison Table indicates that the SR on charging pump operability verification is "already in CTS," and the DOC does not specify where. Identify in an updated DOC where the SR is in the CTS and where it will appear in the ITS.

FLOG RESPONSE: DCPP CTS SR 4.1.2.3.2 already contains the conditions revised by DOC 08-02-M. DCPP CTS SR 4.1.2.3.2 has been unstruck and DOC 08-04-A will be issued to address the administrative change of moving CTS SR 4.1.2.3.2 to ITS SR 3.4.12.2. DOC 08-04-A states: "The MODE 5 and 6 requirements of CTS SR 4.1.2.3.2 are moved by this change to ITS SR 3.4.12.2. Since there are no technical changes (either actual or interpretational) being made, this change is considered administrative (A) in nature."

ATTACHED PAGES:

Encl. 2	3/4 1-10
Encl. 3A	5
Encl. 3B	4

~~CHARGING PUMP - SHUTDOWN~~

08-01-R

~~LIMITING CONDITION FOR OPERATION~~

~~3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.~~

~~APPLICABILITY: MODES 5 and 6.~~

~~ACTION:~~

~~With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.1.2.3.1 At least the above required charging pump shall be demonstrated OPERABLE when tested pursuant to Specification 4.0.5. In addition, when the above required charging pump is a centrifugal charging pump, verify that, on recirculation flow, the centrifugal charging pump develops a differential pressure of greater than or equal to 2400 psid.~~

~~4.1.2.3.2 All centrifugal charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable* at least once per 12 hours, except when the reactor vessel head is removed, by verifying that the motor breaker D.C. control power is de-energized.~~

unstrike

08-04-A

Ø3.1-9

~~*An inoperable pump may be made OPERABLE for testing per Specification 4.0.5 provided the discharge of the pump has been isolated from the Reactor Coolant System by an isolation valve with power removed from the valve operator, or by a sealed closed manual isolation valve.~~

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
05-04	A	CTS SR requires a SDM verification prior to operation above 5 percent power after each refueling with the control rod banks at maximum insertion limits. SDM in MODES 1 and 2 is determined by shutdown and control rods maintained at their insertion limits. The relevant requirements regarding the adequacy of the SDM with rods at their insertion limits is determined through compliance with ITS 3.1.2, which requires a reactivity balance prior to entering MODE 1.after each refueling; and ITS SR 3.1.6.1, which requires a verification of control bank position within insertion limits within 4 hours prior to criticality. Therefore, the requirements of this SR would be performed by other specifications in the ITS. [ ]
05-05	LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown. This is consistent with NUREG-1431, as modified by TSTF-142 TR 3.1-003
05-06	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Insert Q3.1-4
05-07	LS24	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). TR 3.1-003
06-01	R	The CTS 3.1.2.1, "Boration Flow Path Shutdown," and associated SR 4.1.2.1 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-6
07-01	R	The CTS 3.1.2.2, "Boration Flow Path Operating," and associated SR 4.1.2.2 are relocated outside of the TS. This is consistent with NUREG-1431. Q3.1-7 Not Applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-01	R	The CTS 3.1.2.3, "Charging Pump Shutdown," and associated SRs 4.1.2.3.1 and 4.1.2.3.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-8
08-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS19	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-04	A	Insert Q3.1-9
09-01	R	The CTS 3.1.2.4, "Charging Pump Operating," and associated SRs 4.1.2.4.1 and 4.1.2.4.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-10
10-01	R	The CTS 3.1.2.5, "Borated Water Source Shutdown," and associated CTS SR 4.1.2.5 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-11
11-01	R	The CTS 3.1.2.6, "Borated Water Source Operating," and associated CTS SR 4.1.2.6 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-12

DOC 08-04-A

The MODE 5 and 6 requirements of CTS SR 4.1.2.3.2 are moved by this change to ITS SR 3.4.12.2. Since there are not technical changes (either actual or interpretational) being made, this change is considered administrative (A) in nature."

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-05 LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown.	Yes	Yes	No, already in CTS.	No, already in CTS.
05-06 A	CTS SR 4.1.1.5.1 requires that the predicted reactivity values "shall" be adjusted (normalized) at 60 EFPD after refueling. ITS SR 3.1.2.1 states the normalization requirement as "may" be adjusted: This is to recognize that normalization is not necessary if predicted and measured core reactivity are within acceptance tolerance. The scheduling of predicted and measured core reactivity continues to be required at 60 EFPD. Therefore, this change reflects clarification of existing intent and is considered administrative.	~~No, maintaining CTS wording.~~ Yes	~~No, maintaining CTS wording.~~ Yes	Yes Q3.1-4	Yes
06-01 R	Relocates "Boration Flow Path - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 1, relocated to cn ECG	Yes	No, see Amendment 89. Q3.1-6	No, see Amendment 103.
07-01 R	Relocates "Boration Flow Path - Operating" TS to licensee controlled document.	~~Yes, see LAR 95 07 dated 10/4/95, DCL 95-222.~~	Yes No, see Amendments 120/118	No, see Amendment 89. Q3.1-7	No, see Amendment 103.
07-02 A	Moves limitation on charging pumps in MODE 4 to ITS SR 3.4.12.2.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-01 R	Relocates "Charging Pumps - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 3, relocated to cn ECG	Yes	No, see Amendment 89. Q3.1-8	No, see Amendment 103.
08-02 M	Moves charging pump SR when below 350°F to ITS SR 3.4.12.2 and decreases surveillance frequency to 12 hours from 31 days.	No, already in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-03 LS19	Deletes the method of verificating that charging pumps are incapable of injecting into the RCS.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
05-07 LS24	Insert	No, see CN 05-05-LS17	No, see CN 05-05-LS17	Yes	Yes TR 3.1-003
08-04 A	Insert	Yes	No, see CN 8-02-M	No, see Amendment 89	No, see Amendment 103 Q3.1-9

DOC 08-04-A

The MODE 5 and 6 requirements of CTS SR 4.1.2.3.2 are moved by this change to ITS SR 3.4.12.2. Since there are not technical changes (either actual or interpretational) being made, this change is considered administrative (A) in nature."

Applicability:

DC	Yes
CP	No, See CN 8-02-M
WC	No, See Amendment 89
CA	No, See Amendment 103

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-10 **APPLICABILITY:** DC, CP

REQUEST:

CTS 3.1.2.4 Charging Pump - Operating (Comanche Peak & Diablo Canyon)
DOC 09-01-R

Comment: The DOC needs to specify where the CTS specification is being relocated. Correct the DOC.

FLOG RESPONSE: DOC 09-01-R is revised and Technical Specification Screening Form for CTS 3.1.2.4 prepared to provide additional justification for the relocation. CTS 3.1.2.4 is provided to assure negative reactivity control during MODES 1, 2, 3, and 4 operation. This justification shows that the charging pump system is not assumed to operate to provide negative reactivity control to mitigate any accident. The response for a boron dilution event would be to secure appropriate valves in the reactor makeup system. Since the system does not mitigate an accident, there is no installed instrumentation which is used to detect or indicate a significant degradation of the RCS boundary. This system is also not associated with any variable, design feature, or operating limit that is the initial condition of an event which challenges a fission product boundary. This subsystem in the boration system is not a part of, nor does it support a system requiring that support, to function as part of the success path to mitigate a design bases accident.

The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specifications branch reviewers during telephone calls on June 25 and June 30, 1998.

ATTACHED PAGES:

Encl. 3A	5
Encl. 3B	5

CHANGE NUMBER	NSHC	DESCRIPTION
05-04	A	CTS SR requires a SDM verification prior to operation above 5 percent power after each refueling with the control rod banks at maximum insertion limits. SDM in MODES 1 and 2 is determined by shutdown and control rods maintained at their insertion limits. The relevant requirements regarding the adequacy of the SDM with rods at their insertion limits is determined through compliance with ITS 3.1.2, which requires a reactivity balance prior to entering MODE 1.after each refueling; and ITS SR 3.1.6.1, which requires a verification of control bank position within insertion limits within 4 hours prior to criticality. Therefore, the requirements of this SR would be performed by other specifications in the ITS. [ ]
05-05	LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown. This is consistent with NUREG-1431, as modified by TSTF-142 TR 3.1-003
05-06	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Insert Q3.1-4
05-07	LS24	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). TR 3.1-003
06-01	R	The CTS 3.1.2.1, "Boration Flow Path Shutdown," and associated SR 4.1.2.1 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-6
07-01	R	The CTS 3.1.2.2, "Boration Flow Path Operating," and associated SR 4.1.2.2 are relocated outside of the TS. This is consistent with NUREG-1431. Q3.1-7 Not Applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-01	R	The CTS 3.1.2.3, "Charging Pump Shutdown," and associated SRs 4.1.2.3.1 and 4.1.2.3.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-8
08-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS19	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-04	A	Insert Q3.1-9
09-01	R	The CTS 3.1.2.4, "Charging Pump Operating," and associated SRs 4.1.2.4.1 and 4.1.2.4.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-10
10-01	R	The CTS 3.1.2.5, "Borated Water Source Shutdown," and associated CTS SR 4.1.2.5 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-11
11-01	R	The CTS 3.1.2.6, "Borated Water Source Operating," and associated CTS SR 4.1.2.6 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-12

The chanrging pumps are componenets within the Boration System. The function of the Boration System is to ensure that negative reactivity control is available during each mode of facility operation.

This proposed TS revision relocates requirements, which do not meet the TS criteria in 10CFR50.36 (c) (2) (ii), to documents with established control programs. This regulation addresses the scope and purpose of the TS. In doing so, It sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. Relocation of these requirements allow the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS. An elevation of the applicability of these criteria to this specification is provided in Attachment 21.

To ensure an appropriate level of control, these requirements will be relocated to 1) documents that are subject to the provisions of 10CFR50.59, 2) other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in FSAR which is controlled by 10CFR50.54a), or3) to programs that are controlled via the Administrative Controls section of the improved TS. The identification of the specific licensee controlled document containing this requirement is provided in Enclosure 3B of the conversion submittal. This format for specifying the location of relocated requirements was found to be acceptable to the NRC technical specification branch reviewers during telephone calls on June 25 and June 30, 1998.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure the limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

-relocated to an ECG

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
09-01 R	Relocates "Charging Pumps - Operating" TS to licensee controlled document.	Yes, see Attachment 21, Page 5	Yes Q3.1-10	No, see Amendment 89.	No, see Amendment 103.
10-01 R	Relocates "Borated Water Source - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 7	Yes Q3.1-11	No, see Amendment 89.	No, see Amendment 103.
11-01 R	Relocates "Borated Water Source - Operating" TS to licensee controlled document.	Yes, see Attachment 21, Page 9	Yes Q3.1-12	No, see Amendment 89.	No, see Amendment 103.
12-01 A	The reference to "full-length" rods would be deleted.	Yes	No, not in CTS.	Yes	Yes
12-02 M	The ACTIONS required for more than one misaligned, but operable, rod would be changed to be identical to those for inoperable rods.	Yes	Yes	Yes	Yes
12-03 A	The requirement to include an increased allowance in the SDM calculation for the untrippable control rod is inherent in the SDM Definition.	Yes	Yes	Yes	Yes
12-04 A	ACTION for a misaligned rod would be modified to eliminate the notion that a misaligned rod is, by definition, inoperable.	Yes	Yes	Yes	Yes
12-05	Not used.	N/A	N/A	N/A	N/A
12-06 A	ACTION for a misaligned rod would be modified to require boration to restore SDM if not within limits.	Yes	Yes	Yes	Yes
12-07 A	Table 3.1-1, "Accident Analyses Requiring Reevaluation in the Event of an "Inoperable [Full-Length] Rod" would be eliminated.	Yes	Yes	Yes	Yes
12-08 LS9	The requirement to reduce the high neutron flux set point to ≤ 85% of RTP would be deleted.	Yes	Yes	Yes	Yes

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-11 **APPLICABILITY:** DC, CP

REQUEST:

CTS 3.1.2.5 Borated Water Source - Shutdown (Comanche Peak & Diablo Canyon)
DOC 10-01-R

Comment: The DOC needs to specify where the CTS specification is being relocated. Correct the DOC.

FLOG RESPONSE: DOC 10-01-R is revised and Technical Specification Screening Form for CTS 3.1.2.5 prepared to provide additional justification for the relocation. This justification shows that the boric acid storage system is not assumed to operate to provide negative reactivity control to mitigate any accident. The response for a boron dilution event would be to secure appropriate valves in the reactor makeup system. Since the system does not mitigate an accident, there is no installed instrumentation which is used to detect or indicate a significant degradation of the RCS boundary. This system is also not associated with any variable, design feature, or operating limit that is the initial condition of an event which challenges a fission product boundary. This subsystem in the boration system is not a part of, nor does it support a system requiring that support, to function as part of the success path to mitigate a design bases accident.

The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specifications branch reviewers during telephone calls on June 25 and June 30, 1998.

ATTACHED PAGES:

Encl. 3A	5
Encl. 3B	5

CHANGE NUMBER	NSHC	DESCRIPTION
05-04	A	CTS SR requires a SDM verification prior to operation above 5 percent power after each refueling with the control rod banks at maximum insertion limits. SDM in MODES 1 and 2 is determined by shutdown and control rods maintained at their insertion limits. The relevant requirements regarding the adequacy of the SDM with rods at their insertion limits is determined through compliance with ITS 3.1.2, which requires a reactivity balance prior to entering MODE 1.after each refueling; and ITS SR 3.1.6.1, which requires a verification of control bank position within insertion limits within 4 hours prior to criticality. Therefore, the requirements of this SR would be performed by other specifications in the ITS. [ ]
05-05	LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown. This is consistent with NUREG-1431, as modified by TSTF-142 TR 3.1-003
05-06	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Insert Q3.1-4
05-07	LS24	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). TR 3.1-003
06-01	R	The CTS 3.1.2.1, "Boration Flow Path Shutdown," and associated SR 4.1.2.1 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-6
07-01	R	The CTS 3.1.2.2, "Boration Flow Path Operating," and associated SR 4.1.2.2 are relocated outside of the TS. This is consistent with NUREG-1431. Q3.1-7 Not Applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-01	R	The CTS 3.1.2.3, "Charging Pump Shutdown," and associated SRs 4.1.2.3.1 and 4.1.2.3.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-8
08-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS19	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-04	A	Insert Q3.1-9
09-01	R	The CTS 3.1.2.4, "Charging Pump Operating," and associated SRs 4.1.2.4.1 and 4.1.2.4.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-10
10-01	R	The CTS 3.1.2.5, "Borated Water Source Shutdown," and associated CTS SR 4.1.2.5 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-11
11-01	R	The CTS 3.1.2.6, "Borated Water Source Operating," and associated CTS SR 4.1.2.6 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-12

The borated water source is a component within the Boration System. The function of the Boration System is to ensure that negative reactivity control is available during each mode of facility operation.

This proposed TS revision relocates requirements, which do not meet the TS criteria in 10CFR50.36 (c) (2) (ii), to documents with established control programs. This regulation addresses the scope and purpose of the TS. In doing so, It sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. Relocation of these requirements allow the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS. An elevation of the applicability of these criteria to this specification is provided in Attachment 21.

To ensure an appropriate level of control, these requirements will be relocated to 1) documents that are subject to the provisions of 10CFR50.59, 2) other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in FSAR which is controlled by 10CFR50.54a), or3) to programs that are controlled via the Administrative Controls section of the improved TS. The identification of the specific licensee controlled document containing this requirement is provided in Enclosure 3B of the conversion submittal. This format for specifying the location of relocated requirements was found to be acceptable to the NRC technical specification branch reviewers during telephone calls on June 25 and June 30, 1998.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure the limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

-relocated to an ECG

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
09-01 R	Relocates "Charging Pumps - Operating" TS to licensee controlled document.	Yes, see Attachment 21, Page 5	Yes Q3.1-10	No, see Amendment 89.	No, see Amendment 103.
10-01 R	Relocates "Borated Water Source - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 7.	Yes Q3.1-11	No, see Amendment 89.	No, see Amendment 103.
11-01 R	Relocates "Borated Water Source - Operating" TS to licensee controlled document.	Yes, see Attachment 21, Page 9	Yes Q3.1-12	No, see Amendment 89.	No, see Amendment 103.
12-01 A	The reference to "full-length" rods would be deleted.	Yes	No, not in CTS.	Yes	Yes
12-02 M	The ACTIONS required for more than one misaligned, but operable, rod would be changed to be identical to those for inoperable rods.	Yes	Yes	Yes	Yes
12-03 A	The requirement to include an increased allowance in the SDM calculation for the untrippable control rod is inherent in the SDM Definition.	Yes	Yes	Yes	Yes
12-04 A	ACTION for a misaligned rod would be modified to eliminate the notion that a misaligned rod is, by definition, inoperable.	Yes	Yes	Yes	Yes
12-05	Not used.	N/A	N/A	N/A	N/A
12-06 A	ACTION for a misaligned rod would be modified to require boration to restore SDM if not within limits.	Yes	Yes	Yes	Yes
12-07 A	Table 3.1-1, "Accident Analyses Requiring Reevaluation in the Event of an "Inoperable [Full-Length] Rod" would be eliminated.	Yes	Yes	Yes	Yes
12-08 LS9	The requirement to reduce the high neutron flux set point to ≤ 85% of RTP would be deleted.	Yes	Yes	Yes	Yes

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-12 **APPLICABILITY:** DC, CP

REQUEST:

CTS 3.1.2.5 Borated Water Source - Operating (Comanche Peak & Diablo Canyon)
DOC 11-01-R

Comment: The DOC needs to specify where the CTS specification is being relocated. Correct the DOC.

FLOG RESPONSE: DOC 11-01-R is revised and Technical Specification Screening Form for CTS 3.1.2.6 prepared to provide additional justification for the relocation. This justification shows that the boric acid storage system is not assumed to operate to provide negative reactivity control to mitigate any accident. The response for a boron dilution event would be to secure appropriate valves in the reactor makeup system. The SDM requirements provide sufficient reactivity margin to mitigate any anticipated event. Since the system does not mitigate an accident, there is no installed instrumentation which is used to detect or indicate a significant degradation of the RCS boundary. This system is also not associated with any variable, design feature, or operating limit that is the initial condition of an event which challenges a fission product boundary. This subsystem in the boration system is not a part of, nor does it support a system requiring that support, to function as part of the success path to mitigate a design bases accident.

The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specifications branch reviewers during telephone calls on June 25 and June 30, 1998.

ATTACHED PAGES:

Encl. 3A	5
Encl. 3B	5

CHANGE NUMBER	NSHC	DESCRIPTION
05-04	A	CTS SR requires a SDM verification prior to operation above 5 percent power after each refueling with the control rod banks at maximum insertion limits. SDM in MODES 1 and 2 is determined by shutdown and control rods maintained at their insertion limits. The relevant requirements regarding the adequacy of the SDM with rods at their insertion limits is determined through compliance with ITS 3.1.2, which requires a reactivity balance prior to entering MODE 1.after each refueling; and ITS SR 3.1.6.1, which requires a verification of control bank position within insertion limits within 4 hours prior to criticality. Therefore, the requirements of this SR would be performed by other specifications in the ITS. [ ]
05-05	LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown. This is consistent with NUREG-1431, as modified by TSTF-142 TR 3.1-003
05-06	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Insert Q3.1-4
05-07	LS24	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). TR 3.1-003
06-01	R	The CTS 3.1.2.1, "Boration Flow Path Shutdown," and associated SR 4.1.2.1 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-6
07-01	R	The CTS 3.1.2.2, "Boration Flow Path Operating," and associated SR 4.1.2.2 are relocated outside of the TS. This is consistent with NUREG-1431. Q3.1-7 Not Applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-01	R	The CTS 3.1.2.3, "Charging Pump Shutdown," and associated SRs 4.1.2.3.1 and 4.1.2.3.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-8
08-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS19	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-04	A	Insert Q3.1-9
09-01	R	The CTS 3.1.2.4, "Charging Pump Operating," and associated SRs 4.1.2.4.1 and 4.1.2.4.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-10
10-01	R	The CTS 3.1.2.5, "Borated Water Source Shutdown," and associated CTS SR 4.1.2.5 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-11
11-01	R	The CTS 3.1.2.6, "Borated Water Source Operating," and associated CTS SR 4.1.2.6 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-12

The borated water source is a component within the Boration System. The function of the Boration System is to ensure that negative reactivity control is available during each mode of facility operation.

This proposed TS revision relocates requirements, which do not meet the TS criteria in 10CFR50.36 (c) (2) (ii), to documents with established control programs. This regulation addresses the scope and purpose of the TS. In doing so, It sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. Relocation of these requirements allow the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS. An elevation of the applicability of these criteria to this specification is provided in Attachment 21.

To ensure an appropriate level of control, these requirements will be relocated to 1) documents that are subject to the provisions of 10CFR50.59, 2) other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in FSAR which is controlled by 10CFR50.54a), or3) to programs that are controlled via the Administrative Controls section of the improved TS. The identification of the specific licensee controlled document containing this requirement is provided in Enclosure 3B of the conversion submittal. This format for specifying the location of relocated requirements was found to be acceptable to the NRC technical specification branch reviewers during telephone calls on June 25 and June 30, 1998.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure the limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

-relocated to an ECG

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
09-01 R	Relocates "Charging Pumps - Operating" TS to licensee controlled document.	Yes, see Attachment 21, Page 5.	Yes Q3.1-10	No, see Amendment 89.	No, see Amendment 103.
10-01 R	Relocates "Borated Water Source - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 7.	Yes Q3.1-11	No, see Amendment 89.	No, see Amendment 103.
11-01 R	Relocates "Borated Water Source - Operating" TS to licensee controlled document.	Yes, see Attachment 21, Page 9.	Yes Q3.1-12	No, see Amendment 89.	No, see Amendment 103.
12-01 A	The reference to "full-length" rods would be deleted.	Yes	No, not in CTS.	Yes	Yes
12-02 M	The ACTIONS required for more than one misaligned, but operable, rod would be changed to be identical to those for inoperable rods.	Yes	Yes	Yes	Yes
12-03 A	The requirement to include an increased allowance in the SDM calculation for the untrippable control rod is inherent in the SDM Definition.	Yes	Yes	Yes	Yes
12-04 A	ACTION for a misaligned rod would be modified to eliminate the notion that a misaligned rod is, by definition, inoperable.	Yes	Yes	Yes	Yes
12-05	Not used.	N/A	N/A	N/A	N/A
12-06 A	ACTION for a misaligned rod would be modified to require boration to restore SDM if not within limits.	Yes	Yes	Yes	Yes
12-07 A	Table 3.1-1, "Accident Analyses Requiring Reevaluation in the Event of an "Inoperable [Full-Length] Rod" would be eliminated.	Yes	Yes	Yes	Yes
12-08 LS9	The requirement to reduce the high neutron flux set point to ≤ 85% of RTP would be deleted.	Yes	Yes	Yes	Yes

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-13 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

ITS 3.1.4 Rod Group Alignment Limits (Comanche Peak)
DOC 12-07-A
ITS 3.1.4 Bases

Comment: The DOC states, for Required Action B.2.6, that "the ITS Bases discuss the accident analysis affected by rod misalignment." The associated Bases do not list the accident analyses that require re-evaluation, similar to that provided by the other Four Loop Group plants. List in the Bases the accident analyses that require re-evaluation.

FLOG RESPONSE: The APPLICABLE SAFETY ANALYSES section of ITS 3.1.4 Bases provides an appropriate description of the various manners in which a misaligned rod can affect the safety analyses. The requirement in ITS 3.1.4 REQUIRED ACTION B 2.6 is to evaluate the safety analyses; the affected analyses are described more fully by the APPLICABLE SAFETY ANALYSES than by the list transported from the CTS. In fact, many of the analyses listed (e.g., Decrease in Reactor Coolant Inventory in USAR Section 15.6) are not affected by reasonable rod misalignments; whereas some transients that are sensitive to misaligned rods (most of the Power Distribution and Reactivity Anomaly accidents described in USAR Section 15.4) are not listed. Because of the potential conflicts between the APPLICABLE SAFETY ANALYSES section and the list of CTS Table 3.1-1, it is preferable to not add the list, but refer to the APPLICABLE SAFETY ANALYSES section. The ITS Bases ACTIONS B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6 are revised to indicate that the accident analysis presented in USAR Chapter 15 that may be adversely affected will be evaluated to ensure that the analyses results remain valid for the duration of continued operation.

Callaway, Wolf Creek, and Diablo Canyon have reviewed this Comment and concur with the above discussion. Their ITS Bases have been revised to delete the list of accident analyses that require re-evaluation and refer to FSAR Chapter 15.

ATTACHED PAGES:

Encl 5B B 3.1-18

BASES

ACTION reduction (continued)

Time of 2 hours reduction gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System. DC3.1-ED

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$. CP 3.1-13

The Accident Analyses of FSAR Chapter 15 are to be used to identify the appropriate design bases events requiring re-evaluation.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

~~The following accident analyses require re-evaluation for continued operation with a misaligned rod:~~

~~Rod Cluster Control Assembly Insertion Characteristics~~

~~Rod Cluster Control Assembly Misalignment~~

~~Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuate the Emergency Core Cooling System~~

~~Single Rod Cluster Control Assembly Withdrawal at Full Power~~

~~Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)~~

~~Major Secondary Coolant System Pipe Rupture~~

~~Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection).~~

CP3.1-13

(Continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-15 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

ITS 3.1.4 Rod Group Alignment Limits
CTS 3/4.1.3 Movable Control Assemblies (All FLOG Plants)
DOC 12-14-M

Comment: The ITS has changed the wording of the TS from "trippability" to "operability," and references TSTF-107 which is not yet approved (though it is expected to be approved with the WOGs next revision of TSTF-107. The result is that the FLOG plants have inconsistently incorporated generic changes into the Bases (i.e., the Bases paragraphs for B.2.1.1 and B.2.1.2). This change is a less restrictive change in that it precludes LCO 3.0.3 entry for unforeseen inoperabilities. TSTF-107 needs to be discussed/approved at the next TSTF OG/NRC Meeting, and the FLOG will then need to incorporate the resulting generic TS requirements.

FLOG RESPONSE: It is the FLOG's understanding that EXCEL Services Corporation met with the NRC on May 23, 1998, to discuss TSTF-107. The result of that meeting has been reported to be agreement to approve TSTF-107 with a minor Bases change. Revision 1 of TSTF-107 has been incorporated into the FLOG submittals.

In the ITS, rod operability is addressed in the Bases as trippability within the drop time requirements of ITS SR 3.1.4.3. If not met, Condition A would be entered which requires SDM verification and shutdown to MODE 3 in 6 hours, which then exits the LCO.

In the CTS, the action for an untrippable rod is essentially the same as the ITS. No action is provided in the CTS for discovering in MODE 1 or 2 that a rod would not meet insertion time requirements; therefore, CTS LCO 3.0.3 would be entered. LCO 3.0.3 allows 1 hour to initiate a shutdown and 6 additional hours to reach MODE 3. Because the ITS only allows 6 hours to reach MODE 3 (instead of up to 7 as allowed by LCO 3.0.3), the change from "untrippable" to "inoperable" in CTS 3.1.3.1 is more restrictive.

ATTACHED PAGES:

Encl. 5A	Traveler Status Sheet
Encl. 5B	B 3.1-14, B 3.1-16, B 3.1-16a, B3.1-17, B,3.1-17a, B 3.1-19a

Industry Travelers Applicable to Section 3.1

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-9, Rev. 1	Incorporated	3.1-1	NRC approved.
TSTF-12, Rev. 1	Incorporated	3.1-15	NRC approved. ITS Special Test Exception 3.1.10 is retained and re-numbered as 3.8.1, consistent with this traveler and TSTF-136.
TSTF-13, Rev. 1	Incorporated	3.1-4	NRC approved.
TSTF-14, Rev. ~~3~~ 4	Incorporated	3.1-13	NRC approved. TR 3.1-005
TSTF-15, Rev. 1	Incorporated	N/A	NRC approved.
TSTF-89	Incorporated	3.1-8	NRC approved.
TSTF-107, Rev. 1	Incorporated	3.1-6	03.1-15
TSTF-108, Rev. 1	~~Not~~ Incorporated	~~N/A~~ 3.1-21	~~Not~~ NRC approved ~~as of cut-off date~~ TR 3.1-001
TSTF-110, Rev. ~~1~~ 2	Incorporated	3.1-10	NRC Approved TR 3.1-004
TSTF-136	Incorporated	3.1-9, 3.1-15	
TSTF-141	Not Incorporated	N/A	Disagree with change; traveler issued after cut-off date
TSTF-142	~~Not~~ Incorporated	~~N/A~~ 3.1-22	TR 3.1-003 ~~Traveler issued after cut-off date.~~ NRC Approved.
~~WOG-73, Rev. 1~~ TSTF 234	Incorporated	3.1-7	TR 3.1-006
WOG-105	Incorporated	3.1-16	

B 3.1 REACTIVITY CONTROL SYSTEMS

B ~~3.1.5~~ 3.1.4 Rod Group Alignment Limits

DC 3.1-ED

BASES

~~e.g.~~ redline

Q3.1-15

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately ⅝ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

(Continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10CFR50.36(c)(2)(ii). DC 3.1-ED ~~the NRC Policy Statement~~

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability ~~to meet SDM~~) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required time assumed in the safety analyses. Rod control malfunctions that result in the inability to move a rod (e.g., rod ~~urgent~~ failures), but do not impact trippability, do not necessarily result in rod inoperability. (lift coil) Q3.1-15

The requirement to maintain the rod alignment to within plus or minus 12 steps of their group step counter demand position is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

The requirement to maintain rod alignment is met by comparing individual rod DRPI indication and bank demand position indication to be within plus or minus 12 steps. If one of these position indicators become inoperable, the conditions of this LCO are still met by compliance with LCO 3.1.7.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

(Continued)

BASES

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are typically fully inserted and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)—~~T_{avg} > 200°F~~," for SDM in MODES 2 with K_{eff}<1.0, 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable ~~(i.e., untrippable)~~, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM.

With an inoperable rod(s), This ACTION provides for verification of SDM, this is most simply accomplished by verifying rod insertion limits are met. Additionally, actions could include calculation of the current SDM and boration to meet limits specified in the COLR or proceed

(Continued)

BASES

ACTION (continued)

to MODE 3. These actions are consistent with those specified in LCO 3.1.5 and LCO 3.1.6.

A rod is considered trippable if it was demonstrated OPERABLE during the last performance of SR 3.1.4.2 and met the rod drop time criteria during the last performance of SR 3.1.4.3.

In this situation, SDM verification must ~~include the worth~~ account for the absence of the negative reactivity of the untrippable rod(s), as well as the rod of maximum worth. redline redline Q3.1-15

A.2

If the ~~untrippable~~ inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. redline

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1

Q 3.1.6-1

When a rod becomes misaligned, it can usually be moved and is still trippable (i.e., OPERABLE). If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction. redline

An alternative to realigning a single misaligned RCCA to the group demand position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO ~~3.1.6~~ 3.1.5, "Shutdown Bank Insertion Limits," and LCO ~~3.1.7~~ 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

(Continued)

BASES

~~moved~~

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be fully inserted and control bank C must be inserted to approximately 100 steps ~~to 115 steps~~. redline @3.1-15

Power operation may continue with one RCCA ~~OPERABLE (i.e. trippable)~~ ~~but~~ misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration. DC 3.1-ED

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, ~~RTP~~ reactor power must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible. Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 3). The Completion

(Continued)

BASES

SR ~~3.1.5.2~~ 3.1.4.2

Verifying each rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each rod would result in radial or axial power tilts, or oscillations. Exercising each individual rod every 92 days provides confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR ~~3.1.5.1~~ 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between or during required performances of SR ~~3.1.5.2~~ 3.1.4.2 (determination of rod OPERABILITY by movement), if a rod(s) is discovered to be immovable, but remains trippable, the rod(s) is considered to be OPERABLE. At any time, if a rod(s) is immovable, a determination of the trippability (OPERABILITY) of the rod(s) must be made, and appropriate action taken.

redline Q3.1.G-1 Q3.1.G-1 control and aligned Q3.1-15

SR ~~3.1.5.3~~ 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions.

~~Control~~ Q3.1.G-1

(Continued)

DC 3.1-ED

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-16 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

ITS 3.1.4 Rod Group Alignment Limits (All FLOG Plants)
ITS 3.1.4 Bases
Generic Changes

Comment: Generic Bases changes need to be discussed/justified. For example, the Bases Background discussion on the DRPI system has been revised and needs to be explained.

FLOG RESPONSE: As discussed during a telecon with NRC Staff on June 25, 1998, the scope of this RAI will be limited to the ITS 3.1.4 Background Bases. Changes fall into one of five categories:

1. Specification re-numbering.
2. Inclusion of shutdown rods.
3. Addition of plant-specific design information (e.g., number of control banks and shutdown banks).
4. Editorial corrections (e.g., the correct title for GDC-26); and
5. Changes to the last paragraph.

Changes to the last paragraph were made since it was felt that this text went beyond the level of detail required for the ITS Bases. Coil spacing dimensions are not critical to operator understanding of this system. In addition, statements in the last paragraph in the ISTS concerning position indication accuracies are incorrect.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-20 **APPLICABILITY:** DC, CP

REQUEST:

ITS 3.1.7 Rod Position Indication
CTS 3.1.3.2 Position Indication Systems - Operating (Comanche Peak & Diablo Canyon)
DOC 13-08-LS-20 & 13-09-LS-23 & 13-06-A
JFD 3.1-7 & 3.1-12

Comment: The ITS adopts Conditions and associated Required Actions from the Callaway's CTS, addressing more than one inoperable digital rod position indicator (DRPI) per group, which is not addressed in either the STS or the CTS. Furthermore, not all associated CTS Required Actions have been retained in the ITS; the Required Actions to take manual control of the rods and to record reactor coolant temperature every hour have not been retained. These actions, in one case affect rod movement and in the other case provide an indication that the rod(s) position may have changed, and therefore have a bearing on SDM and therefore should not be deleted if the Callaway condition of more than DRPI per group inoperable is retained. Either retain the CTS requirements and adopt the STS requirements, or provide a better justification for the ITS proposals of adopting the Callaway CTS requirements. This change is based upon proposed change WOG-73, Rev 1; which eventually may become a TSTF change request. What is the status of WOG-73, Rev 1? The STS wording of the note permitting separate condition entry should be retained with the STS Conditions and Required Actions.

FLOG RESPONSE: DCPP and CPSES continue pursuing this change pending NRC review of WOG-73, Revision 1, which has been submitted to the NRC as TSTF-234. This proposed change provides a specific set of compensatory actions and a 24 hour AOT for a situation previously unaddressed. Thus, the requirements of LCO 3.0.3 would be applicable and a plant shutdown would be required within 1 hour.

However, the partial loss of RCCA position indication is not an assumed initiating event, nor does it affect the outcome of any analyzed accident. Therefore, a 1 hour shutdown response is considered unnecessary when compared to other situations of similar severity where guidance was provided.

The contribution to plant safety of a well-balanced approach in avoiding unnecessary shutdowns is sufficient to warrant prompt NRC review and approval of TSTF-234. It would also be sufficient to warrant a separate License Amendment Request were the TSTF to be withdrawn from this submittal.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-21 **APPLICABILITY:** DC, CP

REQUEST:

CTS 3.1.3.3 Position Indication Systems - Shutdown (Comanche Peak & Diablo Canyon)
DOC 14-01-R

Comment: The DOC needs to specify where the CTS specification is being relocated. Correct the DOC. A relocated screening form is not provided for this relocated specification.

FLOG RESPONSE: For CPSES, DOC 14-01-R has also been revised to provide additional justification for the relocation and to reference new Attachment 21 containing relocation screening forms.

The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specifications branch reviewers during telephone calls on June 25 and June 30, 1998.

For Diablo Canyon, License Amendment (LA) 120/118 (dated 2/3/98) eliminates this RAI. LA 120/118 relocated ten TSs in accordance with 10 CFR 50.36. Thus, CTS 3.1.3.3 no longer exists and DOC 14-01-R is not applicable to DCPP. CTS 3.1.3.3 was relocated to ECG 41.1 which is part of an NRC approved program controlled by 10CFR50.59 and discussed in FSAR Chapter 16.

ATTACHED PAGES:

Encl. 2	3/4 1-18 and 3/4 1-19
Encl. 3A	10
Encl. 3B	8

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE ~~and capable of determining the control rod positions within ± 12 steps~~. 13-01-LG

APPLICABILITY: MODES 1* and 2*. 13-08-LS20

ACTION:

a. With a maximum of one digital rod position indicator per ~~bank~~ group inoperable for one or more groups either: 13-02-LS15

1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and ~~immediately~~ within 4 hours after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or 13-03-LS12

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours or be in Hot Standby within the next 6 hours. 13-04-M

b. With more than one digital rod position indicator per group inoperable either:

1.a) Determine the position of the nonindicating rods indirectly by the movable incore detectors at least once per 8 hours and within 4 hours after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, and 13-08-LS20

1.b) Restore the digital rod position indicators to OPERABLE status within 24 hour such that a maximum of one digital rod position indicator per group in inoperable, or 13-08-LS20

2. Be in HOT STANDBY within the next 6 hours. 13-08-LS20

~~b~~ c. With a maximum of one demand position indicator per bank inoperable either:

1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours or be in Hot Standby within the next 6 hours. 13-04-M

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps ~~at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours~~ when exercised over the full range of rod travel once prior to criticality after each removal of the reactor head. 13-07-M 12-16-LG

*Separate condition entry is allowed for each inoperable rod position indicator and each demand position indicator per bank. 13-08-LS20

4.1.3.3 Insert (93.1-21)

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full range of rod travel at least once ~~per each REFUELING INTERVAL~~ prior to criticality after each removal of the reactor head. **13-07-M**

Q 3.1-21

~~REACTIVITY CONTROL SYSTEMS~~

~~POSITION INDICATION SYSTEM - SHUTDOWN~~

14-01-R

~~LIMITING CONDITION FOR OPERATION~~

~~3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.~~

~~APPLICABILITY: MODES 3*#, 4*# and 5*#.~~

~~ACTION:~~

~~With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.~~

~~SURVEILLANCE REQUIREMENTS~~

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full range of rod travel at least once ~~per~~ ~~18 months~~ prior to criticality after each removal of the reactor head. ~~each REFUELING INTERVAL~~ DCA U0001

13-07-M

moved to pg 3/4 1-18

~~*With the Reactor Trip System breakers in the closed position.~~
~~#See Special Test Exceptions Specification 3.10.4~~

14-01-R
03-01-A

CHANGE NUMBER	NSHC	DESCRIPTION
13-06	A	Not applicable to DCPP. See Conversion Comparison Table. (Enclosure 3B)
13-07	M	The proposed modifications to the SR would require a verification of agreement between digital and demand indicator systems prior to criticality after each removal of the reactor vessel head, instead of every 12 hours. This reflects a reorganization of SRs in the ITS. The requirement for a 12 hour comparison would be moved to SR 3.1.4.1 in the ITS. The post-vessel head removal requirement would be a new specification that demonstrates rod position system OPERABILITY based on a comparison of indicating systems throughout the full range of rod travel. The Frequency requirement of prior to criticality after each removal of the reactor vessel head would permit this comparison to be performed only during plant outages that involve plant evolutions (vessel head removal) that could affect the OPERABILITY of the rod position indication systems. The Frequency change is based on Traveler TSTF-89.
13-08	LS20	Adds provision from Callaway's current specifications which would, under certain conditions, allow continued operation with more than one inoperable DRPI per group. A separate Condition entry allowance is permitted for each inoperable rod position indicator per group and each demand position indicator per bank. A proposed Traveler TSTF-234 ~~Westinghouse Owners Group (WOG) - 73, Rev. 1~~ is in processing to cover this issue. TR 3.1-006
13-09	LS23	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Q3.1-21
14-01	R	~~The Shutdown Position Indication System Specification 3.1.3.3 is relocated outside of the TS. This is consistent with NUREG-1431.~~
15-01	R	~~The Rod Drop Time Specification 3.1.3.4 is relocated outside of the TS. The RCS temperature limit and reactor coolant pumps operating requirement for rod drop testing are combined with CTS Surveillance 4.1.3.4, then incorporated into ITS SR 3.1.4.3. This is consistent with NUREG-1431.~~ → Not used Q3.1-22
15-02	A	~~The Rod Drop Time SR 4.1.3.4.a is moved to the Control Rod ITS LCO 3.1.4 as SR 3.1.4.3. This change is consistent with NUREG-1431.~~ Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Q3.1-22
16-01	LS14	This TS would be revised to apply to shutdown "banks" instead of shutdown "rods;" this is consistent with NUREG-1431. The current ACTION statement permits one rod to be inserted beyond the limits; the proposed ITS Condition A would allow one or more banks to be inserted beyond the limit.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
13-04 M	A requirement would be added to bring the plant to MODE 3 within 6 hours if the required ACTIONS and Completion Times are not met.	Yes	Yes	Yes	Yes
13-05 A	The proposed change would retain an ACTION statement, currently in the plant TS, that permits continued POWER OPERATION with more than 1 digital rod position indicator per group inoperable.	No, not in CTS - see 13-08-LS20.	No, not in CTS - see 13-08-LS20.	Yes	Yes
13-06 A	The change would allow separate Condition entry for each inoperable DRPI per group or each demand indicator per bank.	No, not in CTS - see 13-08-LS20.	No, not in CTS - see 13-08-LS20.	Yes	Yes
13-07 M	The proposed modifications to the SR would verify agreement between digital and demand indicator systems prior to criticality after the reactor vessel head was removed instead of every 12 hours. The Frequency change is based on Traveler TSTF-89.	Yes	Yes	Yes	Yes
13-08 LS20	Adds provision in Callaway's current specifications which would, under certain Conditions, allow continued operation with more than one inoperable DRPI per group. This is consistent with Traveler ~~WOG-73, Rev. 1~~. → TSTF-234	Yes	Yes TR3.1-006	No, already in CTS.	No, already in CTS.
13-09 LS23	CTS ACTIONS b.1.b) and b.1.c) of LCO 3.1.3.2 are deleted. SDM is ensured in MODES 1 and 2 by rod position. Multiple inoperable DRPIs will have no impact on SDM in MODES 1 and 2 if the control rod positions are verified by alternate means and rod motion is limited consistent with the accident analyses. Deletion of these requirements is consistent with traveler ~~WOG-73, Rev. 1~~. → TSTF-234	No, not in CTS.	No, not in CTS. TR3.1-006	Yes	Yes
14-01 R	Relocates CTS 3.1.3.3 to licensee controlled documents, consistent with NUREG-1431.	~~Yes, see LAR 95-07 dated 10/4/95, DCL 95-222.~~ → No, see Amendments 120/118.	Yes, relocated to TRM.	No, see Amendment 89.	No, see Amendment 103.

D3.1-21

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-22 **APPLICABILITY:** DC

REQUEST:

CTS 3.1.3.4 Rod Drop Time (Diablo Canyon)
DOC 15-01-R

Comment: The DOC needs to specify where the CTS specification is being relocated. Correct the DOC. A relocated screening form is not provided for this relocated specification.

FLOG RESPONSE: License Amendment (LA) 120/118 (dated 2/3/98) eliminates this RAI. LA 120/118 relocated ten TSs in accordance with 10CFR50.36. Thus, CTS 3.1.3.4 no longer exists and DOC 15-01-R is not applicable to DCPP. CTS 3.1.3.4 was relocated to CTS SR 4.1.3.1.3 (New).

ATTACHED PAGES:

Encl. 2	3/4 1-16, 3/4 1-20
Encl. 3A	8, 10
Encl. 3B	7, 9

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour ~~and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER~~ or be in HOT STANDBY within 6 hours — 12-08-LS9 — 12-22-M

c. ~~With more than one rod trippable but inoperable due to causes other than addressed by ACTION a above, POWER OPERATION may continue provided that:~~ — 12-12-LS13

~~1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and~~

~~2. The inoperable rods are restored to OPERABLE status within 72 hours.~~

d. With more than one rod misaligned from its group demand position by more than ± 12 steps (indicated position), determine that the SHUTDOWN MARGIN is within the COLR or initiate boration to restore the SHUTDOWN MARGIN within limits within one hour, and be in HOT STANDBY within 6 hours. — 01-01-LG — 12-02-M

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each ~~full length~~ rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours ~~except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours~~. — 12-01-A — 12-16-LG

4.1.3.1.2 Each ~~full length~~ rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 92 days. — 12-01-A

Q3.1-22

~~4.1.3.1.3 The rod drop time of rods shall be demonstrated less than or equal to 2.7 seconds from beginning of decay of the stationary gripper coil voltage to dash pot entry with t_{avg} greater than or equal to 500 °F and all reactor coolant pumps operating, prior to criticality after each reactor head removal.~~ — ~~12-10-LS10~~

Prior to reactor criticality, verify that the rod drop time of the ~~full length~~ shutdown and control rods is ≤ 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry, with T_{avg} ≥ ~~541°F~~ 500°F, and all reactor coolant pumps operating: — 12-01-A — 12-10-LS10

a. For all rods following each removal of the reactor vessel head, ~~and~~

~~b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods.~~ — 12-11-TR3

~~REACTIVITY CONTROL SYSTEMS~~

~~ROD DROP TIME~~

~~LIMITING CONDITION FOR OPERATION~~

* relocated to CTS SR 4.1.3.1.3

Q3.1-22

~~15-01-R~~

~~3.1.3.4~~ The individual full-length shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to ~~541~~ 500°F, and

b. All reactor coolant pumps operating.

12-10-LS10

~~APPLICABILITY: MODES 1 and 2.~~

~~ACTION~~:

With the drop time ~~of any full length rod~~ determined to exceed the above limit, restore the rod drop time to within the above limit prior to ~~proceeding to MODE 1 or 2~~ reactor criticality.

12-01-A

~~12-20-A~~

~~SURVEILLANCE REQUIREMENTS~~

12-01-A

~~4.1.3.4~~ The rod drop time ~~of full length rods~~ shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head.

~~15-02-A~~

~~b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and~~

12-11-TR3

~~c. At least once per 18 months~~.

each REFUELING INTERVAL

DC-ALL-001

CHANGE NUMBER	NSHC	DESCRIPTION
12-12	LS13	CTS [3.1.3.1] ACTIONS are revised to delete reference to causes of control rod inoperability due to rod control urgent failure or other electrical problems in the rod control system.
12-13		Not used.
12-14	M	This wording is broadened from "untrippable" to "inoperable" to ensure all causes of inoperability are covered. The previous wording covered inoperable rods if they were untrippable (e.g., "immovable as a result of excessive friction or mechanical interference...") but did not cover trippable rods with drop times that exceed the surveillance limit. These slow rods are inoperable. This more restrictive change clarifies the appropriate ACTIONS to be taken for all causes of inoperability, consistent with Traveler TSTF-107.
12-15	A	Not applicable to DCPP. See Comparison Table (Enclosure 3B).
12-16	LG	Several surveillances (e.g., rod position deviation monitor and rod insertion limit monitor in this section) contain ACTIONS in the form of increased surveillance frequency to be performed in the event of inoperable alarms. These ACTIONS are moved from the CTS to license controlled documents since the alarm does not, in itself, directly relate to the limits. This detail is not required to be in the TS. Therefore, moving this detail is acceptable and is consistent with Traveler TSTF-110. Rev. 1. TR3.1-004
12-17	A	Editorial changes are made for clarity. Untrippable rods are addressed through ACTION A; hence, there is no additional need to exclude those rods from these required ACTIONS.
12-18	LG	The technical contents of the ACTION statement which allow continued power operation with a misaligned rod are incorporated into LCO 3.1.4, ACTION B.1 Bases, of the ITS.
12-19	LS18	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-20	A	The ACTION statement in the CTS to restore the rod drop time to within limits as a Condition for MODE 2 is captured in the frequency for the performance of ITS SR 3.1.4.3. Q3.1-22
12-21		Not used.
12-22	M	This change, in accordance with NUREG-1431, provides a new ACTION in the event the allowed outage times are not met for the rod misalignment action. Prior to this change, LCO 3.0.3 would have been entered allowing for 1 hour prior to placing the plant in HOT STANDBY within the next 6 hours. This change is more restrictive in that the 1 hour time frame is eliminated.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
13-06	A	Not applicable to DCPP. See Conversion Comparison Table. (Enclosure 3B)
13-07	M	The proposed modifications to the SR would require a verification of agreement between digital and demand indicator systems prior to criticality after each removal of the reactor vessel head, instead of every 12 hours. This reflects a reorganization of SRs in the ITS. The requirement for a 12 hour comparison would be moved to SR 3.1.4.1 in the ITS. The post-vessel head removal requirement would be a new specification that demonstrates rod position system OPERABILITY based on a comparison of indicating systems throughout the full range of rod travel. The Frequency requirement of prior to criticality after each removal of the reactor vessel head would permit this comparison to be performed only during plant outages that involve plant evolutions (vessel head removal) that could affect the OPERABILITY of the rod position indication systems. The Frequency change is based on Traveler TSTF-89.
13-08	LS20	Adds provision from Callaway's current specifications which would, under certain conditions, allow continued operation with more than one inoperable DRPI per group. A separate Condition entry allowance is permitted for each inoperable rod position indicator per group and each demand position indicator per bank. A proposed Traveler TSTF-234 ~~Westinghouse Owners Group (WOG) - 73, Rev. 1~~ is in processing to cover this issue. TR 3.1-006
13-09	LS23	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Q3.1-21
14-01	R	~~The Shutdown Position Indication System Specification 3.1.3.3 is relocated outside of the TS. This is consistent with NUREG-1431.~~
15-01	R	~~The Rod Drop Time Specification 3.1.3.4 is relocated outside of the TS. The RCS temperature limit and reactor coolant pumps operating requirement for rod drop testing are combined with CTS Surveillance 4.1.3.4, then incorporated into ITS SR 3.1.4.3. This is consistent with NUREG-1431.~~ → Not used Q3.1-22
15-02	A	~~The Rod Drop Time SR 4.1.3.4.a is moved to the Control Rod ITS LCO 3.1.4 as SR 3.1.4.3. This change is consistent with NUREG-1431.~~ Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Q3.1-22
16-01	LS14	This TS would be revised to apply to shutdown "banks" instead of shutdown "rods;" this is consistent with NUREG-1431. The current ACTION statement permits one rod to be inserted beyond the limits; the proposed ITS Condition A would allow one or more banks to be inserted beyond the limit.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
12-17 A	Editorial changes are made for clarity. Untrippable rods are addressed through ACTION A; hence, there is no additional need to exclude those rods from these Required Actions.	Yes	Yes	No, not in CTS.	No, not in CTS.
12-18 LG	The technical contents of the ACTION statement which allow continued POWER OPERATION with a misaligned rod are moved to the Bases for ITS LCO 3.1.4, ACTION B.1.	Yes	Yes	Yes	Yes
12-19 LS18	The frequency at which the rod motion surveillance is performed is extended from 31 days to 92 days.	No, already in CTS.	Yes	Yes	No, already in CTS.
12-20 A	The ACTION Statement in the CTS to restore the rod drop time to within limits as a Condition for MODE 2 is captured in the frequency for the performance of ITS SR 3.1.4.3.	~~Yes~~ No, See Amendments 120/118.	Yes (D3.1-22)	No, see Amendment 89.	No, see Amendment 103.
12-21	Not used.	N/A	N/A	N/A	N/A
12-22 M	This change, in accordance with NUREG-1431, provides a new ACTION in the event the AOTs are not met for the rod misalignment action. Prior to this change, LCO 3.0.3 would have been entered allowing for 1 hour prior to placing the plant in HOT STANDBY within the next 6 hours. This change is more restrictive in that the 1 hour time frame is eliminated.	Yes	Yes	Yes	Yes
13-01 LG	The specific OPERABILITY attributes of the DRPI system would be moved to the Bases.	Yes	Yes	Yes	Yes
13-02 LS15	The requirement for inoperable DPRI is changed from "with a maximum of one per bank " to " one per group for one or more groups."	Yes	Yes	Yes	Yes
13-03 LS12	A 4 hour Completion Time is specified to verify rod position after movement of a rod with inoperable indicators more than 24 steps in one direction.	Yes	Yes	Yes	Yes

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
15-01 R	~~The Rod Drop Time Specification 3.1.3.4 is relocated outside of the TS. The RCS temperature limit and RCPs operating requirement for rod drop testing are combined with CTS Surveillance 4.1.3.4 then incorporated into ITS SR 3.1.4.3.~~ Not Used	~~Yes, see LAR 95-07 dated 10/4/95, DCL 95-222.~~ NA	~~No, not in CTS - see CN 15-02-A~~ NA	~~No, see Amendment 89.~~ NA	~~No, see Amendment 103.~~ NA Q3.1-22
15-02 A	The Rod Drop Time SR 4.1.3.4.a is moved to the Control Rod ITS LCO 3.1.4 as SR 3.1.4.3.	Yes No, See Amendments 120/118	Yes Q3.1-22	No, already in CTS.	No, already in CTS.
16-01 LS14	The requirement for shutdown insertion limits would be applied to shutdown banks rather than shutdown rods.	Yes	Yes	Yes	Yes
16-02 M	ACTION statements would be changed to specify 2 hours to achieve rod alignment and to prohibit POWER OPERATION with a shutdown bank outside insertion limits.	Yes	Yes	Yes	Yes
16-03 LS22	The requirement to verify shutdown bank insertion within 15 minutes prior to withdrawing any control bank rods during startup would be deleted.	Yes	Yes	Yes	Yes
16-04 M	The Applicability would be modified to include MODE 2 with any control bank not fully inserted.	Yes	Yes	Yes	Yes
16-05 M	This change provides a new ACTION in the event the AOTS are not met for the restoration of the shutdown banks to their insertion limit. Prior to this change, LCO 3.0.3 would have been entered allowing 1 hour prior to placing the plant in HOT STANDBY within the next 6 hours. This change is more restrictive in that the 1 hour time frame is eliminated.	Yes	Yes	Yes	Yes
16-06 A	This change eliminates an unnecessary reference to a separate LCO.	Yes	Yes	Yes	Yes

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-24 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

ITS 3.1.4 Rod Group Alignment Limits (All FLOG Plants)
JFD 3.1-5 & 3.1-6

Comment: Rewording of LCO and Condition A approved, contingent upon OG resubmittal of change request TSTF-107 (revision) as discussed with TSTF.

FLOG RESPONSE: See the response to Comment Number 3.1-15. The FLOG has incorporated TSTF-107, Revision 1.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-25 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

ITS 3.1.4 Rod Group Alignment Limits (All FLOG Plants)
JFD 3.1-16

Comment: Inclusion of SR 3.2.1.2 to Required Action B.2.4 is approved; ensure OG submit WOG-105 as a TSTF change request.

FLOG RESPONSE: At the June 23-24, 1998, meeting of the Westinghouse Owners Group MERITS Mini-Group, traveler WOG-105 was discussed. The remaining action on this traveler was assigned to Westinghouse to expand this change to also apply to ISTS 3.2.1A, "F_Q (Z) (F_{xy} Methodology)." However, this additional work has no impact on the manner in which the FLOG has incorporated this traveler's additional restriction. A TSTF will be submitted to NRC expeditiously.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-27 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

ITS 3.1.1 Shutdown Margin (All FLOG Plants)
JFD 3.1-18

Comment: This modification adds a Mode change restriction from Mode 6 to Mode 5, as discussed in CN 1-02-LS-1 of 3.0. The discussion provided is inadequate to evaluate the necessity of the mode change restriction. In general, throughout the submittal, justifications for notes prohibiting mode changes are inadequate. Provide explanations/justifications that present specific conditions that would necessitate the note.

FLOG RESPONSE: A Reviewer's Note in STS LCO 3.0.4 states: "LCO 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, LCO 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified conditions in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in LCO 3.0.4 were previously applicable in all MODES. Before this version of LCO 3.0.4 can be implemented on a plant-specific basis, the licensee must review the existing technical specifications to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change; such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS." Based on this Reviewer's Note, a matrix of this evaluation was placed in the NSHC LS-1 in Enclosure 4 of the Section 3.0 package (Attachment No. 6).

JFD 3.1-18 has been revised to incorporate additional justification from NSHC LS-1 from Enclosure 4 of the Section 3.0 package (Attachment No. 6). JFD 3.1-18 has been revised to include: "LCO 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, LCO 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified conditions in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in LCO 3.0.4 were previously applicable in all MODES. ITS LCO 3.1.1 was modified by a Note stating: "While this LCO is not met, entry into MODE 5 from MODE 6 is not permitted." Entering MODE 5 without SDM limits being met implies that boron concentration in MODE 6 is not met. Under these conditions, a transition to MODE 5 should not be attempted until MODE 5 SDM limits are met. Inadvertent boron dilution events are precluded in MODE 6 via administrative controls [and instrumentation which would promptly identify and terminate an event], whereas dilution events are not [subject to the same controls and monitoring] in MODE 5. Therefore, the transition from MODE 6 to MODE 5 should not be allowed in the SDM initial condition for a MODE 5 dilution event is not met."

ATTACHED PAGES:

Encl 6A 2

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.1

CHANGE NUMBER	JUSTIFICATION
3.1-8	The frequency for SR 3.1.7.1 for comparing DRPI and group demand position over the full range of travel would be changed from 18 months to "Once prior to criticality after each removal of the reactor vessel head." This change makes it clear that the surveillance must be performed each time the head is removed and that it is not tied to an absolute time interval. This change is based on TSTF-89.
3.1-9	This change would eliminate ISTS 3.1.2 because the SDM requirements for MODE 5 have been incorporated into Specification 3.1.1 in accordance with traveler TSTF-136. Traveler TSTF-9, Rev. 1, relocated values for SDM to the COLR which removed the only difference between ITS LCO 3.1.1 and ITS LCO 3.1.2. Differences above and below 200°F will be addressed in the COLR. Subsequent sections have been renumbered.
3.1-10	Several surveillances (e.g., rod position deviation monitor and rod insertion limit monitor in this section) contain ACTIONS in the form of increased surveillance frequency to be performed in the event of inoperable alarms. These ACTIONS are relocated from the TS to licensee controlled documents since the alarms do not themselves directly relate to the limits. This detail is not required to be in the TS to provide adequate protection of the public health and safety. Therefore, relocation of this detail is acceptable and is consistent with TSTF-110, ~~Rev. 1~~ TR 3.1-004
3.1-11	Not Used.
3.1-12	The Required ACTIONS for inoperable DRPI in ITS 3.1.7 are revised per the CTS to note that the use of movable incore detectors for rod position verification is an indirect assessment at best. The position of some rods can not be ascertained by this method.
3.1-13	This change adds an LCO requirement and SR to MODE 2 PHYSICS TESTS Exceptions, STS 3.1.8 to verify that THERMAL POWER is less than or equal to 5 percent RTP. The LCO requirement and SR were added to verify that THERMAL POWER is within the defined power level for MODE 2 during the performance of PHYSICS TESTS, since there is an ACTION that addresses thermal power not within limit yet there was no corresponding LCO or SR. The surveillance frequency of 1 hour is retained from the CTS. Thses changes are based on TSTF-14, ~~Rev. 3~~ TR 3.1-005
3.1-14	Not used.
3.1-15	Consistent with TSTF-12, ~~Rev. 1~~ TR 3.1-006 Special Test Exception LCOs 3.1.9 and 3.1.11 are deleted. The physics tests contained in LCO 3.1.9 were only contained in some plant initial plant startup testing programs. The physics test can be deleted since these physics tests are never performed during post-refueling outages. The physics test that LCO 3.1.11 required was the rod worth measurement in the N-1 condition. The use of other rod worth measurement techniques will maintain the shutdown margin during the entire measurement process and still provide the necessary physics data verification. Since the N-1 measurement technique is no longer used, the SDM test exception can be deleted. This change and traveler TSTF-136 renumbers ITS 3.1.10 to 3.1.8.
3.1-16	This change adds the requirement to perform SR 3.2.1.2 in addition to SR 3.2.1.1 during performance of ITS 3.1.4, Required Action B 2.4. The intent of Required Action B 2.4 is to verify that $F_Q(z)$ is within ITS limits. $F_Q(z)$ is approximated by $F_Q^C(z)$ (which is obtained via SR 3.2.1.1) and $F_Q^W(z)$ (which is obtained via SR 3.2.1.2). Thus, both $F_Q^C(z)$ and $F_Q^W(z)$ must be established to verify $F_Q(z)$. This change is consistent with Traveler WOG-105.
3.1-17	Consistent with CTS LCO 3.1.3.2, ITS 3.1.7, Condition C is clarified to state that the inoperable position indicators are inoperable DRPIs.
3.1-18	A MODE change restriction has been added to ITS 3.1.1, in LCO Applicability, per the matrix discussed in CN 01-02-LS1 of 3.0 Package (See LS - 1 NSHC in the CTS Section 3/4.0, ITS Section 3.0 package). Insert Q3.1-27

JFD 3.1-18

LCO 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, LCO 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified conditions in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in LCO 3.0.4 were previously applicable in all MODES. ITS LCO 3.1.1 was modified by a Note stating: "While this LCO is not met, entry into MODE 5 from MODE 6 is not permitted." Entering MODE 5 without SDM limits met implies that boron concentration in MODE 6 is not met. Under these conditions, a transition to MODE 5 should not be attempted until MODE 5 SDM limits are met. Inadvertent boron dilution events are precluded in MODE 6 via administrative controls [and instrumentation which would promptly identify and terminate an event], whereas dilution events are not [subject to the same controls and monitoring] in MODE 5. Therefore, the transition from MODE 6 to MODE 5 should not be allowed in the SDM initial condition for a MODE 5 dilution event is not met."

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1-28 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

Relocated Specifications (All FLOG Plants)

Comment: Comanche Peak, Wolf Creek, and Callaway have not provided relocated screening evaluations/forms for any of their specifications relocated to licensee controlled documents. Diablo Canyon has not provided relocated screening forms for all of their specifications relocated to licensee controlled documents. Provide necessary relocation screening evaluations/forms.

FLOG RESPONSE: All relocated specifications have been provided the necessary relocation screening evaluations/forms which are contained in Attachment 21.

This RAI is not applicable to Callaway or Wolf Creek. No relocated specification DOCs were used by those plants in Section 3.1.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC 3.1-ED **APPLICABILITY:** DC

REQUEST:

Various changes that do not impact the technical content of the submittal or other FLOG members.

Changes are noted with DC 3.1-ED in the margin and noted below:

1) During the review performed for General Comment 3.1.G-1, it was identified that some headers were improperly placed and page numbering was missing due to word processing pagination effects in Enclosure 5B. PG&E verified that Attachment 20 to the ITS conversion submittal (clean copy of the ITS Bases) properly shows the placement of the headers and page numbering in ITS 3.1. An example of these pagination issues is attached (B 3.1-5a); however, not every page was included since the clean copy of the ITS Bases is in the appropriate format.
2) Misspelled words.
3) Missing word or use of another word with same meaning.
4) Enclosure 2 typos on LCO 3.1.1.1 which show "greater than" rather then "less than."

ATTACHED PAGES:

Encl. 2	3/4 1-1, 3/4 1-2
Encl. 5B	B 3.1-1a, B 3.1-4, B 3.1-5a, B 3.1-7a, B 3.1-9, B 3.1-10a, B 3.1-11a, B 3.1-12, B 3.1-13, B 3.1-14, B 3.1-15a, B 3.1-16, B 3.1-17a, B 3.1-18, B 3.1-21, B 3.1-22, B 3.1-22a, B 3.1-24, B 3.1-25

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN ~~— T_{avg} GREATER THAN 200°F~~ 02-01-A

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ~~greater than or equal to 1.6% Δk/k~~ within the limits provided in the COLR. 01-01-LG

APPLICABILITY: MODES ~~1~~, 2*, 3, ~~and~~ 4, and 5 02-01-A

ACTION: 01-06-A

With the SHUTDOWN MARGIN ~~less than 1.6% Δk/k, immediately~~ not within limit, within 15 minutes initiate ~~and continue~~ boration ~~at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or equivalent until the required~~ to restore SHUTDOWN MARGIN ~~is restored~~ to within limit. 01-01-LG 01-07-LS16 01-03-LS1

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ~~greater than or equal to 1.6% Δk/k~~ within limits at least once per 24 hour: 01-01-LG

a. ~~Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable.~~ 01-04-LS2
If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); 12-03-A

b. When in MODES 1 or 2 with K_{eff} greater than or equal to 1, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6; 01-08-A

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6; 01-09-A

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e., below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and 05-04-A

~~*See Special Test Exceptions Specification 3.10.1.~~ With K_{eff} ~~≥~~ < 1.0 03-01-A 01-06-A

DC 3.1-ED

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 2*, 3 or 4, at least once per 24 hours ~~by consideration of the following factors:~~ 01-10-M

~~1) Reactor Coolant System boron concentration,~~ 01-05-LG

~~2) Control rod position,~~

~~3) Reactor Coolant System average temperature,~~

~~4) Fuel burnup based on gross thermal energy generation,~~

~~5) Xenon concentration, and~~

~~6) Samarium concentration.~~

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k once prior to entering MODE 1 after each refueling and at least once per 31 Effective Full Power Days (EFPD) after burnup > 60 EFPD. ~~This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above.~~ The predicted reactivity values ~~shall~~ be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. If the reactivity balance is not within limits, within ~~72 hours~~ 7 days evaluate the Safety Analyses and establish appropriate operating restrictions and/or surveillance requirements, or be in at least MODE 3 within the next 6 hours. TR 3.1-003 05-01-M 05-03-LG 05-02-LS7 05-05-LS17 05-06-A ~~03.1-4~~ 03.1-4

may

01-10-M

* With K_{eff} ~~≥~~ 1.0 <

DC 3.1-ED

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are not exceeded. This is done by ensuring that:

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

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ ~~280~~ 200 cal/gm ~~energy desposition~~ average fuel pellet enthalpy at the hot spot in irradiated fuel for the rod ejection accident, Ref. 5); and

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

The most limiting accidents for the SDM requirements ~~is based on a~~ are the main steam line break (MSLB) and inadvertent boron dilution accidents, as described in the ~~accident~~

BASES

SURVEILLANCE REQUIREMENTS (Continued)

verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

In MODES 2 (with k_{eff} < 1.0), 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects (SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth):

a. RCS boron concentration;

b. Control and shutdown rod position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation. (n) DC3.1-ED

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Chapter 15, Section 15.4.2.1.
3. FSAR, Chapter 15, Section 15.2.4.
4. 10 CFR 100.
5. FSAR, Chapter 15, Section 15.4.6.1.6.

DC 3.1-ED

BACKGROUND (continued)

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluate.

DC 3.1-ED

APPLICABLE SAFETY ANALYSIS

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant ~~SAFETY ANALYSES~~ operation is maintained within the assumptions of the safety analyses. delete

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to

Example of Pagination Issues

(Continued)

DC 3.1-ED

DC 3.1-ED

LCO (continued)

When measured core reactivity is within 1% Δk/k of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely. (4) DC 3.1-ED

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel

(Continued)

DC 3.1-ED

SURVEILLANCE REQUIREMENTS (continued)

DC 3.1-ED

60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly. (4) DC 3.1-ED

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
2. FSAR, Chapter 15.

BACKGROUND (continued)

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup. (P) DC3.1-ED

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The FSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

(Continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

MTC satisfies Criterion 2 of ~~the NRC POLICY STATEMENT~~ 10CFR50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration. DC 3.1-ED DC 3.1-ED

LCO

LCO 3.1.43 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive near BOC when core reactivity and required boron concentration are at their maximum values; this upper bound must not be exceeded. This maximum upper limit is evaluated near BOC, all rods out (ARO), hot zero power conditions. At EOC the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOC positive limit and the EOC negative limit are established in the COLR to allow specifying limits for

(Continued)

DC 3.1-ED

LCO (Continued)

each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedul. (e)

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled control rod assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODE. (S)

DC 3.1-ED

ACTIONS

A.1

DC 3.1-ED

If the ~~BOC~~ upper redline MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

~~at BOC~~ DC 3.1-ED

If the required administrative withdrawal limits are not established within 24 hours, the unit must be brought to MODE 2 with $k_{eff} < 1.0$ to prevent operation with an MTC that is more positive than that assumed in safety analyses.

(Continued)

SURVEILLANCE REQUIREMENTS

SR ~~3.1.4.1~~ 3.1.3.1

This SR requires measurement of the MTC at BOC prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the BOC limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOC MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the BOC MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

SR ~~3.1.4.2~~ 3.1.3.2 ~~and SR 3.1.4.3~~

In similar fashion, the LCO demands that the MTC be less negative than the specified value for EOC full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOC LCO limit. The 300 ppm SR value is sufficiently less negative than the EOC LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

SR 3.1.~~4.3~~3.2 is modified by ~~a~~ three Notes that include the following requirements:

1. The SR is required to be performed once each cycle within 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm;

~~a~~2. If the 300 ppm Surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned EOC. Because the MTC changes slowly with core depletion, the Frequency of 14 EFPD is sufficient to avoid exceeding the EOC limit.

redline

~~EFFECTIVE FULL POWER DAYS~~

DC 3.1-ED

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B ~~3.1.5~~ 3.1.4 Rod Group Alignment Limits

DC 3.1-ED

BASES

~~e.g.~~ redline

Q3.1-15

BACKGROUND

The OPERABILITY (~~i.e.~~, trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately ⅝ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

(Continued)

Two types of analysis are performed in regard to static rod misalignment (Ref. 3). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from ~~a~~ bank D inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 4).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod

(Continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10CFR50.36(c)(2)(ii). DC 3.1-E2 ~~the NRC Policy Statement~~

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability ~~to meet SDM~~) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required time assumed in the safety analyses. Rod control malfunctions that result in the inability to move a rod (e.g., rod ~~urgent~~ lift coil failures), but do not impact trippability, do not necessarily result in rod inoperability. 03.1-15

The requirement to maintain the rod alignment to within plus or minus 12 steps of their group step counter demand position is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

The requirement to maintain rod alignment is met by comparing individual rod DRPI indication and bank demand position indication to be within plus or minus 12 steps. If one of these position indicators become inoperable, the conditions of this LCO are still met by compliance with LCO 3.1.7.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

(Continued)

BASES

~~moved~~

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be fully inserted and control bank C must be inserted to approximately 100 steps ~~to 115 steps~~. redline CP3.1-15

Power operation may continue with one RCCA ~~OPERABLE (i.e. trippable)~~ ~~but~~ misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration. DC3.1-ED

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, ~~RTP~~ reactor power must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible. Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 3). The Completion

(Continued)

BASES

ACTION reduction (continued)

reduction

DC3.1-ED

Time of 2 hours gives the operator sufficient time to accomplish an orderly power without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Q3.1-13

The Accident Analyses of FSAR Chapter 15 are to be used to identify the appropriate design bases events requiring re-evaluation.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

~~The following accident analyses require re-evaluation for continued operation with a misaligned rod:~~

~~Rod Cluster Control Assembly Insertion Characteristics~~

~~Rod Cluster Control Assembly Misalignment~~

~~Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuate the Emergency Core Cooling System~~

~~Single Rod Cluster Control Assembly Withdrawal at Full Power~~

~~Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)~~

~~Major Secondary Coolant System Pipe Rupture~~

~~Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection).~~

Q3.1-13

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B ~~3.1.6~~ 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among four control banks and four shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. All ~~plants have four control banks and at least two shutdown banks~~four control banks contain two rod groups. Two shutdown banks (A and B) contain two rod groups and the remaining two shutdown banks (C and D) consist of a single group. See LCO ~~3.1.6~~ 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO ~~3.1.8~~ 3.1.7, "Rod Position Indication," for position indication requirements.

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B ~~3.1.6~~ 3.1.5 Shutdown Bank Insertion Limits

BASES

APPLICABLE SAFETY ANALYSIS (Continued)

maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bankinsertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion safety limits and inoperability or misalignment is that:

a. There be no violations of:

1. Specified acceptable fuel design limits, or
2. RCS pressure boundary integrity; and

b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3).

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10CFR50.36(c)(2)(ii). ~~(the NRC Policy Statement)~~ DC 3.1-ED

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B ~~3.1.6~~ 3.1.5 Shutdown Bank Insertion Limits

DC 3.1-ED

BASES

APPLICABILITY

The shutdown banks must be within their insertion limits with the reactor in MODES 1 and 2. The applicability in MODE 2 begins at initial control bank withdrawal, during an approach to criticality, and continues throughout MODE 2, until all control bank rods are again fully inserted by reactor trip or by shutdown. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks are typically fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 ~~and LCO 3.1.2~~ for SDM requirements in MODES 2 with k_{eff} < 1.0, 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.~~5.24.2~~. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1.1, A.1.2 and A.2

When one or more shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B ~~3.1.7~~ 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among four control banks and four shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. ~~A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. All plants have four control banks and at least two shutdown banks.~~ A group consists of two or more RCCAs that are electrically paralleled to step simultaneously and are moved in a staggered fashion, but always within one step of each other. Two shutdown banks (C and D) consist of a single group. See LCO ~~3.1.6~~ 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO ~~3.1.8~~ 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. ~~An example is provided for information only in Figure B 3.1.7-1.~~ The control banks are required to be at or above the insertion limit lines.

(Continued)

BASES

BACKGROUND

event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident (continued)requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

a. There be no violations of:

1. Specified acceptable fuel design limits, or

2. Reactor Coolant System pressure boundary integrity; and

b. The core remains subcritical after accident transients.

As such, the shutdown and control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. 3).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 4).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 5).

The insertion limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii), in that they are initial conditions assumed in the safety analysis. DC3.1-E0

(Continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC 3.1-001 **APPLICABILITY:** DC, CP

REQUEST:

Revise DOC 12-11-TR3 in enclosure 3A to reflect lead plant wording.

ATTACHED PAGES:

Encl 3A 7

CHANGE NUMBER	NSHC	DESCRIPTION
12-08	LS9	Consistent with NUREG-1431, the requirement to reduce the high neutron flux set point to ≤ 85 percent of RTP would be deleted. This is acceptable because the underlying safety limits are not of a nature that requires immediate shutdown of the plant if they are exceeded. This is evidenced by the allowance of 72 hours to verify peaking factors. It is assumed that during this 72-hour period an event will not occur which will raise the power level and cause a high neutron flux trip at 100 percent RTP. If a power excursion would occur from the 75 percent RTP ACTION statement limit, the initial peaking factors would not be critical to the analysis, since the analysis is based on the peaking factors at 100 percent RTP. Therefore, the risk of a reactor trip caused by adjusting the power range trip set points is not justified by the potential consequences of failing to reduce the trip set points.
12-09	M	Not applicable to DCPP. See Conversion Comparison Table. (Enclosure 3B)
12-10	LS10	The requirement to maintain RCS T_{avg} ≥541°F during rod drop testing would be revised to maintain T_{avg} ≥500°F. NUREG-1431, allows the tests to be performed at temperatures as low as 500°F. Because the RCS coolant is more dense at lower temperatures, the rod drop time would be greater at the lower temperatures than at the higher temperatures. In addition, the RCS is borated such that the SDM remains within its limits at the Conditions existing during these tests. Nevertheless, this change, which allows more flexibility of plant conditions for conducting rod drop testing, is a relaxation in plant operations in the CTS.
12-11	TR3	It is proposed to move the requirement to perform rod drop testing on individual rods following maintenance that could affect the drop time ~~to~~ ~~licensee controlled documents~~ [and to delete the 18 month requirement]. The requirement to perform drop time testing following each removal of the reactor vessel head would not be modified. The proposed change is justified, because in addition to being consistent with NUREG-1431, Rev. 1, good maintenance practices would require a retest following any maintenance on a rod or its drive system that could affect drop time. Furthermore, it is difficult to postulate any maintenance on a rod that could affect its drop time without requiring the reactor vessel head to be removed in the process. The components of the rod and rod drive system that affect drop time (as defined in TS) are all inside the reactor coolant pressure boundary. Therefore, moving this requirement outside the TS would have essentially no impact on rod OPERABILITY. [Measuring rod drop time following each removal of the reactor vessel head is considered equivalent if not more restrictive than an 18 month frequency requirement; therefore, deleting the 18 month requirement where it exists (not all plants have it in the CTS) is an administrative change.]

[24] DC-ALL-001

DC 3.1-001

However, those post-maintenance test requirements are not appropriate for inclusion in the Technical Specifications

[24] DC-ALL-001

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC ALL-001 **APPLICABILITY:** DC

REQUEST:

LAs 119/117 and 118/116 were issued 7/13/97 and addressed CTS surveillance interval increases due to 24-month fuel cycles. These changes on pages affected by NRC comment numbers are indicated with "DC-ALL-001." These changes were previously submitted to the NRC in an errata to LAR 97-09 via DCL-98-003 (dated January 8, 1998).

ATTACHED PAGES:

See notations on applicable pages for each comment number.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC ALL-002 **APPLICABILITY:** DC

REQUEST:

An errata to LAR 97-09 was submitted to the NRC January 8, 1998, in DCL-98-003. Errata changes on pages affected by NRC comment numbers are indicated with "DC-ALL-002."

ATTACHED PAGES:

See notations on applicable pages for each comment number.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: TR 3.1-001 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

Incorporate NRC-approved traveler TSTF-108, Revision 1, to delete the words "within 12 hours" from the Frequency of CTS SR 4.10.3.2 and ITS SR 3.1.8.1.

ATTACHED PAGES:

Section 3.1

Encl. 5A	Traveler Status Sheet, 3.1-18
Encl. 5B	B 3.1-37
Encl. 6A	3
Encl. 6B	3

Section 3.10

Encl. 2	3/4 10-3
Encl. 3A	2
Encl. 3B	1
Encl. 4	14, 15

Industry Travelers Applicable to Section 3.1

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-9, Rev. 1	Incorporated	3.1-1	NRC approved.
TSTF-12, Rev. 1	Incorporated	3.1-15	NRC approved. ITS Special Test Exception 3.1.10 is retained and re-numbered as 3.8.1, consistent with this traveler and TSTF-136.
TSTF-13, Rev. 1	Incorporated	3.1-4	NRC approved.
TSTF-14, Rev. ~~0~~ 4	Incorporated	3.1-13	NRC approved. TR 3.1-005
TSTF-15, Rev. 1	Incorporated	N/A	NRC approved.
TSTF-89	Incorporated	3.1-8	NRC approved.
TSTF-107, Rev. 1	Incorporated	3.1-6	Q3.1-15
TSTF-108, Rev. 1	~~Not~~ Incorporated	~~N/A~~ 3.1-21	~~Not~~ NRC approved ~~as of cut-off date~~ TR 3.1-001
TSTF-110, Rev. ~~0~~ 2	Incorporated	3.1-10	NRC Approved TR 3.1-004
TSTF-136	Incorporated	3.1-9, 3.1-15	
TSTF-141	Not Incorporated	N/A	Disagree with change; traveler issued after cut-off date
TSTF-142	~~Not~~ Incorporated	~~N/A~~ 3.1-22	TR 3.1-003 ~~Traveler issued after cut-off date.~~ NRC Approved.
~~WOG-73, Rev. 1~~ TSTF 234	Incorporated	3.1-7	TR 3.1-006
WOG-105	Incorporated	3.1-16	

3.1-9
3.1-15

SURVEILLANCE REQUIREMENTS.

SURVEILLANCE		FREQUENCY
SR 3.1.~~10.1~~ 8.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1.	~~Within 12 hours~~ Prior to initiation of PHYSICS TESTS
SR 3.1.~~10.2~~ 8.2	Verify the RCS lowest operating loop average temperature is ≥531°F.	30 minutes
SR 3.1.~~10.3~~ 8.3	Verify THERMAL POWER is ≤ 5% RTP.	1 hour
SR 3.1.~~10~~.8.4	Verify SDM is ~~≥ 1.6% Δk/k~~ within the limits provided in the COLR.	24 hours

B
3.1-21
TR3.1-001
3.1-20
B
3.1-13
3.1-1

BASES

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.~~10~~8.1

TR 3.1-001

The required power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each OPERABLE power range and intermediate range channels ~~within 12 hours~~ prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. ~~The 12 hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.~~

SR 3.1.~~10~~8.2

Verification that the RCS lowest operating loop T_{avg} is ≥ 531°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.3

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of ~~[30 minutes]~~ 1 hour during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.~~10~~8.4

Verification that the SDM is within limits specified in the COLR ensures that, for the specific RCCA and RCS temperature manipulations performed during PHYSICS TESTS, the plant is not operating in a condition that could invalidate the safety analysis assumptions.

(Continued)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.1

CHANGE NUMBER	JUSTIFICATION
3.1-19	Not used.
3.1-20	Consistent with CTS 3/4.10.3, "Physics Tests," ITS LCO 3.1.8 and its Condition C and SR 3.1.8.2 are modified to refer to "operating" RCS loops. If an operating loop were secured while operating in MODE 2, Required Action A.1 of ITS 3.4.4 directs the plant to be in MODE 3 within 6 hours. However, depending on initial power level and plant Conditions, securing an RCS loop may result in a loop's average temperature of [531]°F fairly quickly due to the reverse flow through the secured loop and the cooling achieved by the steam generator. In this scenario, it would not be appropriate to invoke Required Action C.1 of ITS 3.1.8 with its 15 minute Completion Time to restore loop average temperature to within limits since the flow in the secured loop is not passing through the core and ITS LCO 3.4.4 provides appropriate corrective ACTION, consistent with CTS.
3.1-21	Insert TR 3.1-001
3.1-22	Insert TR 3.1-003

JFD 3.1-21 The ISTS SR 3.1.8.1 requirement to perform a CHANNEL OPERATIONAL TEST (COT) on the intermediate and power range NIS channels within 12 hours prior to initiating PHYSICS TESTS is revised to delete phrase "within 12 hours." COT testing is performed on these channels prior to reactor startup per LCO 3.3.1. This change is consistent with traveler TSTF-108.

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.1-16	This change adds the requirement to perform SR 3.2.1.2 in addition to SR 3.2.1.1 during performance of ITS 3.1.4 Required Action B 2.4, consistent with traveler WOG-105.	Yes	Yes	Yes	Yes
3.1-17	Consistent with CTS LCO 3.1.3.2 and the wording of ITS 3.1.7, Condition A and B, ITS 3.1.7, Condition C is clarified to state that the inoperable position indicators are inoperable position indicators are inoperable DRIPs.	Yes	Yes	Yes	Yes
3.1-18	A MODE change restriction has been added to ITS 3.1.1, in LCO Applicability, per the matrix discussed in CN 01-02-LS1 of the 3.0 Package.	Yes	Yes	Yes	Yes
3.1-19	Not used.	N/A	N/A	N/A	N/A
3.1-20	Consistent with CTS 3/4.10.3, "Physics Tests," ITS LCO 3.1.8 and its Condition C and SR 3.1.8.2 are modified to refer to "operating" RCS loops.	Yes	Yes	Yes	Yes
3.1-21	Insert	Yes	Yes	Yes	Yes (TR 3.1-001)
3.1-22	Insert	Yes	Yes	Yes	Yes (TR 3.1-003)

JFD 3.1-21 The ISTS SR 3.1.8.1 requirement to perform a CHANNEL OPERATIONAL TEST (COT) on the intermediate and power range NIS channels within 12 hours prior to initiating PHYSICS TESTS is revised to delete phrase "within 12 hours." COT testing is performed on these channels prior to reactor startup per LCO 3.3.1.

Applicability:

DC	Yes
CP	Yes
WC	Yes
CA	Yes

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. ~~The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and~~ SHUTDOWN MARGIN is within the limits provided in the COLR; and 03-02-A 03-01-M

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 531°F.

APPLICABILITY: ~~MODE 2~~ during PHYSICS TESTS. 03-04-A

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 531°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

"New" With SHUTDOWN MARGIN not within its limits, within 15 minutes, initiate boration to restore SHUTDOWN MARGIN to within its limits and, within one hour, suspend PHYSICS TESTS exception. 03-01-M

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to a CHANNEL OPERATIONAL TEST ~~within 12 hours~~ prior to initiating PHYSICS TESTS. 03-05-LSI TR 3.1-001

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 531°F at least once per 30 minutes during PHYSICS TESTS.

(NEW) Verify SHUTDOWN MARGIN to be within limits provided on the COLR once per 24 hours. 03-01-M

DESCRIPTION OF CHANGES TO TS SECTION 3/4.10
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
03-05	LS1	Insert TR 3.1-001
04-01	M	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
05-01	R	STE LCO 3.10.4,"Position Indication System - Shutdown," would be relocated based on relocation of LCO 3.1.3.3, "Position Indicating Systems - Shutdown." NRC application of TS Criteria concluded that the STE LCOs could be included with corresponding LCOs remaining in TS and that LCO 3.10.4 could be relocated with LCO 3.1.3.3.
5-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).

ITS Section 3.10:

Encl. 3A Page 2

3-05 LS1 The current SR requiring the performance of [an ANALOG] CHANNEL OPERATIONAL TEST on each intermediate and power range NIS channel within 12 hours prior to initiating PHYSICS TESTS is revised to delete the phrase "within 12 hours." [A]COT testing is performed on these channels prior to reactor startup per LCO 3.3.1. Current TS LCO 3.3.1, Reactor Trip System (RTS) Instrumentation, requires the performance of an [A]COT on the power range low setpoint and intermediate range NIS channels prior to each reactor startup, if not performed within the previous 31 days (revised to 92 days in the conversion to ITS 3.3). These RTS SRs must be performed prior to entering the LCO 3.3.1 Applicabilities for these RTS trip functions since there are no CTS SR 4.0.4 exceptions. Current SR 4.10.3.2 requires an arbitrary estimate of when the plant is within 12 hours of intiating PHYSICS TESTING. This has no basis from the accident analyses, which are satisfied as long as the surveillances are current prior to entering plant MODES where these trip functions provide protection. When these surveillances are current, they have previously been determined to remain valid for 92 days. The initiation of PHYSICS TESTING does not impact the ability of the channels to perform their required function, does not affect the trip setpoints or trip capability of these channels and does not invalidate the previous surveillances. This change is consistent with traveler TSTF-108.

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 M	LCO 3.10.1 would be deleted consistent with Traveler TSTF-12. The SDM STE requires rod worth measurement in the N-1 condition. Other rod worth measurements techniques will maintain the SDM during the measurement process and provide the necessary physics data. The N-1 measurement technique is no longer used.	Yes, see Attachment 21, page 29	Yes	No, Amendment 89 previously deleted this STE.	No, Amendment 103 previously deleted this STE.
02-01 M	The MODE 1 STE LCO 3.10.2 would be deleted in accordance with Traveler TSTF-12. TSTF-12 would delete ITS LCO 3.1.9, which contains the STEs for MODE 1 because the tests implemented by ITS LCO 3.1.9 were contained in some initial plant startup test programs and are not performed during post-refueling startup testing. Post-refueling tests may be performed in accordance with other LCOs.	Yes	Yes	Yes	Yes
03-01 M	Consistent with NUREG-1431, Rev. 1, the SDM parameter is added to the list of preconditions required prior to invoking this STE. The actual value of the SDM is located in the COLR. Associated ACTION statements and surveillances are also added.	Yes	Yes	Yes	Yes
03-02 A	The requirement to ensure that the reactor trip setpoints are operable already exists in the reactor trip system LCO, Table 3.3.1-1, functional items 3 and 4 of the ITS.	Yes	Yes	Yes	Yes
03-03	Not Used.	NA	NA	NA	NA
03-05 LSI	Insert	Yes	Yes	Yes	Yes

TR 3.1-001

3-05-LS1 The current SR requiring the performance of [an ANALOG] CHANNEL OPERATIONAL TEST on each intermediate and power range NIS channel within 12 hours prior to initiating PHYSICS TESTS is revised to delete the phrase "within 12 hours.

Applicability:

DC	Yes
CP	Yes
WC	Yes
CA	Yes

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS-1
10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The current SR requiring the performance of [an ANALOG] CHANNEL OPERATIONAL TEST on each intermediate and power range NIS channel within 12 hours prior to initiating PHYSICS TESTS is revised to delete the phrase "within 12 hours." This change is consistent with traveler TSTF-108.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> 1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
>
> 2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
>
> 3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. Current TS LCO 3.3.1, Reactor Trip System (RTS) Instrumentation, requires the performance of an [A]COT on the power range low setpoint and intermediate range NIS channels prior to each reactor startup, if not performed within the previous 31 days (revised to 92 days in the conversion to ITS 3.3). These RTS SRs must be performed prior to entering the LCO 3.3.1 Applicabilities for these RTS trip functions since there are no CTS SR 4.0.4 exceptions. Current SR 4.10.3.2 requires an arbitrary estimate of when the plant is within 12 hours of initiating PHYSICS TESTING. This has no basis from the accident analyses, which are satisfied as long as the surveillances are current prior to entering plant MODES where these trip functions provide protection. When these surveillances are current, they have previously been determined to

remain valid for 92 days. The initiation of PHYSICS TESTING does not impact the ability of the channels to perform their required function, does not affect the trip setpoints or trip capability of these channels and does not invalidate the previous surveillances. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change to the SR will not affect the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3 Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria, analysis assumptions, methodologies, or credited equipment for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS-1" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: TR 3.1-003 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

Incorporate NRC-approved traveler TSTF-142 to increase the [CTS 3.1.1.5 AOT and] ITS 3.1.2 Required Actions A.1 and A.2 Completion Time from 72 hours to 7 days when the core reactivity balance is not within its limit.

ATTACHED PAGES:

Encl. 2	3/4 1-2, 3/4 1-7
Encl. 3A	5
Encl. 3B	4
Encl. 4	Table of Contents, 42, 43
Encl. 5A	Traveler Status Sheet, 3.1-2
Encl. 6A	3
Encl. 6B	3

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 2*, 3 or 4, at least once per 24 hours ~~by consideration of the following factors:~~ 01-10-M

~~1) Reactor Coolant System boron concentration,~~

01-05-LG

~~2) Control rod position,~~

~~3) Reactor Coolant System average temperature,~~

~~4) Fuel burnup based on gross thermal energy generation,~~

~~5) Xenon concentration, and~~

~~6) Samarium concentration.~~

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k once prior to entering MODE 1 after each refueling and at least once per 31 Effective Full Power Days (EFPD) after burnup > 60 EFPD. ~~This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above.~~ The predicted reactivity values ~~shall~~ may be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. If the reactivity balance is not within limits, within ~~72 hours~~ 7 days evaluate the Safety Analyses and establish appropriate operating restrictions and/or surveillance requirements, or be in at least MODE 3 within the next 6 hours. TR 3.1-003

05-01-M 05-03-LG 05-02-LS7 05-05-LS17 05-06-A @3.1-4 @3.1-4

01-10-M

* With K_{eff} ~~≥1.0~~ <1

DC 3.1-ED

LIMITING CONDITION FOR OPERATION

3.1.1.5 The measured core reactivity shall be within ± 1% Δk/k of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity not within limit.	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	~~72 hours~~ 7 days TR3.1-003 05-05-LS17
	AND	
	A.2 Establish appropriate operating restrictions and SRs.	~~72 hours~~ 7 days TR3.1-003
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

CHANGE NUMBER	NSHC	DESCRIPTION
05-04	A	CTS SR requires a SDM verification prior to operation above 5 percent power after each refueling with the control rod banks at maximum insertion limits. SDM in MODES 1 and 2 is determined by shutdown and control rods maintained at their insertion limits. The relevant requirements regarding the adequacy of the SDM with rods at their insertion limits is determined through compliance with ITS 3.1.2, which requires a reactivity balance prior to entering MODE 1.after each refueling; and ITS SR 3.1.6.1, which requires a verification of control bank position within insertion limits within 4 hours prior to criticality. Therefore, the requirements of this SR would be performed by other specifications in the ITS. [ ]
05-05	LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown. This is consistent with NUREG-1431, as modified by TSTF-142 TR 3.1-003
05-06	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Insert Q3.1-4
05-07	LS24	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). TR 3.1-003
06-01	R	The CTS 3.1.2.1, "Boration Flow Path Shutdown," and associated SR 4.1.2.1 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-6
07-01	R	The CTS 3.1.2.2, "Boration Flow Path Operating," and associated SR 4.1.2.2 are relocated outside of the TS. This is consistent with NUREG-1431. Q3.1-7 Not Applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-01	R	The CTS 3.1.2.3, "Charging Pump Shutdown," and associated SRs 4.1.2.3.1 and 4.1.2.3.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-8
08-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS19	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-04	A	Insert Q3.1-9
09-01	R	The CTS 3.1.2.4, "Charging Pump Operating," and associated SRs 4.1.2.4.1 and 4.1.2.4.2 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-10
10-01	R	The CTS 3.1.2.5, "Borated Water Source Shutdown," and associated CTS SR 4.1.2.5 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-11
11-01	R	The CTS 3.1.2.6, "Borated Water Source Operating," and associated CTS SR 4.1.2.6 are relocated outside of the TS. This is consistent with NUREG-1431. Insert Q3.1-12

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-05 LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown.	Yes	Yes	No, already in CTS.	No, already in CTS.
05-06 A	CTS SR 4.1.1.5.1 requires that the predicted reactivity values "shall" be adjusted (normalized) at 60 EFPD after refueling. ITS SR 3.1.2.1 states the normalization requirement as "may" be adjusted: This is to recognize that normalization is not necessary if predicted and measured core reactivity are within acceptance tolerance. The scheduling of predicted and measured core reactivity continues to be required at 60 EFPD. Therefore, this change reflects clarification of existing intent and is considered administrative.	~~No, maintaining CTS wording.~~ Yes	~~No, maintaining CTS wording.~~ Yes	Yes Q3.1-4	Yes
06-01 R	Relocates "Boration Flow Path - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 1. - relocated to Cn ECG	Yes	No, see Amendment 89. Q3.1-6	No, see Amendment 103.
07-01 R	Relocates "Boration Flow Path - Operating" TS to licensee controlled document.	~~Yes, see LAR 95 07 dated 10/4/95, DCL 95-222.~~ No, see Amendments 120/118	Yes	No, see Amendment 89. Q3.1-7	No, see Amendment 103.
07-02 A	Moves limitation on charging pumps in MODE 4 to ITS SR 3.4.12.2.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-01 R	Relocates "Charging Pumps - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 3. - relocated to Cn ECG	Yes	No, see Amendment 89. Q3.1-8	No, see Amendment 103.
08-02 M	Moves charging pump SR when below 350°F to ITS SR 3.4.12.2 and decreases surveillance frequency to 12 hours from 31 days.	No, already in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-03 LS19	Deletes the method of verificating that charging pumps are incapable of injecting into the RCS.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
05-07 LS24	Insert	No, see CN 05-05-LS17	No, See CN 05-05-LS17	Yes	Yes TR 3.1-003
08-04 A	Insert	Yes	No, see CN 8-02-M	No, see Amendment 89	No, see Amendment 103 Q3.1-9

DOC 05-07-LS24:

The Allowed Outage Time (AOT) in the ACTION Statement of current TS LCO 3.1.1.5 is increased from 72 hours to 7 days.

Applicability:

DC	No - See CN 05-05 LS-17.
CP	No - See CN 05-05 LS-17.
WC	Yes
CA	Yes

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS17
10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS.

The proposed activity would add required ACTIONS if the overall core reactivity balance was not within ±1 percent $\Delta k/k$ of the predicted values. In the CTS, there are no Required ACTIONS; thus LCO 3.0.3 would be entered. LCO 3.0.3 requires, that within 1 hour, ACTIONS be initiated to place the plant in a condition in which the LCO did not apply. Because this particular SR is only required in MODES 1 and 2, LCO 3.0.3 would further require that the plant be placed in HOT STANDBY (MODE 3) within the following 6 hours. The proposed change, consistent with NUREG-1431, would allow ~~72 hours~~ 7 days to evaluate the safety analyses and establish appropriate operating restrictions and/or SRs. If these activities were not completed within the ~~72 hour~~ 7 day period, then the plant would be placed in MODE 3 within the following 6 hours as modified by TSTF-142. TR 3.1-003

The requirement to periodically compare the measured and predicted overall core reactivity balances provides assurance that the analytical predictions upon which the safety analyses are based accurately represent the actual core response. Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of ~~72 hours~~ 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical Condition of the reactor and complete any required evaluations of the core design and safety analyses. TR 3.1-003

Following evaluations of the measurement, the core design, and the safety analysis, the cause of the reactivity anomaly may be resolved. If it is concluded that the reactor core is acceptable for continued operation, then the predicted core reactivity balance may be renormalized and Power Operation may continue. If operation restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The Required Completion time of ~~72 hours~~ 7 days is adequate for preparing whatever operating restrictions or surveillance that may be required to allow continued reactor operation. TR 3.1-003

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:*
>
> 1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
>
> 2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
>
> 3. *Involve a significant reduction in a margin of safety."*

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change does not involve any new operating activities or hardware changes; thus the proposed changes has no effect on the probability of an accident. Although a small effect, the proposed change may slightly reduce the probability of an accident by allowing additional time to resolve discrepancies, and thus avoid an unnecessary plant transient (shutdown).

   Satisfaction of the SR acceptance criterion provides assurance that the core-related reactivity parameters used in the safety analysis adequately represent the actual core conditions. During the 72 hour action time following an initial failure of the SR acceptance criterion, ACTIONS are established which would ensure continued agreement between the safety analysis and the actual core conditions; thereby, maintaining the validity of the safety analyses. Therefore, there is no effect on the consequences of an accident previously evaluated.

   Because the available time is increased from 1 hour to 7 days ~~to 72 hours~~ (TR 3.1-003), the probability of an accident occurring during the time period when the plant condition is under review is slightly increased; however, the increase is small and has been previously found to be acceptable by the NRC staff through the approval of NUREG-1431.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Operation for a period of time with a discrepancy between the measured and predicted core reactivity balances is allowed by the CTS; therefore, there is no possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

   The accident analyses are assumed to be initiated from conditions which are consistent with the TS LCO. Thus, for the purposes of the accident analyses, it assumed that the agreement between the predicted and measured core reactivity balance is within an acceptable range. These assumptions remain valid since there is no design, operation, maintenance, or testing revision associated with this change. Therefore, there is no significant reduction in margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS17" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

Industry Travelers Applicable to Section 3.1

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-9, Rev. 1	Incorporated	3.1-1	NRC approved.
TSTF-12, Rev. 1	Incorporated	3.1-15	NRC approved. ITS Special Test Exception 3.1.10 is retained and re-numbered as 3.8.1, consistent with this traveler and TSTF-136.
TSTF-13, Rev. 1	Incorporated	3.1-4	NRC approved.
TSTF-14, Rev. ~~0~~ 4	Incorporated	3.1-13	NRC approved. TR 3.1-005
TSTF-15, Rev. 1	Incorporated	N/A	NRC approved.
TSTF-89	Incorporated	3.1-8	NRC approved.
TSTF-107, Rev. 1	Incorporated	3.1-6	Q3.1-15
TSTF-108, Rev. 1	~~Not~~ Incorporated	~~N/A~~ 3.1-21	~~Not~~ NRC approved ~~as of cut-off date~~ TR 3.1-001
TSTF-110, Rev. ~~0~~ 2	Incorporated	3.1-10	NRC Approved TR 3.1-004
TSTF-136	Incorporated	3.1-9, 3.1-15	
TSTF-141	Not Incorporated	N/A	Disagree with change; traveler issued after cut-off date
TSTF-142	~~Not~~ Incorporated	~~N/A~~ 3.1-22	TR 3.1-003 ~~Traveler issued after cut-off date.~~ NRC Approved.
~~WOG-73, Rev. 1~~ TSTF 234	Incorporated	3.1-7	TR 3.1-006
WOG-105	Incorporated	3.1-16	

3.1 REACTIVITY CONTROL SYSTEMS

~~3.1.3~~ 3.1.2 Core Reactivity

LCO ~~3.1.3~~ 3.1.2 The measured core reactivity shall be within ± 1% Δk/k of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity not within limit.	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	~~72 hours~~ 7 days
	AND	
	A.2 Establish appropriate operating restrictions and SRs.	~~72 hours~~ 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

TR 3.1-003

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.1

CHANGE NUMBER	JUSTIFICATION
3.1-19	Not used.
3.1-20	Consistent with CTS 3/4.10.3, "Physics Tests," ITS LCO 3.1.8 and its Condition C and SR 3.1.8.2 are modified to refer to "operating" RCS loops. If an operating loop were secured while operating in MODE 2, Required Action A.1 of ITS 3.4.4 directs the plant to be in MODE 3 within 6 hours. However, depending on initial power level and plant Conditions, securing an RCS loop may result in a loop's average temperature of [531]°F fairly quickly due to the reverse flow through the secured loop and the cooling achieved by the steam generator. In this scenario, it would not be appropriate to invoke Required Action C.1 of ITS 3.1.8 with its 15 minute Completion Time to restore loop average temperature to within limits since the flow in the secured loop is not passing through the core and ITS LCO 3.4.4 provides appropriate corrective ACTION, consistent with CTS.
3.1-21	Insert TR 3.1-001
3.1-22	Insert TR 3.1-003

JFD 3.1-22 The Completion Times for ITS 3.1.2, Required Actions A.1 and A.2 are increased from 72 hours to 7 days, consistent with traveler TSTF-142.

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.1-16	This change adds the requirement to perform SR 3.2.1.2 in addition to SR 3.2.1.1 during performance of ITS 3.1.4 Required Action B 2.4, consistent with traveler WOG-105.	Yes	Yes	Yes	Yes
3.1-17	Consistent with CTS LCO 3.1.3.2 and the wording of ITS 3.1.7, Condition A and B, ITS 3.1.7, Condition C is clarified to state that the inoperable position indicators are inoperable position indicators are inoperable DRIPs.	Yes	Yes	Yes	Yes
3.1-18	A MODE change restriction has been added to ITS 3.1.1, in LCO Applicability, per the matrix discussed in CN 01-02-LS1 of the 3.0 Package.	Yes	Yes	Yes	Yes
3.1-19	Not used.	N/A	N/A	N/A	N/A
3.1-20	Consistent with CTS 3/4.10.3, "Physics Tests," ITS LCO 3.1.8 and its Condition C and SR 3.1.8.2 are modified to refer to "operating" RCS loops.	Yes	Yes	Yes	Yes
3.1-21	Insert	Yes	Yes	Yes	Yes (TR 3.1-001)
3.1-22	Insert	Yes	Yes	Yes	Yes (TR 3.1-003)

JFD 3.1-22 The Completion Times for ITS 3.1.2, Required Actions A.1 and A.2 are increased from 72 hours to 7 days.

Applicability:

DC	Yes
CP	Yes
WC	Yes
CA	Yes

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: TR 3.1-004 Thru TR 3.1-006 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

Revise Traveler Status Sheet to reflect NRC approval and latest revision number of travelers TSTF-14, Revision 4, TSTF-110, Revision 2, and TSTF-136. Change WOG-73, Revision. 1 to TSTF-234 (still under NRC review). Remove traveler revision numbers everywhere except on the Traveler Status Sheet. There no changes involved to any CTS mark-ups, ITS mark-ups, DOCs, or JFDs.

ATTACHED PAGES:

Encl. 3A	4, 8, 10
Encl. 3B	1, 3, 6, 8
Encl. 4	24, 25
Encl. 5A	Traveler Status Sheet
Encl. 6A	1, 2
Encl. 6B	1, 2

CHANGE NUMBER	NSHC	DESCRIPTION
04-02	LS6	The proposed change would revise the conditional SR for verifying that reactor coolant system (RCS) temperature (T_{avg}) is within limits by changing the frequency to once per 12 hours in accordance with TSTF-27. ~~Rev. 2.~~ The original frequency requirements were within 15 minutes prior to achieving reactor criticality and at least once per 30 minutes when the reactor is critical and the (T_{avg}-T_{ref}) deviation alarm is not reset. The RCS temperature is maintained within limit: (1) to assure that the MTC is within the limits assured in the accident analysis, (2) to assure that the neutron detectors are not adversely affected by attenuation caused by low RCS temperature, (3) to assure that the RCS and pressurizer response to thermal hydraulic transients is as predicted, and (4) to assure that the reactor vessel temperature is above the nil-ductility transition reference temperature. The plant design incorporates monitoring of T_{avg} and provides an alarm, the (T_{avg}-T_{ref}) deviation alarm, as T_{avg} approaches its limit. This alarm Condition requires a response by the operating staff. Therefore, at any time that T_{avg} is approaching its limiting value, the plant operators would receive an alarm and initiate corrective ACTION.
04-03	A	The LCO for Minimum Temperature For Criticality, CTS 3/4.1.1.4, is moved to ITS 3.4.2 in the RCS section. This is consistent with NUREG-1431.
05-01	M	The CTS SR that requires a comparison of measured reactivity with predicted would be modified to add the requirement for performance prior to entry into MODE 1 after refueling outages. This is a new requirement from the ITS that is not in CTS. The CTS Bases indicate that the comparison should be made at RATED THERMAL POWER (RTP) after startup from a refueling outage.
05-02	LS7	This proposed change would specify that the overall core reactivity balance comparison shall be done every 31 effective full power days (EFPD) after burnup exceeds 60 EFPD. This clarifies when the 31 EFPD surveillance should start. The current SR specifies a frequency of 31 EFPD. The delay in initiating the monthly surveillance is acceptable because of the slow rate of changes in the core due to fuel depletion and the presence of other indicators for prompt determination of an anomaly. As noted in the Bases for Specification 3.1.2, the reactivity balance comparison must be done within the first 60 EFPDs after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.
05-03	LG	The list of specific items to be considered in the performance of a reactivity balance verification is moved to the Bases for ITS 3.1.2. This change is of the type that moves unnecessary details from the specifications while leaving the overall safety requirement intact. See also CN 01-05-LG.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
12-12	LS13	CTS [3.1.3.1] ACTIONS are revised to delete reference to causes of control rod inoperability due to rod control urgent failure or other electrical problems in the rod control system.
12-13		Not used.
12-14	M	This wording is broadened from "untrippable" to "inoperable" to ensure all causes of inoperability are covered. The previous wording covered inoperable rods if they were untrippable (e.g., "immovable as a result of excessive friction or mechanical interference...") but did not cover trippable rods with drop times that exceed the surveillance limit. These slow rods are inoperable. This more restrictive change clarifies the appropriate ACTIONS to be taken for all causes of inoperability, consistent with Traveler TSTF-107.
12-15	A	Not applicable to DCPP. See Comparison Table (Enclosure 3B).
12-16	LG	Several surveillances (e.g., rod position deviation monitor and rod insertion limit monitor in this section) contain ACTIONS in the form of increased surveillance frequency to be performed in the event of inoperable alarms. These ACTIONS are moved from the CTS to license controlled documents since the alarm does not, in itself, directly relate to the limits. This detail is not required to be in the TS. Therefore, moving this detail is acceptable and is consistent with Traveler TSTF-110. Rev. 1. T23.1-004
12-17	A	Editorial changes are made for clarity. Untrippable rods are addressed through ACTION A; hence, there is no additional need to exclude those rods from these required ACTIONS.
12-18	LG	The technical contents of the ACTION statement which allow continued power operation with a misaligned rod are incorporated into LCO 3.1.4, ACTION B.1 Bases, of the ITS.
12-19	LS18	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-20	A	The ACTION statement in the CTS to restore the rod drop time to within limits as a Condition for MODE 2 is captured in the frequency for the performance of ITS SR 3.1.4.3. Q3.1-22
12-21		Not used.
12-22	M	This change, in accordance with NUREG-1431, provides a new ACTION in the event the allowed outage times are not met for the rod misalignment action. Prior to this change, LCO 3.0.3 would have been entered allowing for 1 hour prior to placing the plant in HOT STANDBY within the next 6 hours This change is more restrictive in that the 1 hour time frame is eliminated.

CHANGE NUMBER	NSHC	DESCRIPTION
13-06	A	Not applicable to DCPP. See Conversion Comparison Table. (Enclosure 3B)
13-07	M	The proposed modifications to the SR would require a verification of agreement between digital and demand indicator systems prior to criticality after each removal of the reactor vessel head, instead of every 12 hours. This reflects a reorganization of SRs in the ITS. The requirement for a 12 hour comparison would be moved to SR 3.1.4.1 in the ITS. The post-vessel head removal requirement would be a new specification that demonstrates rod position system OPERABILITY based on a comparison of indicating systems throughout the full range of rod travel. The Frequency requirement of prior to criticality after each removal of the reactor vessel head would permit this comparison to be performed only during plant outages that involve plant evolutions (vessel head removal) that could affect the OPERABILITY of the rod position indication systems. The Frequency change is based on Traveler TSTF-89.
13-08	LS20	Adds provision from Callaway's current specifications which would, under certain conditions, allow continued operation with more than one inoperable DRPI per group. A separate Condition entry allowance is permitted for each inoperable rod position indicator per group and each demand position indicator per bank. A proposed Traveler TSTF-234 ~~Westinghouse Owners Group (WOG) 73, Rev. 1~~ is in processing to cover this issue. TR 3.1-006
13-09	LS23	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Q3.1-21
14-01	R	~~The Shutdown Position Indication System Specification 3.1.3.3 is relocated outside of the TS. This is consistent with NUREG-1431.~~
15-01	R	~~The Rod Drop Time Specification 3.1.3.4 is relocated outside of the TS. The RCS temperature limit and reactor coolant pumps operating requirement for rod drop testing are combined with CTS Surveillance 4.1.3.4, then incorporated into ITS SR 3.1.4.3. This is consistent with NUREG-1431.~~ → Not used Q3.1-22
15-02	A	~~The Rod Drop Time SR 4.1.3.4.a is moved to the Control Rod ITS LCO 3.1.4 as SR 3.1.4.3. This change is consistent with NUREG-1431.~~ Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). Q3.1-22
16-01	LS14	This TS would be revised to apply to shutdown "banks" instead of shutdown "rods;" this is consistent with NUREG-1431. The current ACTION statement permits one rod to be inserted beyond the limits; the proposed ITS Condition A would allow one or more banks to be inserted beyond the limit.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 LG	In accordance with TSTF-9 Rev 1 this change would move the specified limit for SDM from TS to the COLR.	Yes TR 3.1-006	No, already in CTS.	Yes	Yes
01-02 M	MODE 2 with Keff < 1.0 would be added to Applicability of the SDM specification.	No, different CTS	No, different CTS.	Yes	Yes
01-03 LS1	The ACTION statement would be modified to reflect that the requirement to initiate boration at a specified rate with fluid at a specified boron concentration is generalized to simply require boration.	Yes	Yes	Yes	Yes
01-04 LS2	The requirement of SR to verify SDM within 1 hour of detecting an inoperable rod and once per 12 hours thereafter would be deleted from the SDM LCO.	Yes	Yes	Yes	Yes
01-05 LG	The list of specific items to be considered in the performance of a SDM verification is moved to the ITS 3.1.1 Bases.	Yes	Yes	Yes	Yes
01-06 A	This change revises the SDM LCO Applicability to MODE 2 with K_{eff}<1.0, MODE 3, and MODE 4. This change also creates a new core reactivity LCO based on ITS 3.1.2.	Yes	Yes	No, see CN 1-02-M and CTS.	No, see CN 1-02-M and CTS.
01-07 LS16	Changes the time required for the initiation of boration from "immediately" to "within 15 minutes."	Yes	Yes	No, already in CTS.	No, already in CTS.
01-08 A	The technical contents of this SR (verification of SDM through compliance with rod insertion limits) in MODE 1 and MODE 2 with K_{eff}≥1.0 have been incorporated into LCO 3.1.6 of the ITS.	Yes	Yes	No, see Amendment 89.	No, see Amendment 103.
01-09 A	Moves requirement to verify estimated critical rod position prior to criticality to ITS SR 3.1.6.1.	Yes	Yes	No, see Amendment 89.	No, see Amendment 103.
01-10 M	CTS SR 4.1.1.1.1.e is modified by this change to be applicable to MODE 2 with k_{eff}<1.0 as well as the current MODES 3 and 4. This is consistent with NUREG 1431. See also CN 01-06-A.	Yes	Yes	No, see Amendment 89.	No, see Amendment 103.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
04-01 LS5	This change would alter the ACTION statement shutdown requirement time limit from a combination of 15 minutes to restore T_{avg} to within limits followed by 15 minutes to be in MODE 3, if T_{avg} could not be restored, to a single 30 minute limit to exit the Applicability if T_{avg} were not within its limit. In addition, the ACTION statement would be changed to require achieving MODE 2 with $k_{eff} < 1.0$ instead of achieving HOT STANDBY if the LCO is not met (refer to TSTF-26).	Yes	Yes	Yes	Yes
04-02 LS6	The SR interval to measure RCS loop average temperature is revised to 12 hours in accordance with industry Traveler TSTF-27. ~~Rev 2.~~	Yes TR 31-006	Yes	Yes	Yes
04-03 A	The LCO for Minimum Temperature for Criticality, CTS 3.1.1.4, is moved to ITS 3.4.2 in the RCS Section.	Yes	Yes	Yes	Yes
05-01 M	The SR that requires a comparison of measured reactivity to predicted would be modified to add the requirement to compare core reactivity against the predicted prior to entry into MODE 1 after refueling outages.	Yes	Yes	Yes	Yes
05-02 LS7	This proposed change would specify that the overall core reactivity balance comparison shall be done every 31 EFPD after burnup exceeds 60 EFPD.	Yes	Yes	Yes	Yes
05-03 LG	The list of specific items to be considered in the performance of a reactivity balance verification is moved to the ITS 3.1.2 Bases.	Yes	Yes	Yes	Yes
05-04 A	The SR requiring SDM to be verified prior to initial operation in MODE 1 after each refueling is effectively performed under other specifications. This SR required a SDM verification prior to operation above 5 percent power after each refueling with the control rod banks at maximum insertion limits and is being deleted since it is redundant.	Yes	Yes	Yes	Yes

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
12-09 M	This proposed change would reinstate an SDM verification requirement that had been eliminated in a previous license amendment.	No, already in CTS.	No, already in CTS.	Yes	Yes
12-10 LS10	The requirement to maintain RCS $T_{avg} \geq [541]°F$ during rod drop testing would be revised to maintain $T_{avg} \geq 500°F$.	Yes	Yes	Yes	Yes
12-11 TR3	The requirement to perform drop testing on rods following maintenance would be removed from the CTS.	Yes, also deletes redundant ~~18~~ 24 month interval.	Yes, also deletes redundant 18 month interval.	Yes DC-ALL-001	Yes
12-12 LS13	CTS [3.1.3.1] ACTIONS are revised to delete reference causes of control rod inoperability due to rod control urgent failure or other electrical problems in the rod control system.	Yes	Yes	Yes	No, CTS already revised to incorporate.
12-13	Not used.	N/A	N/A	N/A	N/A
12-14 M	This wording is broadened from "untrippable" to "inoperable" to ensure all causes of inoperability are covered. This more restrictive change clarifies the appropriate ACTIONS to be taken for all causes of inoperability, consistent with Traveler TSTF-107.	Yes	Yes	Yes	Yes
12-15 A	Rod misalignment is determined based on a comparison between the rod's DRPI and its group step counter demand position, not on a rod to rod position verification. This change is administrative in nature in that there is no effect on the manner in which the operating staff would determine whether a misalignment event had occurred.	No, already in CTS.	No, already in CTS.	Yes	Yes
12-16 LG	Several surveillances (e.g., rod position deviation monitor and rod insertion limit monitor in this section) contain ACTIONS in the form of increased surveillance frequency to be performed in the event of inoperable alarms. These ACTIONS are relocated from the TS to licensee controlled documents. This is consistent with TSTF-110, ~~Rev. 1.~~	Yes, moved to the FSAR. TR3.1-004	Yes, moved to TRM.	Yes, moved to the USAR.	Yes, moved to the FSAR, Section 16.1.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
13-04 M	A requirement would be added to bring the plant to MODE 3 within 6 hours if the required ACTIONS and Completion Times are not met.	Yes	Yes	Yes	Yes
13-05 A	The proposed change would retain an ACTION statement, currently in the plant TS, that permits continued POWER OPERATION with more than 1 digital rod position indicator per group inoperable.	No, not in CTS - see 13-08-LS20.	No, not in CTS - see 13-08-LS20.	Yes	Yes
13-06 A	The change would allow separate Condition entry for each inoperable DRPI per group or each demand indicator per bank.	No, not in CTS - see 13-08-LS20.	No, not in CTS - see 13-08-LS20.	Yes	Yes
13-07 M	The proposed modifications to the SR would verify agreement between digital and demand indicator systems prior to criticality after the reactor vessel head was removed instead of every 12 hours. The Frequency change is based on Traveler TSTF-89.	Yes	Yes	Yes	Yes
13-08 LS20	Adds provision in Callaway's current specifications which would, under certain Conditions, allow continued operation with more than one inoperable DRPI per group. This is consistent with Traveler ~~WOG-73, Rev. 1~~. TSTF-234	Yes	Yes TR3.1-006	No, already in CTS.	No, already in CTS.
13-09 LS23	CTS ACTIONS b.1.b) and b.1.c) of LCO 3.1.3.2 are deleted. SDM is ensured in MODES 1 and 2 by rod position. Multiple inoperable DRPIs will have no impact on SDM in MODES 1 and 2 if the control rod positions are verified by alternate means and rod motion is limited consistent with the accident analyses. Deletion of these requirements is consistent with traveler ~~WOG-73, Rev. 1~~. TSTF-234	No, not in CTS.	No, not in CTS. TR3.1-006	Yes	Yes
14-01 R	Relocates CTS 3.1.3.3 to licensee controlled documents, consistent with NUREG-1431.	~~Yes, see LAR 95-07 dated 10/4/95, DCL 95-222.~~ No, see Amendments 120/118.	Yes, relocated to TRM.	No, see Amendment 89.	No, see Amendment 103.

Q3.1-21

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6
10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

TR 3.1-006

The proposed change would revise the SR for verifying that RCS temperature (T_{avg}) is within limits by changing the frequency to once per 12 hours in accordance with TSTF-27, Rev. 2. The current frequency requirements were within 15 minutes prior to achieving reactor criticality, which is redundant and unnecessary since T_{avg} must be within its limit prior to entering the LCO Applicability, and at least once per 30 minutes when the reactor is critical and the (T_{avg}-T_{ref}) deviation alarm is not reset. The RCS temperature is maintained within limit: (1) to assure that the MTC is within the limits assumed in the accident analyses, (2) to assure that the neutron detectors are not adversely affected by neutron attenuation caused by low coolant temperature, (3) to assure that the RCS and pressurizer response to thermal-hydraulic transients is as predicted, and (4) to assure that the reactor vessel temperature is above the nil-ductility transition reference temperature.

The plant design incorporates monitoring of T_{avg} and provides an alarm, the (T_{avg}-T_{ref}) deviation alarm, as T_{avg} approaches its limit. This alarm Condition requires a response by the operating staff. Therefore, at any time that T_{avg} is approaching its limiting value, the plant operators would receive an alarm and initiate corrective ACTION.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:*
>
> 1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
> 2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
> 3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The CTS requires that the RCS average temperature is required to be verified within 15 minutes prior to reactor criticality. No additional verification is required during reactor criticality unless the (T_{avg}-T_{ref}) deviation alarm actuated. Therefore, the CTS relies on the (T_{avg}-T_{ref}) deviation alarm to assure T_{avg} is within limits during reactor operation. The proposed change would introduce a 12 hour requirement to verify T_{avg} independent of the (T_{avg}-T_{ref}) deviation alarm. Therefore, the proposed change would provide additional assurance beyond the CTS SR requirement that T_{avg} was within the limits assumed in accident and transient analyses.

   With regard to the requirement to verify temperature within 15 minutes of achieving reactor criticality, the (T_{avg}-T_{ref}) deviation alarm will still provide warning that RCS temperature is not within limit. In addition, during an approach to criticality, the plant is operated such that rapid or significant temperature changes in the RCS are avoided. Since the specification has no CTS 4.0.4 exception, T_{avg} must still be within limit prior to entering the LCO Applicability, i.e. prior to criticality.

   Based on the foregoing discussion, the proposed change would provide additional assurance that RCS T_{avg} would remain within limits assumed in accident analyses during reactor operation. Therefore, the proposed change would not involve a significant increase in the probability or consequences of a previously evaluated accident.

NSHC LS6
(Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change deals with the frequency of monitoring a parameter (RCS temperature) that is an initial Condition of accident and transient analyses. Changes in SR frequency would not lead to changes in plant system operations or other conditions that could cause an accident of a new or different type. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

As noted in the evaluation of Criterion 1 above, the main method of monitoring RCS temperature, during normal plant operation and during approach to criticality, is via the (T_{avg}-T_{ref}) deviation alarm. This situation would not be altered by the proposed changes. However, the proposed change would add an additional requirement to verify RCS temperature every 12 hours. The minimum T_{avg} for criticality is not changed nor are there any changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_Q, $F^N_{\Delta H}$, LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS6" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

Industry Travelers Applicable to Section 3.1

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-9, Rev. 1	Incorporated	3.1-1	NRC approved.
TSTF-12, Rev. 1	Incorporated	3.1-15	NRC approved. ITS Special Test Exception 3.1.10 is retained and re-numbered as 3.8.1, consistent with this traveler and TSTF-136.
TSTF-13, Rev. 1	Incorporated	3.1-4	NRC approved.
TSTF-14, Rev. ~~3~~ 4	Incorporated	3.1-13	NRC approved. TR 3.1-005
TSTF-15, Rev. 1	Incorporated	N/A	NRC approved.
TSTF-89	Incorporated	3.1-8	NRC approved.
TSTF-107, Rev. 1	Incorporated	3.1-6	Q3.1-15
TSTF-108, Rev. 1	~~Not~~ Incorporated	~~N/A~~ 3.1-21	~~No~~ NRC approved ~~as of cut off date~~ TR 3.1-001
TSTF-110, Rev. ~~1~~ 2	Incorporated	3.1-10	NRC Approved TR 3.1-004
TSTF-136	Incorporated	3.1-9, 3.1-15	
TSTF-141	Not Incorporated	N/A	Disagree with change; traveler issued after cut-off date
TSTF-142	~~Not~~ Incorporated	~~N/A~~ 3.1-22	TR 3.1-003 ~~Traveler issued after cut off date.~~ NRC Approved.
~~WOG-73, Rev. 1~~ TSTF 234	Incorporated	3.1-7	TR 3.1-006
WOG-105	Incorporated	3.1-16	

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.1

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[ ]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER	JUSTIFICATION
3.1-1	In accordance with TSTF-9, Rev. 1, this change would relocate the specified limit for SDM from ITS to the COLR. This change occurs in several specifications including the specification for SDM and those specifications with ACTIONS that require verifying SDM within limits.
3.1-2	Q3.1-4 The Note for SR 3.1.2.1 indicates that predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 EFPD after each refueling. However, both the Bases for Specification 3.1.3 and the CTS requirements in Specification 3.1.1.5 state that the normalization shall be done prior to exceeding a fuel burnup of 60 EFPD after each refueling. Not used.
3.1-3	~~Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).~~ Not Used
3.1-4	Q3.1-23 SR 3.1.4.2 of NUREG-1431, Rev. 1 would be deleted. In accordance with TSTF-13, Rev. 1, the intent of this SR is only to determine the next frequency for SR 3.1.4.3. TR 3.1-006 Performance of SR 3.1.4.2 is not necessary to assure that the LCO is met; SR 3.1.4.3 fulfills that purpose. Therefore, SR 3.1.4.2 may be deleted. In addition, the note in the frequency column of SR 3.1.4.2 would be moved to become Note 1 in the surveillance column of SR 3.1.4.3. This is for clarification purposes. As discussed in CN 3.1-9, section renumbering results in SR 3.1.4.3 of NUREG-1431, Rev. 1 becoming ITS SR 3.1.3.2.
3.1-5	Per CTS [3.1.3.1], the words "with all" have been removed from ITS LCO 3.1.4. This is a clarification that ensures the proper interpretation of the LCO. The change makes it clear that only one channel of DRPI is necessary to meet the alignment accuracy requirement of the LCO. With the word "all" in the statement, it may be possible for those unfamiliar with the DRPI design to interpret the LCO as applying to all channels of DRPI.
3.1-6	LCO 3.1.4 would be split into two separate statements to clarify that the alignment limit is separate from OPERABILITY of the control rod. The Condition A wording is broadened from "untrippable" to "inoperable" to ensure the Condition encompasses all causes of inoperability. Previous wording was ambiguous for rods that, for instance, had slow drop times but were still trippable. These slow rods are inoperable rods, and the change clarifies the appropriate ACTIONS. The Bases are changed to reflect the changes to the LCO and Condition A. These changes are based on TSTF-107.
3.1-7	This change to the ISTS would incorporate, into LCO 3.1.7, an ACTION statement that was previously approved as part of the Callaway and Wolf Creek licensing basis as revised in Enclosure 2. The ACTION statement would permit continued POWER OPERATION for up to 24 hours with more than one DRPI channel per rod group inoperable. The ACTION statement specifies additional Required ACTIONS beyond those applicable to the Condition of 1 DRPI channel per group inoperable. The Bases for this change also would be incorporated into the Bases for the plant ITS. These changes are consistent with Traveler ~~WOG-73, Rev. 1~~. The note under the ACTIONS is changed to be consistent with the new Required Actions. TSTF-234 TR 3.1-006

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.1

CHANGE NUMBER	JUSTIFICATION
3.1-8	The frequency for SR 3.1.7.1 for comparing DRPI and group demand position over the full range of travel would be changed from 18 months to "Once prior to criticality after each removal of the reactor vessel head." This change makes it clear that the surveillance must be performed each time the head is removed and that it is not tied to an absolute time interval. This change is based on TSTF-89.
3.1-9	This change would eliminate ISTS 3.1.2 because the SDM requirements for MODE 5 have been incorporated into Specification 3.1.1 in accordance with traveler TSTF-136. Traveler TSTF-9, Rev. 1, relocated values for SDM to the COLR which removed the only difference between ITS LCO 3.1.1 and ITS LCO 3.1.2. Differences above and below 200°F will be addressed in the COLR. Subsequent sections have been renumbered.
3.1-10	Several surveillances (e.g., rod position deviation monitor and rod insertion limit monitor in this section) contain ACTIONS in the form of increased surveillance frequency to be performed in the event of inoperable alarms. These ACTIONS are relocated from the TS to licensee controlled documents since the alarms do not themselves directly relate to the limits. This detail is not required to be in the TS to provide adequate protection of the public health and safety. Therefore, relocation of this detail is acceptable and is consistent with TSTF-110, ~~Rev. 1~~ TR 3.1-004
3.1-11	Not Used.
3.1-12	The Required ACTIONS for inoperable DRPI in ITS 3.1.7 are revised per the CTS to note that the use of movable incore detectors for rod position verification is an indirect assessment at best. The position of some rods can not be ascertained by this method.
3.1-13	This change adds an LCO requirement and SR to MODE 2 PHYSICS TESTS Exceptions, STS 3.1.8 to verify that THERMAL POWER is less than or equal to 5 percent RTP. The LCO requirement and SR were added to verify that THERMAL POWER is within the defined power level for MODE 2 during the performance of PHYSICS TESTS, since there is an ACTION that addresses thermal power not within limit yet there was no corresponding LCO or SR. The surveillance frequency of 1 hour is retained from the CTS. Thses changes are based on TSTF-14, ~~Rev. 3~~ TR 3.1-005
3.1-14	Not used.
3.1-15	Consistent with TSTF-12, ~~Rev. 1~~ TR 3.1-006 Special Test Exception LCOs 3.1.9 and 3.1.11 are deleted. The physics tests contained in LCO 3.1.9 were only contained in some plant initial plant startup testing programs. The physics test can be deleted since these physics tests are never performed during post-refueling outages. The physics test that LCO 3.1.11 required was the rod worth measurement in the N-1 condition. The use of other rod worth measurement techniques will maintain the shutdown margin during the entire measurement process and still provide the necessary physics data verification. Since the N-1 measurement technique is no longer used, the SDM test exception can be deleted. This change and traveler TSTF-136 renumbers ITS 3.1.10 to 3.1.8.
3.1-16	This change adds the requirement to perform SR 3.2.1.2 in addition to SR 3.2.1.1 during performance of ITS 3.1.4, Required Action B 2.4. The intent of Required Action B 2.4 is to verify that $F_Q(z)$ is within ITS limits. $F_Q(z)$ is approximated by $F_Q^C(z)$ (which is obtained via SR 3.2.1.1) and $F_Q^W(z)$ (which is obtained via SR 3.2.1.2). Thus, both $F_Q^C(z)$ and $F_Q^W(z)$ must be established to verify $F_Q(z)$. This change is consistent with Traveler WOG-105.
3.1-17	Consistent with CTS LCO 3.1.3.2, ITS 3.1.7, Condition C is clarified to state that the inoperable position indicators are inoperable DRPIs.
3.1-18	A MODE change restriction has been added to ITS 3.1.1, in LCO Applicability, per the matrix discussed in CN 01-02-LS1 of 3.0 Package (See LS - 1 NSHC in the CTS Section 3/4.0, ITS Section 3.0 package). Insert T Q3.1-27

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.1-1	In accordance with industry Traveler TSTF-9, Rev. 1,this change would relocate the specified limits for SDM from several TS to the COLR.	Yes	Yes	Yes	Yes
3.1-2	Changes the note to SR 3.1.2.1, which deals with verifying core reactivity within limits, to state that the normalization of predicted reactivity values to correspond to measured values shall be done prior to exceeding a fuel burnup of 60 EFPD after each refueling.	~~Yes~~ NA Not Used	~~Yes~~ NA	~~No, maintaining ITS wording.~~ NA	~~No, maintaining ITS wording.~~ NA Q3.1-4
3.1-3	The Wolf Creek ITS LCO 3.1.6 Required Action C.1 is revised from "Be in MODE 3." to "Be in MODE 2 with k_{eff} < 1.0." Not Used	~~No~~ NA	~~No~~ NA	~~Yes~~ NA	~~No~~ NA Q3.1-23
3.1-4	In accordance with industry Traveler TSTF-13 Rev. 1, SR 3.1.4.2, which requires verifying MTC within the 300 ppm boron limit, is deleted and the note in that SR is moved to the SR that requires the lower MTC limit to be verified. The deleted SR is not a requirement separate from the lower MTC verification SR, but is essentially a clarification of when the SR for the lower MTC limit should be performed.	Yes TR3.1-006	Yes	Yes	Yes
3.1-5	Per CTS [3.1.3.1], the words "with all" are removed from the LCO for control rod alignment limits. This ensures that the number of channels of DRPI required to be OPERABLE will not be misconstrued.	Yes	Yes	Yes	Yes
3.1-6	In accordance with TSTF - 107, the change provides additional clarification that the alignment limits in the LCO are separate from the OPERABILITY of a control rod.	Yes	Yes	Yes	Yes
3.1-7	An ACTION statement that was previously approved as part of the current licensing basis of Callaway and Wolf Creek would be added to ITS 3.1.7, as revised in Enclosure 2. The ACTION statement would permit operation for up to 24 hours with more than one digital rod position indicator per group inoperable.	Yes	Yes	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.1-8	In accordance with TSTF - 89, the requirement to compare DRPI against group demand position would be required whenever the reactor vessel head is removed, not every 18 months.	Yes	Yes	Yes	Yes
3.1-9	This change would eliminate ITS 3.1.2 because the SDM requirements for MODE 5 have been incorporated into Specification 3.1.1 in accordance with Traveler TSTF-136.	Yes	Yes	Yes	Yes
3.1-10	Several surveillances (e.g., rod position deviation monitor and rod insertion limit monitor in this section) contain ACTIONS in the form of increased surveillance frequency to be performed in the event of inoperable alarms. These ACTIONS are relocated from the TS to licensee controlled documents. This is consistent with TSTF-110, Rev 1 e	Yes TR 3.1-004	Yes	Yes	Yes
3.1-11	Not Used.	N/A	N/A	N/A	N/A
3.1-12	The required ACTIONS for inoperable DRPI are revised per the current licensing basis to note that the use of movable incore detectors for rod position verification is an indirect assessment at best. The position of some rods can not be ascertained by this method.	Yes	Yes	Yes	Yes
3.1-13	In accordance with traveler TSTF-14, Rev. 3 the LCO and SR are modified to verify that THERMAL POWER ≤ 5% RTP. This provides an LCO requirement to correspond to Condition B which requires RTP to be within limit.	Yes TR 3.1-005	Yes	Yes	Yes
3.1-14	Not used.	N/A	N/A	N/A	N/A
3.1-15	In accordance with TSTF-12, ~~Rev. 1,~~ this change would delete its LCOs 3.1.9 and 3.1-11. This change and TSTF-136 renumbers ITS 3.1:10 to ITS 3.1.8.	Yes TR 3.1-006	Yes	Yes	Yes

JLS CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS

CTS 3/4.2 - POWER DISTRIBUTION SYSTEMS
ITS 3.2 - POWER DISTRIBUTION SYSTEMS

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION AND LICENSEE INITIATED ADDITIONAL CHANGES

INDEX OF ADDITIONAL INFORMATION

ADDITIONAL INFORMATION NUMBER	APPLICABILITY	ENCLOSED
3.2.G-1	DC, CP, WC, CA	YES
3.2-1	CP	NA
3.2-2	DC	YES
3.2-3	DC, CP, WC, CA	YES
3.2-4	DC, CP, WC, CA	YES
3.2-5	CA	NA
3.2-6	DC, CP, WC, CA	YES
3.2-7	WC, CA	NA
3.2-8	WC	NA
3.2-9	WC	NA
3.2-10	DC, CP, WC, CA	YES
CA 3.2-001	DC, WC, CA	YES
CA 3.2-002	DC, WC, CA	YES
CP 3.2-ED	CP	NA
CP 3.2-001	DC, CP, WC, CA	YES
DC 3.2-001	DC	YES
DC 3.2-ED	DC	YES
DC ALL-005 (3.2 changes only)	DC	YES
TR 3.2-004	DC, CP, WC, CA	YES
WC 3.2-001	WC, CP	NA

JOINT LICENSING SUBCOMMITTEE METHODOLOGY FOR PROVIDING ADDITIONAL INFORMATION

The following methodology is followed for submitting additional information:

1. Each licensee is submitting a separate response for each section.

2. If an RAI does not apply to a licensee (i.e., does not actually impact the information that defines the technical specification change for that licensee), "NA" has been entered in the index column labeled "ENCLOSED" and no information is provided in the response for that licensee.

3. If a licensee initiated change does not apply, "NA" has been entered in the index column labeled "ENCLOSED" and no information is provided in the response for that licensee.

4. The common portions of the "Additional Information Cover Sheets" are identical, except for brackets, where applicable (using the same methodology used in enclosures 3A, 3B, 4, 6A and 6B of the conversion submittals). The list of attached pages will vary to match the licensee specific conversion submittals. A licensee's FLOG response may not address all applicable plants if there is insufficient similarity in the plant specific responses to justify their inclusion in each submittal. In those cases, the response will be prefaced with a heading such as "PLANT SPECIFIC DISCUSSION."

5. Changes are indicated using the redline/strikeout tool of WordPerfect or by using a hand markup that indicates insertions and deletions. If the area being revised is not clear, the affected portion of the page is circled. The markup techniques vary as necessary, based on the specifics of the area being changed and the complexity of the changes, to provide the clearest possible indication of the changes.

6. A marginal note (the Additional Information Number from the index) is added in the right margin of each page being changed, adjacent to the area being changed, to identify the source of each change.

7. Some changes are not applicable to one licensee but still require changes to the Tables provided in Enclosures 3A, 3B, 4, 6A, and 6B of the original license amendment request to reflect the changes being made by one or more of the other licensees. These changes are not included in the additional information for the licensee to which the change does not apply, as the changes are only for consistency, do not technically affect the request for that licensee, and are being provided in the additional information being provided by the licensees for which the change is applicable. The complete set of changes for the license amendment request will be provided in a licensing amendment request supplement to be provided later.

8. The item numbers are formatted as follows:

   [Source] [ITS Section]-[nnn]

   Source = Q - NRC Question
   CA - AmerenUE
   DC - PG&E
   WC - WCNOC
   CP - TU Electric
   TR - Traveler

   ITS Section = The ITS section associated with the item (e.g., 3.3). If all sections are potentially impacted by a broad change or set of changes, "ALL" is used for the section number.

   nnn = a three digit sequential number

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.2.G-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST:

ITS 3.2.x Bases

General

There have been a number of instances that the specific changes to the STS Bases are not properly identified with redline or strikeout marks.

Comment: Perform an audit of all STS Bases markups and identify instances where additions and/or deletions of Bases were not properly identified in the original submittal.

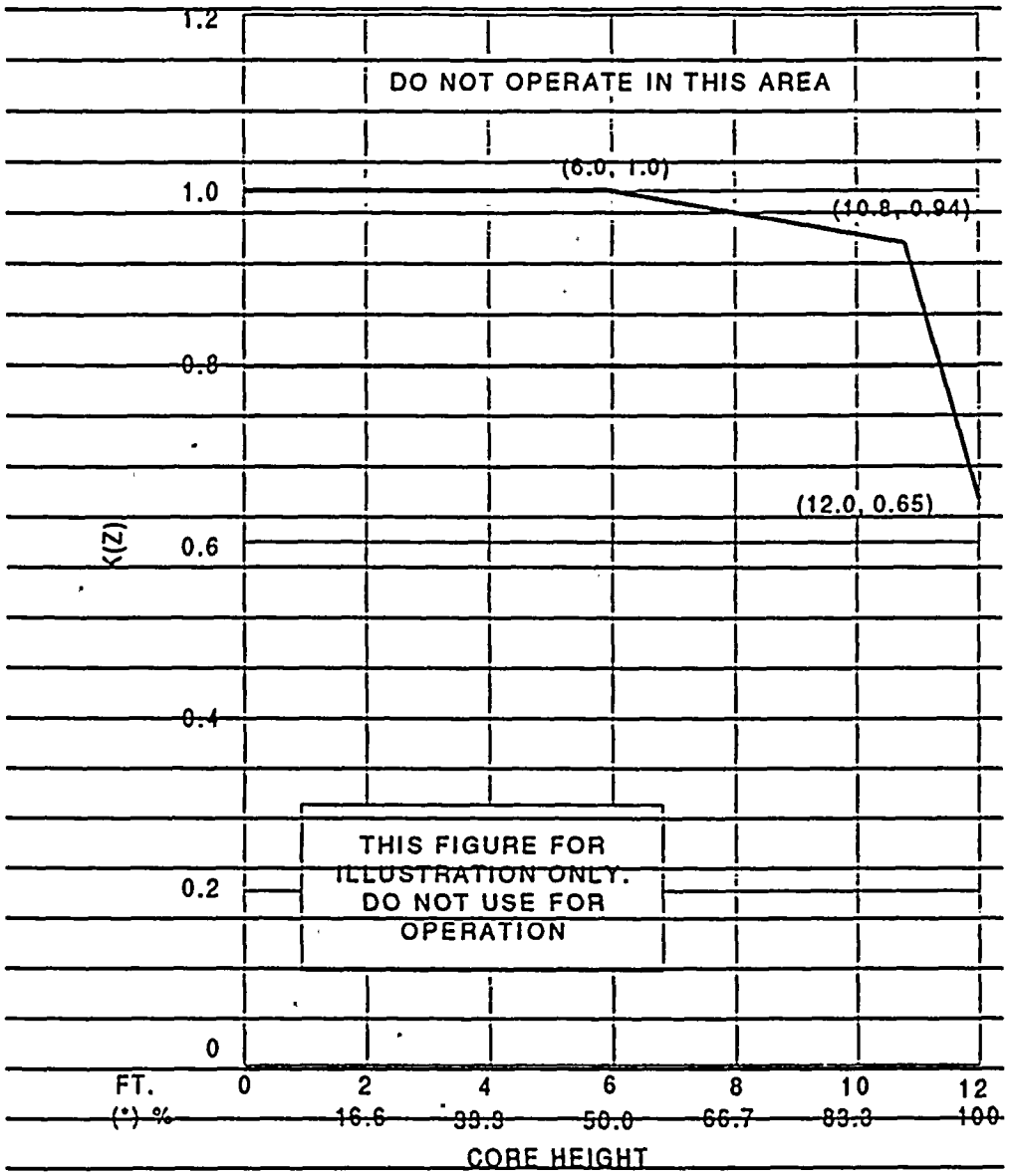
FLOG response: The submitted ITS Bases markups for Section 3.2 have been compared to the STS Bases. Some differences that were identified were in accordance with the markup methodologies (e.g., deletion of brackets and reviewer's notes). Most of the differences were editorial in nature and would not have affected the review. Examples of editorial changes are:

1) Capitalizing a letter with only a "redline" but not striking out the lower case letter that it replaced.
2) Changing a verb from singular to plural by adding an "s" without "redlining" the "s."
3) Deleting instead of striking-out the A, B, C, etc., following a specification title (e.g., SR3.6.6A.7).
4) Changing a bracketed reference (in the reference section) with only a "redline" for the new reference but failing to include the strike-out of the old reference.
5) In some instances, the brackets were retained (and struck-out) but the unchanged text within the brackets was not redlined.
6) Not redlining a title of a bracketed section. The methodology calls for the section title to be redlined when an entire section was bracketed.
7) Additional text not contained in the STS Bases was added to the ITS Bases by the lead FLOG member during the development of the submittal. Once it was determined to not be applicable, the text was then struck-out and remains in the ITS Bases mark-up.

Differences of the above editorial nature will not be provided as attachments to this response. The pages requiring changes that are more than editorial and are not consistent with the markup methodology are attached.

ATTACHED PAGES:

Encl 5B B3.2-10, B3.2-11, B3.2-13, B3.2-16, B3.2-20



*For core height of 12 feet

header for next page

F^N B 3.2.2

3.2.G-1

B 3.2 POWER DISTRIBUTION LIMITS

Figure B 3.2.1B-1 (page 1 of 1)
K(Z) - Normalized Fo(Z) as a Function of Core Height

(continued)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during ~~either~~ normal operation~~,~~ ~~or a postulated accident~~ operational transients, and any transient condition arising from events of moderate frequency analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod. $F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. ~~$F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.~~

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the ~~three dimensional~~ power distribution map are analyzed ~~by a computer~~ to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables. Compliance with these LCOs, along with the LCOs governing shutdown and control rod insertion and alignment, maintains the core limits on power distribution on a continuous basis.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. ~~The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to [1.3] using the [W3] CHF correlation.~~ All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an limiting initial $F^N_{\Delta H}$ as a function of power level defined by the ~~COLR~~ $F^N_{\Delta H}$ limit equation in the COLR. 3.2.G-1

redline

The LOCA safety analysis ~~indirectly models~~ also uses $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature [Ref. 3].

The fuel is protected in part by compliance with Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.~~7~~ 6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit ~~identifies~~ is representative of the coolant flow channel with the maximum enthalpy rise. This channel has ~~the least heat removal capability and thus~~ the highest probability for a DNB condition.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional ~~margin~~ allowance for higher radial peaking factors from reduced

(continued)

BASES

ACTIONS

A.2 (continued)

is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

redline

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to ~~the $F^N_{\Delta H}$~~ exceeding ~~its~~ the $F^N_{\Delta H}$ limit is identified, to the extent necessary, and corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP. SR 3.2.2.1 must be satisfied prior to increasing power above the extrapolated allowable power level or restoration of any reduced Reactor Trip System setpoints. When $F^N_{\Delta H}$ is measured at reduced power levels, the allowable power level is determined by evaluating $F^N_{\Delta H}$ for higher power levels.

redline

3.2.6-1

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

SR 3.2.2.1 is modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving MODE 1). The Note allows for power ascensions if the surveillances are not current. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. Equilibrium conditions are achieved when the core is sufficiently stable such that uncertainties associated with the measurement are valid.

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for

(continued)

BASES

LCO (continued)

of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from ~~power level~~ turbine load changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom power range detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

The AFD limits are provided in the COLR.~~Figure B 3.2.3B-1 shows typical RAOC AFD limits.~~ The AFD limits for RAOC do not depend on the target flux difference. However, the target ~~flux difference~~ AFD may be used to minimize changes in the axial power distribution. redline

Violating this LCO on the AFD could produce unacceptable consequences if a Condition ~~2, 3, or 4~~ II, III, or IV event occurs while the AFD is outside its specified limits. 3.2.G-1

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion Time of

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.2-2 **APPLICABILITY:** DC

REQUEST:

ITS 3.2.3 Axial Flux Difference
CTS 3/4.2.1 Axial Flux Difference (Diablo Canyon)
DOC 01-07-LG

Comment: The DOC/Conversion Comparison Table needs to specify where the CTS requirement is being relocated. Correct the DOC.

FLOG RESPONSE: The additional surveillance frequency when an alarm is not operable will be moved to DCPP's ECGs. DCPP has ECGs that are controlled by DCPP Department-Level Administrative Procedure (DLAP) OP1.DC16, "Control of Plant Equipment Not Required by the Technical Specifications." DCPP ECGs are similar to other plants' TRM. Changes to ECGs are made under the provisions of 10 CFR 50.59, as required by DLAP OP1.DC16 and FSAR Chapter 16. The NRC has accepted ECGs as a licensee-controlled document. This is confirmed most recently on page 2 of the NRC's safety evaluation for License Amendment 120/118 dated February 3, 1998.

ATTACHED PAGES:

Encl 3B 1

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 A	Removes reference to the special test exception in accordance with the format of NUREG-1431.	Yes	Yes	Yes	Yes
01-02 LS1	The CTS allow 15 minutes to either restore AFD or reduce THERMAL POWER to less than 90% RTP. The ITS allow 15 minutes to restore AFD and an additional 15 minutes for the reduction in the THERMAL POWER if AFD cannot be restored within the original 15 minutes.	No, this is a CAOC requirement not in CTS.	Yes	No, this is a CAOC requirement not in CTS.	No, this is a CAOC requirement not in CTS.
01-03 LG	In accordance with Wolf Creek ITS for AFD, the details regarding how AFD is measured would be moved to the Bases.	No	No	Yes	No
01-04 M	The CTS allows the 16 hours of operation outside of the target band for surveillance testing of the power range neutron flux channels. In the ITS, the practical application is identified, and the defined surveillance testing only includes the incore/excore calibration.	No, this is a CAOC requirement not in CTS.	Yes	No, this is a CAOC requirement not in CTS.	No, this is a CAOC requirement not in CTS.
01-05 M	Additional requirements are imposed in the event reactor power is required to be reduced to less than or equal to 50% RTP due to accumulated AFD penalty minutes.	No, these requirements are not in CTS.	Yes	No, these requirements are not in CTS.	No, these requirements are not in CTS.
01-06 LS2	Eliminates the requirement to reduce the power range neutron flux - high reactor Trip Setpoints.	No, refer to 01-16-LS9.	Yes	No, refer to 01-16-LS9.	No, refer to 01-16-LS9.
01-07 LG	Moves additional surveillance frequencies, if an alarm is not OPERABLE, to licensee controlled documents. This change is consistent with TSTF 110.	Yes, to ECG ~~and FSAR~~. Q3.2-2	Yes, to TRM	Yes, to USAR Chapter 16.	Yes, to FSAR Chapter 16.
01-08	Not Used.	N/A	N/A	N/A	N/A
01-09 A	The description of when the AFD is considered to be outside its [limits] will be a note to the ITS LCO, instead of the CTS SR .	Yes	Yes	Yes	Yes

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.2-3 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

ITS 3.2.1 Heat Flux Hot Channel Factor
CTS 3/4.2.2 Heat Flux Hot Channel Factor (All FLOG Plants)
DOC 02-06-A
JFD 3.2-12
ITS SR 3.2.1.1 & 3.2.1.2 Frequency

Comment: The ITS SR frequency has been changed from the STS frequency of 12 hours to 24 hours. This is based upon the incorrect justification that the CTS would allow 24 hours based upon ITS SR 3.0.3, since the CTS does not specify a frequency. Adopt the STS SR frequency of 12 hours.

FLOG RESPONSE: The change descriptions (DOC 2-06-A & JFD 3.2-12) will be revised to provide a basis for the 24 hours that is predicated on the time required to perform the surveillance.

Callaway and Wolf Creek are incorporating this change (DOC 02-06-A, JFD 3.2-12) in lieu of maintaining CTS which did not specify any completion time. DOC 02-13-LG (applicable to Callaway only) and JFD 3.2-17 are no longer used.

ATTACHED PAGES:

Encl 2	3/4 2-7
Encl 3A	2, 3
Encl 3B	3, 4
Encl 6A	2
Encl 6B	2

SURVEILLANCE REQUIREMENTS

~~4.2.2.1 The provisions of Specification 4.0.4 are not applicable.~~ 02-07-A

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if it E_Q~~(Z)~~ is within its limits by:

(new) Verifying $F_Q^C(Z)$ and $F_Q^W(Z)$ satisfy the relationships in the COLR. 02-01-LG

~~a. Using the moveable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.~~

~~b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.~~ 02-01-LG

~~c. Satisfying the following relationship:~~

$$\cancel{F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \text{ for } P > 0.5}$$

$$\cancel{F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{W(z) \times 0.5} \text{ for } P \leq 0.5}$$

~~where $F_Q^M(Z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. F_Q^{RTP} is the F_Q limit. K(z) is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(z), and W(z) are specified in the COLR.~~ 02-07-A

d. Measuring ~~$F_QM(z)$~~ $F_Q^C(Z)$ and $F_Q^W(Z)$ according to the following schedule:* 02-12-A

(new) Once after each refueling prior to THERMAL POWER exceeding 75% RTP. 02-05-M

1. Once within ~~12~~ 24 hours after ~~Upon~~ achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q^C(Z)$ ~~$E_Q(z)$~~ was last determined.* ~~or~~ and 02-06-~~A~~ M Q3.2-3

2. At least once per 31 Effective Full Power Days (EFPD) thereafter ~~whichever occurs first~~.

e. With measurements indicating

maximum over z $\frac{F_Q^{CM}(Z)}{K(Z)}$

has increased since the previous determination of $F_Q^{CM}(Z)$ either of the following actions shall be taken:

*During power escalation ~~at the beginning of each cycle~~ following shutdown, power level may be increased until ~~a~~ an equilibrium power level for extended operation has been achieved and a power distribution map obtained. 02-07-A

CHANGE NUMBER	NSHC	DESCRIPTION
01-13	A	The ACTION statement regarding restoring AFD to within limits within 15 minutes would be deleted . This has no effect on the time allowed for completion of required actions and restoring AFD to within limits is implicit in requirements for exiting the ACTION statement. Therefore, this change is administrative.
01-14		Not used.
01-15	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-16	LS9	Eliminates the requirement to reduce the power range neutron flux - high reactor trip setpoints when AFD is outside the RELAXED AXIAL OFFSET CONTROL (RAOC) limits.
02-01	LG	In accordance with NUREG-1431, the proposed change moves the details of the entire CTS LCO of the allowable $F_Q(Z)$ values, including the K(Z) and W(Z) parameters to the CORE OPERATING LIMITS REPORT (COLR) and/or Bases. Previously, only the full power value of $F_Q(Z)$, in addition to K(Z) and W(Z) had been in the COLR. Now, the dependence of $F_Q(Z)$ on THERMAL POWER is also located in the COLR. Details of the $F_Q(Z)$ measurement, including the treatment of uncertainties, are moved to the Bases. The Required Actions are rewritten for consistency with NUREG-1431. The specific changes include the more appropriate use of $F_Q^C(Z)$ and $F_Q^W(Z)$ versus $F_Q(Z)$.
02-02	LS3	The Required Actions are rewritten for consistency with NUREG-1431 and industry Traveler TSTF-95. The specific changes include the relaxation of the Completion Time requirement to reduce the high neutron flux reactor Trip Setpoints [from 4 hours] to 72 hours. The reduction of the setpoints is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the amount of work required to be done to reduce the setpoints, the small likelihood of a severe transient in this time period, and the prompt reduction in THERMAL POWER required upon discovery of the out-of-limit condition.
02-03	M	For consistency with NUREG-1431, the requirement is added to be in at least MODE 2 within 6 hours should any of the ACTIONS not be completed within the required time. This requirement is more restrictive than the previous requirement to enter CTS 3.0.3, which allowed 1 hour before the 6 hour shutdown requirement became effective.
02-04	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-05	M	Consistent with NUREG-1431, $F_Q^C(Z)$ and $F_Q^W(Z)$ must be verified to be within limits prior to exceeding 75 percent RTP after each refueling. This requirement is not explicit in the CTS. The TS are made more restrictive by stating this requirement.
02-06	~~A~~ M	~~Consistent with the Bases of CTS 4.0.3, which allows 24 hours for completing SRs that become applicable when an exception to Specification 4.0.4 is allowed, the frequency for assessing $F_Q(Z)$ is clarified by requiring that the measurement be performed within 24 hours after reaching equilibrium conditions.~~ Insert Q3.2-3

INSERT for 02-06-M:

02-06 M In the ITS SR 3.2.1.1 and SR 3.2.1.2, a time limit for assessing $F_Q(Z)$ after reaching equilibrium conditions is specified. Because the CTS does not have such a time restriction, this change is more restrictive. The time limit for completion of this surveillance, 24 hours following the establishment of equilibrium conditions, has been selected based on plant experience. Twenty-four hours is a reasonable time for obtaining and evaluating a flux map and then completing the required procedural steps associated with this surveillance. Further, the 24 hour time limit does not allow for plant operation in an uncertain condition for a protracted time period. The time limit of 24 hours is consistent with Amendment No. 116 for Wolf Creek in which the NRC approved allowing the performance of a flux map 24 hours after achieving equilibrium conditions from a Thermal Power reduction required with QPTR determined to exceed 1.02.

CHANGE NUMBER	NSHC	DESCRIPTION
02-07	A	The footnote allowing the power to be increased until the THERMAL POWER for extended operation has been achieved has been incorporated in the note preceding SR 3.2.1.1 and SR 3.2.2.1 in the ITS allowing power to be increased until an equilibrium power level has been achieved. This footnote replaces the Specification 4.0.4 exemption in the CTS. Therefore, the change is administrative, and no technical changes would result.
02-08	LS4	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-09	M	The optional action to comply with CTS 3.2.2, if $F_Q^W(Z)$ exceeds its limit would be deleted. This eliminates an option and is more restrictive.
02-10		Not used.
02-11	LS15	The proposed change, consistent with NUREG-1431, would delete the requirement that the reactor be in at least HOT STANDBY while performing the overpower ΔT Trip Setpoint reduction. It is sufficient to reduce power 1 percent for each 1 percent $F_Q(z)$ exceeds its limit and then perform the required trip setpoint reduction at reduced power. The CTS requirement to be in at least HOT STANDBY is not a Westinghouse design basis requirement.
02-12	A	$F_Q^W(Z)$ must be verified to be within limits whenever $F_Q(Z)$ is measured as required by the CTS. 032-3
02-13	LS ~~Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).~~ Not used.	
02-14	M	This change is similar to CN 02-08-LS4 for the JLS. For DCPP, there is currently no time limit; therefore, the imposition of a time limit is a more restrictive change.
03-01	LG	The details of the $F^N_{\Delta H}$ limits would be moved to the COLR. Previously, the equation for the dependence of $F^N_{\Delta H}$ on THERMAL POWER had been located in the LCO and the COLR. The full power limit value of $F^N_{\Delta H}$ and the power factor multiplier had been located only in the COLR. Now, the equation is also located only in the COLR. Definitions and details of the measurement, including the treatment of uncertainties, are moved to the Bases. The Required Actions are rewritten for consistency with NUREG-1431. The changes are acceptable because they remove details not required to be in TS to support operational safety.
03-02	LS5	The Completion Times would be revised to be consistent with NUREG-1431. The adequacy of these completion times is discussed in the applicable Bases section of NUREG-1431. In summary, 4 hours (versus 2 hours in the CTS) is provided to attempt to restore $F^N_{\Delta H}$ to within its limit or to reduce power to below 50 percent RTP.
03-03	M	The requirement to reduce power to less than or equal to 5 percent RTP (exit MODE 1) within the next 6 hours is added in lieu of the use of CTS LCO 3.0.3. This requirement is more restrictive than the previous requirement to enter LCO 3.0.3, because LCO 3.0.3 allowed 1 hour before the 6-hour shutdown requirement became effective.
03-04	LS6	With the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) having been outside its limits, the current TS require that within 24 hours after exceeding the $F^N_{\Delta H}$ limit, an incore flux map be performed to verify that the $F^N_{\Delta H}$ has been restored

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-16 LS9	Eliminates the requirement to reduce the power range neutron flux - high reactor Trip Setpoints when AFD is outside the RAOC limits.	Yes	No, CPSES uses CAOC methodology; change applies to RAOC methodology.	Yes	Yes
02-01 LG	The details of the entire CTS LCO of the allowable $F_Q(Z)$ values, including the K(Z) and W(Z) parameters, would be moved to the COLR and/or Bases.	Yes	Yes	Yes	Yes
02-02 LS3	Consistent with Traveler TSTF-95, the Required Actions are relaxed to extend the time allowed to reduce the high neutron flux reactor Trip Setpoints [from 4 hours] to 72 hours.	Yes	Yes	Yes	Yes
02-03 M	The required ACTIONS are revised to include the addition of a requirement to be in at least MODE 2 within 6 hours should any of the ACTIONS not be completed within the required time.	Yes	Yes	Yes	Yes
02-04 M	$F_Q^W(Z)$ must be verified to be within limits whenever $F_Q(Z)$ is measured, not just at the time of target flux determination, as required by the CTS.	No, requirement not in CTS.	Yes	No, requirement not in CTS.	No, requirement not in CTS.
02-05 M	$F_Q^C(Z)$ and $F_Q^W(Z)$ must be verified to be within limits prior to exceeding 75% RTP after each refueling.	Yes	Yes	Yes	Yes
02-06 ~~A~~ M	~~Consistent with the Bases of CTS 4.0.3, which allows 24 hours for completing SRs that become applicable when an exception to Specification 4.0.4 is allowed,~~ The frequency for assessing $F_Q(Z)$ is ~~clarified~~ revised by requiring that the measurement be performed within 24 hours after reaching equilibrium Conditions	Yes	Yes Q 3.2-3	~~No, CLB will be retained.~~ Yes	~~No, CLB will be retained.~~ Yes

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-07 A	The footnote allowing power to be increased until the THERMAL POWER for extended operation has been achieved has been incorporated in the notes preceding ITS SR 3.2.1.1 and SR 3.2.2.1 for achieving equilibrium power level. This footnote replaces the Specification 4.0.4 exemption in the CTS.	Yes	Yes	Yes	Yes
02-08 LS4	The time allowed to reduce the acceptable operation limits on AFD is changed to 4 hours, consistent with Traveler TSTF- 99.	No, refer to 02-14-M.	Yes	Yes	Yes
02-09 M	The optional ACTION to comply with CTS 3.2.2 if $F_Q^W(Z)$ exceeds its limit would be deleted. This eliminates an option and is more restrictive.	Yes	No, identified option not in CTS.	Yes	Yes
02-10	Not Used.	N/A	N/A	N/A	N/A
02-11 LS15	The ACTION requirement, with $F_Q(Z)$ exceeding its limit, that the overpower Delta-T Trip Setpoint reduction be performed in at least HOT STANDBY would be deleted.	Yes	No, requirement not in CTS.	No, requirement not in CTS.	No, requirement not in CTS.
02-12 A	$F_Q^W(Z)$ must be verified to be within limits whenever $F_Q(Z)$ is measured as required by the CTS.	Yes	No, not required by CTS.	Yes	Yes
02-13 ~~LG~~	~~The definition of extended operation (expected operation at a power level for greater than 72 hours) is moved to the Bases.~~ Not Used.	~~No, term not in CTS.~~ NA	~~No, term not in CTS.~~ NA	~~No, definition not in CTS.~~ NA	~~Yes~~ NA Q3.2-3
02-14 M	Similar to CN 02-08-LS4. For DCPP, there is currently no time limit; therefore, the imposition of a time limit is a more restrictive change.	Yes	No	No	No
03-01 LG	Moves the details of the $F^N_{\Delta H}$ limits to the COLR and/or Bases.	Yes	Yes	Yes	Yes
03-02 LS5	Revises the completion time to 4 hours (versus 2 hours in the CTS to attempt to restore $F^N_{\Delta H}$ to within its limit or to reduce power to below 50% RTP.	Yes	Yes	No, 4 hours already in CTS.	Yes

3.2-08 Consistent with Traveler TSTF-99, the LCO 3.2.1 (F_Q Methodology), Required ACTION B.1. Completion Time for the reduction of the AFD limits if $F^W_Q(Z)$ is not within limits is increased from 2 hours to 4 hours. This makes it consistent with the Completion Time associated with Required ACTION A.2. of LCO 3.2.1 (F_{XY} methodology). The change is acceptable because it eliminates an inconsistency in the ITS.

3.2-09 For consistency with CTS 3.2.4 and ITS 3.3.1, Condition D, the breakpoints for the Applicability of the surveillances in the notes in ITS SR 3.2.4.1 and SR 3.2.4.2 are modified to be applicable at less than or equal to 75 percent RTP, and greater than 75 percent RTP, respectively. ~~This is an~~ administrative change that retains CTS requirements, and is consistent with TSTF-241. Q3.2-6

3.2-10 Consistent with Traveler TSTF-110, this change moves requirements for increased surveillance frequencies in the event of inoperable alarms to licensee controlled documents. This change is acceptable because it removes requirements regarding alarms and alarm responses that are not necessary to be in the TS to protect public health and safety. Q3.2-3

3.2-11 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

3.2-12 ~~Consistent with CTS,~~ The required time for completion of a flux map (3.2.1.2.) for determination of the heat flux hot channel factor is changed from 12 hours to 24 hours after achieving equilibrium Conditions. The proposed change affects SR 3.2.1.1 and SR ~~3.2.2.1~~. Based on plant experience, The proposed time (24 hours) is a reasonable time period for the completion of the surveillance and does not allow for plant operation in an uncertain condition for a protracted time period. This change is consistent with the TS requirements of Specification 3.0.4 (and associated Bases) that allow 24 hours for the completion of a surveillance after prerequisite plant conditions are attained and for which an exception to Specification 4.0.4 was provided.

Insert

3.2-13 This change retains the CTS for the performance of peaking factor determinations following plant shutdowns. The CTS, through the exemption to Specification 4.0.4, allows prerequisite plant conditions to be obtained prior to requiring that the surveillance be completed. Insert

3.2-14 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B). Q3.2-4

3.2-15 This change incorporates Traveler TSTF-109. ACTION A.2. would require the QPTR be determined rather than performing a specific surveillance because more than one surveillance can be used to determine QPTR. SR 3.2.4.1 was revised to retain allowance that SR 3.2.4.2 may be performed in lieu of SR 3.2.4.1. ~~The note for SR 3.2.4.2 is changed to require performance if one "or more" QPTR inputs are inoperable with RTP > 75 percent.~~ Q3.2-10 These changes are acceptable because they clarify the ITS regarding frequency and use of incore flux monitoring for QPTR measurement. The changes reflect that incore detectors provide an acceptable QPTR determination during all plant Conditions.

3.2-16 This change would require both transient and static F_Q measurements be determined when performed for Required ACTIONS 3.2.4 A.3 and A.6. The intent of the Required ACTIONS is to verify that $F_Q(Z)$ is within its limit. $F_Q(Z)$ is approximated by $F^C_Q(Z)$ (which is obtained via SR 3.2.1.1) and $F^W_Q(Z)$ (which is obtained via SR 3.2.1.2). Thus, both $F^C_Q(Z)$ and $F^W_Q(Z)$ must be established to verify $F_Q(Z)$. This change is consistent with Traveler WOG-105.

3.2-17 ~~Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).~~ Q3.2-3

Not used

3.2-18 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

3.2-19 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

The note and frequency for SR 3.2.4.2 are revised consistent with typical presentation formats that provide for a period of time after establishing conditions. Q3.2-10

3.2-20 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B). Q 3.2-1

INSERT for 3.2-12:

obtaining and evaluating a flux map and then completing the procedural steps associated with this surveillance. Further, the 24 hours time period does not allow for plant operation in an uncertain condition for a protracted time period.

	TECH SPEC CHANGE	APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.2-11	This change rearranges the notes applicable to LCO 3.2.3. AFD (Constant Axial Offset Control plants only, CPSES specific) from three separate notes affecting different portions of the LCO into a common note block. This change is consistent with TSTF-164.	No	Yes	No	No
3.2-12	~~Consistent with CTS, the required time for completion of a flux map for determination of the heat flux hot channel factor is changed from 12 hours to 24 hours after achieving equilibrium conditions. The proposed change affects SR 3.2.1.1 and SR 3.2.2.1.~~ Insert	Yes	Yes Q 3.2-3	~~No, see CN 3.2-17.~~ Yes	~~No, see CN 3.2-17.~~ Yes
3.2-13	This change retains the CTS for the performance of peaking factor determinations following plant shutdowns.	Yes	Yes	Yes	Yes
3.2-14	This change retains the Wolf Creek current licensing basis (License Amendment 61) for Required ACTION 3.2.2 A.2.	No	No	Yes	No
3.2-15	This change partially incorporates the industry traveler TSTF-109. ACTION A.2 now requires the QPTR be determined rather than performing a specific surveillance because more than 1 surveillance can be used to determine QPTR. ~~The note for SR 3.2.4.2 is changed to require performance if one "or more" QPTR inputs are inoperable.~~	Yes Q 3.2-10	Yes	Yes	Yes
3.2-16	Requires both transient and static F_Q measurements when performed for Required ACTIONS 3.2.4 A.3 and A.6.	Yes	Yes	Yes	Yes
3.2-17	~~The frequency requirement for performing F_Q measurements has been revised to conform to the CTS which did not specify a Completion Time.~~	~~No, see CN 3.2-12.~~ NA	~~No, see CN 3.2-12.~~ NA	~~Yes~~ NA	~~Yes~~ NA Q 3.2-3
3.2-18	This change modifies the QPTR requirements in NUREG-1431, Rev. 1, for Wolf Creek, to retain ~~some~~ the CTS requirements ~~and to incorporate revisions to Required ACTIONS proposed by Traveler WOG-95.~~ approved in Amendment No. 116.	No	No	Yes Q 3.2-6	No

INSERT for 3.2-12:

Based on plant experience of a reasonable time to obtain, evaluate, and complete associated procedural steps, the time allocated for the performance of the F_Q surveillance is set to 24 hours after achieving equilibrium conditions. The proposed change affects SR 3.2.1.1 and SR 3.2.1.2.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.2-4 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

ITS 3.2.2 Nuclear Enthalpy Rise Hote Channel Factor
CTS 3/4.2.3 Nuclear Enthalpy Rise Hot Channel (All FLOG Plants)
DOC 02-07-A
JFD 3.2-13
SR 3.2.2.1 NOTE and related Bases.

Comment: Justify the need for the note related to permitting power ascension after shutdown to a level at which a power distribution map is obtained. It appears that this note is unnecessary, considering the phraseology of the SR Frequency ("Once after each refueling prior Thermal Power exceeding 75% RTP"). Explain the need for this note. The SR 3.2.2.1 Bases also mentions "(leaving Mode 1)" which appears to be the incorrect mode.

FLOG RESPONSE: The Note, as described in JFD 3.2-13, was incorporated to address the rare situation where, during a mid-cycle shutdown, through further review of the previous surveillance, it was determined that the surveillance was invalid; or the required surveillance frequency is not met due to the shutdown. The amended Note would be required to return the reactor to a power level at which a new surveillance could be performed.

The "leaving MODE 1" clarification is based on the Applicability of the LCO (MODE 1, only) and is intended to avoid confusion in a scenario where the plant may be taken off-line (typically, MODE 2), but not "shutdown" (commonly considered to be MODE 3 or lower).

ATTACHED PAGES:

Encl 6A 2

3.2-08 Consistent with Traveler TSTF-99, the LCO 3.2.1 (F_Q Methodology), Required ACTION B.1. Completion Time for the reduction of the AFD limits if $F^W_Q(Z)$ is not within limits is increased from 2 hours to 4 hours. This makes it consistent with the Completion Time associated with Required ACTION A.2. of LCO 3.2.1 (F_{XY} methodology). The change is acceptable because it eliminates an inconsistency in the ITS.

3.2-09 For consistency with CTS 3.2.4 and ITS 3.3.1, Condition D, the breakpoints for the Applicability of the surveillances in the notes in ITS SR 3.2.4.1 and SR 3.2.4.2 are modified to be applicable at less than or equal to 75 percent RTP, and greater than 75 percent RTP, respectively. ~~This is an~~ administrative change that retains CTS requirements, and is consistent with TSTF-241. Q3.2-6

3.2-10 Consistent with Traveler TSTF-110, this change moves requirements for increased surveillance frequencies in the event of inoperable alarms to licensee controlled documents. This change is acceptable because it removes requirements regarding alarms and alarm responses that are not necessary to be in the TS to protect public health and safety. Q3.2-3

3.2-11 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

3.2-12 ~~Consistent with CTS,~~ The required time for completion of a flux map 3.2.1.2. for determination of the heat flux hot channel factor is changed from 12 hours to 24 hours after achieving equilibrium Based on plant experience, Conditions. The proposed change affects SR 3.2.1.1 and SR ~~3.2.2.1~~. The proposed time (24 hours) is a reasonable time period for the completion of the surveillance and does not allow for plant operation in an uncertain condition for a protracted time period. This change is consistent with the TS requirements of Specification 3.0.4 (and associated Bases) that allow 24 hours for the completion of a surveillance after prerequisite plant conditions are attained and for which an exception to Specification 4.0.4 was provided.

Insert

3.2-13 This change retains the CTS for the performance of peaking factor determinations following plant shutdowns. The CTS, through the exemption to Specification 4.0.4, allows prerequisite plant conditions to be obtained prior to requiring that the surveillance be completed. Insert

3.2-14 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B). Q3.2-4

3.2-15 This change incorporates Traveler TSTF-109. ACTION A.2. would require the QPTR be determined rather than performing a specific surveillance because more than one surveillance can be used to determine QPTR. SR 3.2.4.1 was revised to retain allowance that SR 3.2.4.2 may be performed in lieu of SR 3.2.4.1. ~~The note for SR 3.2.4.2 is changed to require performance if one "or more" QPTR inputs are inoperable with RTP > 75 percent.~~ Q3.2-10 These changes are acceptable because they clarify the ITS regarding frequency and use of incore flux monitoring for QPTR measurement. The changes reflect that incore detectors provide an acceptable QPTR determination during all plant Conditions.

3.2-16 This change would require both transient and static F_Q measurements be determined when performed for Required ACTIONS 3.2.4 A.3 and A.6. The intent of the Required ACTIONS is to verify that $F_Q(Z)$ is within its limit. $F_Q(Z)$ is approximated by $F^C_Q(Z)$ (which is obtained via SR 3.2.1.1) and $F^W_Q(Z)$ (which is obtained via SR 3.2.1.2). Thus, both $F^C_Q(Z)$ and $F^W_Q(Z)$ must be established to verify $F_Q(Z)$. This change is consistent with Traveler WOG-105.

3.2-17 ~~Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).~~ Q3.2-3
Not used

3.2-18 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

3.2-19 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
The note and frequency for SR 3.2.4.2 are revised consistent with typical presentation formats that provide for a period of time after establishing conditions. Q3.2-10

DCPP Description of ~~Changes to Improved TS~~

3.2-20 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B). Q3.2-1

Insert for JFD 3.2-13

The note was incorporated to address the rare situation where, during a mid-cycle shutdown, through further review of the previous surveillance, it was determined that the surveillance was invalid; or the required surveillance frequency is not met due to the shutdown.. The amended Note would be required to return the reactor to a power level at which a new surveillance could be performed.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.2-6 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

ITS 3.2.4 Quadrant Power Tilt Ratio
CTS 3/4.2.4 Quadrant Power Tilt Ratio (All FLOG Plants)
DOC 04-01-A
JFD 3.2-05
ITS Required Action A.5

Comment: The ITS proposes to change the STS wording for Required Action A.5 from "Calibrate excore detectors to show zero QPTR," to "Normalize excore detectors to eliminate tilt," based upon WOG-95 (and rejected TSTF-25). A preferred wording would be that proposed in the Comanche Peak CTS mark-up, "Calibrate excore detectors to show zero Quadrant Power Tilt." What is status of WOG-95?

FLOG RESPONSE: Traveler WOG-95 was transmitted to the NRC in February 1998 as TSTF-241. The FLOG is incorporating TSTF-241 including the latest revisions discussed at the June 1998 WOG MERITS Mini-Group meeting. These revisions corrected errors made during the development of TSTF-241.

Additionally, Wolf Creek submitted a License Amendment Request to CTS 3/4.2.4, Quadrant Power Tilt Ratio, on February 4, 1998, which was approved on April 27, 1998, in Amendment No. 116. This amendment incorporated the changes proposed in TSTF-241.

The FLOG believes that it is appropriate to incorporate the proposed TSTF-241 changes based on the NRC approval of the Wolf Creek amendment request.

ATTACHED PAGES:

Encl 2	3/4 2-18
Encl 3A	4 and 6
Encl 3B	5 and 6
Encl 4	Table of Contents
Encl 5A	Traveler status page, 3.2-10, and 3.2-11
Encl 5B	B 3.2-25, B 3.2-26, B3.2-27
Encl 6A	1 and 2
Encl 6B	1 and 2

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 ~~but less than or equal to 1.09:~~ 04-06-LS13

~~1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:~~ 04-02-LS10

~~a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or~~

~~b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.~~

2. Within 2 hours after each QPTR determination ~~either~~: 2-01-A Q3.2-6

~~a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or~~

b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 ~~and similarly reduce the Power Range Neutron Flux High Trip Setpoints within the next 4 hours~~. 04-05-LS11 04-06-LS13

(new *) At least once per 12 hours, calculate QUADRANT POWER TILT RATIO ~~and reduce THERMAL POWER by at least 3% from RATED THERMAL POWER for each 1% of QUADRANT POWER TILT RATIO in excess of 1~~, and Q3.2-6 04-05-LS11 04-01-A 04-06-LS13

(new) Within 24 hours, and once per 7 days thereafter, confirm that the Heat Flux Hot Channel Factor $F_Q^C(Z)$ is within its limit by performing Surveillance Requirement 4.2.2.2 and confirm that Nuclear Enthalpy Rise Hot Channel Factor $F^N_{\Delta H}$ is within its limit by performing Surveillance Requirement 4.2.3.2. 04-05-LS11

(new **) Prior to increasing THERMAL POWER above the limit of Actions a.2.b and new*: 04-05-LS11

Re-evaluate the safety analyses and confirm that the results remain valid for the duration of operation under this condition, and then 04-06-LS13

restore QPTR to within limit Q3.2-6

Normalize excore detectors to ~~eliminate tilt~~. 04-01-A

After Action (new**) is implemented and within 24 hours after reaching RTP or within 48 hours after increasing THERMAL POWER above the limit of Required Actions a.2.b and (new*), confirm that $F_Q(Z)$ is within its limits by performing Surveillance Requirement 4.2.2.2 and that $F^N_{\Delta H}$ is within its limits by performing Surveillance Requirement 4.2.3.2; and 04-05-LS11

If the requirements above are not met, reduce thermal power to ≤ 50% RATED THERMAL POWER within the next 4 hours. 04-05-LS11

~~3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and~~ 04-06-LS13 04-05-LS11

CHANGE NUMBER	NSHC	DESCRIPTION
		to within its limits. If this activity is not completed, the CTS require that the plant be taken to MODE 2 within the next 2 hours. The proposed change, revises the time allowed to reduce the reactor power to a Condition where the LCO does not apply (MODE 2, < 5 percent RTP) to be consistent with NUREG-1431. The adequacy of these Completion Times are discussed in the applicable Bases section of NUREG-1431. In summary, 6 hours (versus 2 hours in the CTS) is provided to perform an orderly shutdown of the plant. The proposed change is acceptable based on operational experience regarding performance of an orderly plant shutdown together with the negligible probability of an accident occurring during the extended shutdown interval.
03-05	M	If the enthalpy rise hot channel factor ACTION statements requiring flux mapping and correction of the cause or power reductions are entered, they must be completed, even if compliance with the LCO is restored. These requirements from NUREG-1431 are more restrictive than the corresponding requirements from CTS.
03-06	A	Consistent with NUREG-1431, a note would be added to state that THERMAL POWER does not need to be reduced below the power required by ACTION A. in order to comply with the series of flux maps required by ACTION C. This is a clarification of the CTS in that if compliance with the LCO is restored prior to reducing power level below 50 percent, flux maps need only be performed for those plateaus traversed. If power level did not drop below 95 percent, no flux map would be required. No flux map would be required by ACTION C, but would be required by ACTION [B].
03-07	A	The Parameter R, which is a derived value based on $F^N_{\Delta H}$, would be deleted. Using a combination of reactor coolant system (RCS) total flow rate and R to determine operation in the acceptable region of CTS Figures 3.2-3a and 3.2-3b would be deleted. RCS total flow would be moved to the departure from nucleate boiling (DNB) parameters specification. $F^N_{\Delta H}$ would be a separate specification from the requirements of RCS flow. This is an administrative change since the limit for R in Figures 3.2-3a and 3.2-3b is a constant value (1.00); therefore, the equation deriving the Parameter R can be reduced and shown in terms of $F^N_{\Delta H}$.
03-08	A	Figures 3.2-3a and 3.2-3b would be revised to display only parameters RCS flow and power level in tables. The tables show required reduction in power level to account for reduced RCS total flow. The parameter R would be deleted since it is a constant value. This is an administrative change since the only change is in the presentation of these figures.
03-09	M	The allowed ACTIONS for reduced RCS flow would be deleted and replaced with the more restrictive ITS requirement to restore RCS flow within 2 hours.
03-10	LG	The note on measurement uncertainty for flow is moved to the Bases.
04-01	A restored to within limit	Clarifies that when the excore detectors are calibrated, the quadrant power tilt is ~~zeroed out~~ (The QUADRANT POWER TILT RATIO (QPTR) is normalized to unity.) This requirement from NUREG-1431 as modified by Traveler TSTF-~~25~~ 241, 3.2-6 is consistent with the CTS ACTION requirements for verifying QPTR is within limit during power escalation subsequent to identifying and correcting the cause of QPTR out of limit. Additionally, the actions are modified to provide the appropriate allowance for subsequent power reductions based on subsequent determination of QPTR.
04-02	LS10	The required CTS ACTION to calculate QPTR once per hour until THERMAL

CHANGE NUMBER	NSHC	DESCRIPTION
		violated; and (3) the ITS required ACTIONS prior to and subsequent to power ascension provide assurance that POWER OPERATION at or near RTP will be in accordance with the safety analyses, and therefore acceptable.
04-07	A	The statement that Specification 3.0.4 does not apply is no longer needed as revised ACTIONS permit continued operation for an unlimited period of time.
04-08		Not used. {WC 3.2-001}
04-09	A	~~Consistent with NUREG-1431, a note is added to permit 3 OPERABLE excore channels to be used to calculate QPTR when 1 channel is inoperable and power is ≥75 percent.~~ Insert (Q 3.2-6)
04-10	LS14	~~Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).~~ Not Used.
04-11	A	
05-01	LG	The designation of how instrument uncertainties are treated (nominal, in the analysis or in the development of the TS limit) is moved to the Bases. The movement of this level of detail out of the specification is consistent with NUREG-1431 and is an example of removing unnecessary details from the TS in accordance with 10 CFR 50.36.
05-02	LS7	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-03	LG	~~Consistent with NUREG-1431, the requirement to perform a CHANNEL CALIBRATION on the RCS flow meters at least once per 18 months and the requirement to normalize the channels are moved to the Bases for the RCS flow low reactor trip function in ITS Section 3.3.1.~~ (CP 3.2-001)
05-04	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-05	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-06	LS8	In accordance with NUREG-1431, if any of the DNB related parameters of pressure, temperature, or RCS flow are found to be outside their limits, the time period required to perform a power reduction would be extended to 6 hours. The DNB related parameters of RCS average temperature, pressurizer pressure, and RCS flow rate are maintained within specified limits in order to ensure consistency with the assumed initial conditions of the accident analyses. The limits placed on the RCS temperature, pressure, and flow ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed. Compliance with the above limits is verified every 12 hours. If a parameter is found to be outside the required limit, 2 hours are allowed in order to restore the parameter to within the limit. If the parameter is not restored to compliance within the required time, the plant must be shut down. The revised Completion Time of 6 hours is acceptable to allow transition to the required plant Conditions in an orderly manner without unnecessarily initiating any undue plant transients and on the small likelihood of a severe event occurring during the extended time period.
05-07	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-08		Not used.
05-09	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-11	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). {WC 3.2-001}

INSERT for 04-09-A:

04-09 A Consistent with NUREG-1431, Rev. 1, a Note is added to permit three OPERABLE excore channels to be used to calculate QPTR when one channel is inoperable and power is ≤ 75% RTP. This is consistent with operations permitted by the combination of the CTS definition of QPTR and the CTS surveillance requirement for verifying QPTR by use of the incore detectors when an excore channel is inoperable.

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-03 M	The requirement to reduce power to less than or equal to 5% RTP (exit MODE 1) within the next 6 hours is added in lieu of the use of CTS 3.0.3.	Yes	Yes	Yes	Yes
03-04 LS6	Revises the time allowed to reduce the reactor power to a Condition where the LCO does not apply (MODE 2, < 5% RTP) to 6 hours (versus 2 hours in the CTS).	Yes	Yes	No, 6 hours already in CTS.	Yes
03-05 M	Requires $F_{\Delta H}$ ACTION Statements to be completed if entered.	Yes	Yes	Yes	Yes
03-06 A	A note would be added to state that THERMAL POWER does not need to be reduced in order to comply with the series of flux maps that must be taken upon a return to power.	Yes	Yes	Yes	Yes
03-07 A	The DCPP specific parameter R, which is a derived value based on $F^N_{\Delta H}$, would be deleted.	Yes	No	No	No
03-08 A	Revise DCPP specific Figures 3.2-3a and 3.2-3b to display only parameters RCS flow and power level. The Parameter R would be deleted since it is a constant value.	Yes	No	No	No
03-09 M	The DCPP specific allowed ACTIONS for reduced RCS flow would be deleted and replaced with the more restrictive ITS requirement to restore RCS flow within 2 hours.	Yes	No	No	No
03-10 LG	The DCPP note on measurement uncertainty for flow is moved to the Bases.	Yes	No	No	No
04-01 A	Clarifies that when the excore detectors are calibrated, the quadrant power tilt is zeroed out. (The QPTR is normalized to unity.)	Yes	Yes	~~Yes~~ No, incorporated by Amendment 116.	Yes @ 3.2-6

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
04-02 LS10	The required CTS ACTION to calculate QPTR once per hour until THERMAL POWER was within limit or reduced to less than 50% RTP would be eliminated and replaced by new requirements from NUREG-1431. This represents a reduction in requirements for monitoring and reducing power.	Yes	No, requirement not in CTS.	~~Yes~~ No, incorporated by Amendment 116.	Yes Q 3.2-6
04-03 LG	The details regarding obtaining QPTR using the incore detectors would be moved to the Bases.	Yes	No, details not in CTS.	Yes	Yes
04-04 LS12	The requirements and capabilities for measuring QPTR when 1 or more excore detector channels are inoperable are clarified.	Yes	Yes, see also change 01-29-LS in Section 1.0.	No, see CN ~~04-10-LS14.~~ 04-11-A	Yes WC 3.2-001
04-05 LS11	CTS ACTIONS requiring QPTR to be restored within 24 hours, QPTR to be verified during return to power, and power range neutron flux-high trip setpoint to be reset to ≤ 55% would be eliminated.	Yes	No, requirement not in CTS.	No, ~~see CN 04-10-LS14.~~ Amendment 116 maintained resetting flux trip setpoint.	Yes Q 3.2-6
04-06 LS13	CTS ACTIONS involving QPTR exceeding 1.09 would be eliminated in conformance with NUREG-1431.	Yes	No, actions not in CTS.	~~Yes~~ No, deleted by Amendment 116.	Yes Q 3.2-6
04-07 A	The statement that Specification 3.0.4 does not apply is no longer needed as revised ACTIONS permit continued operation for unlimited period of time.	Yes	No, exception not in CTS.	Yes	Yes
04-08 A	Not Used.	N/A	N/A	NA	N/A
04-09 A	Consistent with NUREG-1431, a note is added to permit 3 OPERABLE excore channels to be used to calculate QPTR when 1 channel is inoperable and power is ≤ 75%.	Yes	No, already in CTS.	~~No, maintaining CTS wording.~~ Yes	Yes WC 3.2-001
04-10 ~~LS14-e~~	~~The allowed time for the requirement to reset the power range neutron flux-high setpoint during power reduction required by QPTR ACTIONS would be extended to 72 hours for Wolf Creek.~~	~~No~~ NA	~~No~~ NA	~~Yes~~ NA	~~No~~ NA Q 3.2-6
04-11 A	Insert WC 3.2-001	No - See CN 04-LS-12	No-See CN 04-LS12	Yes	No-See CN 04-LS12

INSERT for 04-11-A

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
04-11 A	A note its added to Wolf Creek CTS SR 4.2.4.2 to indicate that the surveillance is not required to be performed until 12 hours after input from one Power Range Neutron Flux channel is inoperable with THERMAL POWER > 75% RTP. Additionally, a note is added to CTS SR 4.2.4.1 to indicate that CTS SR 4.2.4.2 may be performed in lieu of this surveillance requirement to confirm the indication of the remaining three excore channels.	No - See CN 04-04-LS-12	No - See CN 04-04-LS-12	Yes	No - See CN 04-04-LS-12

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

Industry Travelers Applicable to Section 3.2

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-24	Not Incorporated	NA	Not NRC approved as of traveler cut-off date
TSTF-95	Incorporated	3.2-06	Approved by NRC.
TSTF-97	Incorporated	3.2-07	Approved by NRC.
TSTF-98, R1	Incorporated	3.2-03	
TSTF-99	Incorporated	3.2-08	Approved by NRC.
TSTF-109	Incorporated	3.2-15	Approved by NRC.
TSTF-110, ~~R1~~ R2	Incorporated	3.2-10	Approved by NRC. TR 3.2-004
TSTF-112, R1	~~Not~~ Incorporated	~~NA~~ 3.2-20	~~Not NRC approved as of traveler cut-off date~~
TSTF-136	Incorporated	NA	Approved by NRC. TR 3.2-004
TSTF-164	Incorporated	3.2-11	Applicable to CAOC only. (CPSES)
~~WOG-95, proposed Rev. 2~~ TSTF-241	Incorporated	3.2-05 ~~3.2-18~~ 3.2-09	Q 3.2-6
WOG-105	Incorporated	3.2-16	

Approved by NRC. Applicable to CAOC plants (CPSES ONLY) Q 3.2-1

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPTR not within limit.	A.1 Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination 3.2-05 Q3.2-6
	AND	
	A.2 Determine QPTR ~~Perform SR 3.2.4.1 and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.~~	Once per 12 hours 3.2-15 3.2-05 Q3.2-6
	AND	
	A.3 Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	24 hours 3.2-16
		AND
		Once per 7 days thereafter Q3.2-6
After achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1	AND	
	A.4 Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Actions A.1 ~~and A.2~~ ED
	AND	

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.5 ----------NOTE--------- 1. Perform Required Action A.5 only after Required Action A.4 is completed. ----------	3.2-05 Q3.2-6
2. Required Action A.6 shall be completed whenever Required Actions A.5 is performed.	~~Calibrate~~ Normalize excore detectors to ~~eliminate tilt~~ ~~show zero QPTR~~. restore QPTR to within limit.	Prior to increasing THERMAL POWER above the limit of Required Actions A.1 ~~and~~ ~~A.2~~ 3.2-05 ED
	AND	
	A.6 ---------NOTE--------- ~~Perform~~ Required Action A.6 ~~must be completed when~~ ~~only after~~ Required Action A.5 is ~~implemented~~ ~~completed~~. Remove Strike-out ----------	Q 3.2-6
	Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	~~Within~~ 24 hours after ~~reaching~~ ~~RTP~~ achieving equilibrium conditions not to exceed 3.2-16 3.2-05 Q3.2-6 OR ~~Within~~ 48 hours after increasing THERMAL POWER above the limit of Required Actions A.1 ~~and~~ ~~A.2~~ ED
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

BASES

ACTIONS

A.1 (continued)

time to a identify the cause and correct the tilt. Note that ~~the~~ power reduction ~~itself~~ may cause a change in the tilted condition. (Q3.2-6)

Insert

A.2

After completion of Required Action A.1, the QPTR ~~alarm may still be in its alarmed state~~ may still exceed its limits. ~~As such,~~ a Any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to support flux mapping.

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit. (Q3.2-6) (DC 3.2-E2)

to verify peaking factors and that the incore quadrant power tilt and QPTR are consistent

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded

(continued)

Enclosure 5B - page B3.2-25

INSERT for A.1

The maximum allowable THERMAL POWER level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require a THERMAL POWER reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable THERMAL POWER level. Decreases in QPTR would allow raising the maximum allowable THERMAL POWER level and increasing THERMAL POWER up to this revised limit.

BASES

ACTIONS

A.4 (continued)

and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This ~~re-~~evaluation is required to ensure that, before increasing THERMAL POWER ~~to~~ above the limit of Required Actions A.1 ~~and A.2~~, the reactor core conditions are consistent with the assumptions in the safety analyses.

incore

DC 3.2-ED

Q3.2-6

A.5

If the QPTR ~~has exceeded~~ remains above the 1.02 limit and a ~~re-~~evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized ~~to eliminate the indicated tilt~~ ~~recalibrated to show a zero QPTR~~ prior to increasing THERMAL POWER to above the limit of Required Actions A.1 ~~and A.2~~. This is done to detect any subsequent significant changes in QPTR.

to restore QPTR to within limit

Q3.2-6

Two notes. Note 1

Required Action A.5 is modified by ~~a Note that~~ states that the ~~indicated tilt is not eliminated~~ QPT is not ~~zeroed out~~ until after the ~~re-~~evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). ~~This Note is~~ intended to prevent any ambiguity about the required sequence of actions.

excore detectors are not normalized to restore QPTR to within limit

Q3.2-6

These Notes are

Insert A

A.6

Once the excore detectors are normalized to ~~eliminate the indicated tilt~~ ~~flux tilt is zeroed out~~ (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the peaking factor verification cannot be performed within 24 hours due to ~~the~~ non-equilibrium core conditions, a maximum time of 48 hours is allowed for the completion of the verification.

restore QPTR to within limit

Q3.2-6

Insert B

achieving equilibrium conditions

DC 3.2-ED

(continued)

Enclosure 5B - page B3.2-26

INSERT A for A.5

Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limit, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors per Required Action A.6.

INSERT B for A.6

Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to support flux mapping.

BASES

ACTIONS

A.6 (continued)

~~core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun. These~~ This Completion Times ~~are~~ is intended to allow adequate time to increase THERMAL POWER to above the limit of Required Actions A.1 ~~and A.2~~ while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Q3.2-6

Required Action A.6 is modified by a Note that states that the peaking factor surveillances ~~may only be done after~~ must be completed when the excore detectors have been normalized to ~~eliminate the indicated tilt calibrated to show zero tilt~~ restore QPTR to within limit (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which ~~can only be accomplished after~~ are only required if the excore detectors ~~are calibrated~~ were normalized to show zero tilt ~~and the core returned to power~~ per Required Action A.5.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to ≤ 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1. ~~if more than one input from Power Range Neutron Flux channels are inoperable.~~

Input from a Power Range Neutron Flux channel is considered to be OPERABLE if the upper and lower detector currents are obtainable. The remaining portion of the channel (the electronics required to provide the channel input to the QPTR alarm) need not be OPERABLE.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is

(continued)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.2

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[ ]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER	JUSTIFICATION
3.2-01	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.2-02	Consistent with the CTS, retain the requirement for performing F_Q after a 20 percent change in power (versus the 10 percent value specific in the ITS).
3.2-03	Consistent with Traveler TSTF-98, Rev. 1, the factor by which the F_Q must be adjusted on increasing F_Q measurements is moved to the COLR. This change is acceptable because the factor is normally contained in the COLR, and it removes detail not required to be contained in TS.
3.2-04	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.2-05	~~In accordance with Traveler TSTF-25, clarify that the excore detectors are normalized to indicate zero tilt. This is only a clarification of NUREG-1431 wording and is acceptable.~~ @3.2-6
3.2-06	Insert Consistent with Traveler TSTF-95, the time allowed for resetting the power range neutron flux - high setpoint if F_Q or $F^N_{\Delta H}$ is outside their limits is extended from 8 hours to 72 hours. As written, the Completion Time of 8 hours to reduce the power range neutron flux-high Trip Setpoints presents an unjustified burden on the operation of the plant. A Completion Time of 72 hours will allow time to perform a second flux map to confirm the results, or determine that the Condition was temporary, without implementing an unnecessary Trip Setpoint change, during which there is increased potential for a plant transient and human error. Following a significant power reduction, at least 24 hours are required to reestablish steady state xenon prior to taking a flux map, and approximately 8 to 12 hours to obtain a flux map, and analyze the data. A significant potential for human error can be created through requiring the Trip Setpoints to be reduced within the same time frame that a unit power reduction is taking place, and within the current 8 hour period. Setpoint adjustment is estimated to take approximately, 4 hours per channel (review of plant Condition supportive of removing channels from service, tripping of bistables, setpoint adjustments, and channel restoration), adding 2 hours for necessary initial preparations (procedure preps, calibration equipment checks, obtaining tools, and approvals); it is reasonable to expect a total of 18 hours. Further, setpoint changes should only be required for extended operation in this Condition. Finally, the Bases for making this setpoint change is the same as the NUREG Bases provided for the 72 hour Completion Time of LCO 3.2.1 Required Action A.4, which is also a setpoint reduction. In summary, this change is acceptable because it would permit time to perform required flux mapping, permit orderly resetting of the high flux Trip Setpoints, and reduce the chances of an inadvertent reactor trip during the required power reduction.
3.2-07	Consistent with Traveler TSTF-97, the Note in SR 3.2.1.2 is revised by removing the phrase "is within limits and" to clarify that the actions to be taken if $F_Q^C(Z)$ is increasing are required regardless of whether $F_Q^C(Z)$ is within its limits.

INSERT for 3.2-05

3.2-05 Consistent with TSTF-241, ISTS 3.2.4, Quadrant Power Tilt Ratio, is revised to provide more appropriate Actions. Required Action A.2 contains a redundant action to reduce THERMAL POWER. This redundant action is deleted and the THERMAL POWER limit of Required Action A.1 is revised to provide the appropriate allowance for subsequent power reductions based on subsequent determination of QPTR. [ ]

The Completion Time of Required Action A.3 requires the peaking factors to be verified within 24 hours of achieving equilibrium conditions with THERMAL POWER reduced by Required Action A.1. In the current Required Action, a significant fraction of the 24 hours could be spent waiting for the plant to stabilize at the new power level leaving insufficient time to measure and analyze the peaking factors or resulting in the peaking factors being measured when the plant is not stable yielding inaccurate information. Since the peaking factors are of prime importance, the proposed change will allow sufficient time to obtain an accurate measurement. [ ]

Required Action [A.5] is revised to add a new Note stating "Required Action [A.6] shall be completed if Required Action [A.5] is performed." As discussed in Section 1.3 of the ITS, an Actions Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not with the LCO Applicability. Therefore, when Required Action [A.5] is completed, QPTR should be back within limit and the LCO may be exited. Adding this Note ensures that the peaking factors are verified after normalization of the excore detectors. Additionally, Required Action [A.5] is revised to state "Normalize excore detectors to restore QPTR to within limit." Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. Thus, the absence of a tilt will manifest itself as QPTR=1.00 rather than zero since quadrant power tilt is expressed as a ratio. Also, from a literal compliance standpoint, the tilt cannot be restored to exactly 1.00. [ ]

3.2-08 Consistent with Traveler TSTF-99, the LCO 3.2.1 (F_Q Methodology), Required ACTION B.1. Completion Time for the reduction of the AFD limits if $F^W_Q(Z)$ is not within limits is increased from 2 hours to 4 hours. This makes it consistent with the Completion Time associated with Required ACTION A.2. of LCO 3.2.1 (F_{XY} methodology). The change is acceptable because it eliminates an inconsistency in the ITS.

3.2-09 For consistency with CTS 3.2.4 and ITS 3.3.1, Condition D, the breakpoints for the Applicability of the surveillances in the notes in ITS SR 3.2.4.1 and SR 3.2.4.2 are modified to be applicable at less than or equal to 75 percent RTP, and greater than 75 percent RTP, respectively. ~~This is an~~ administrative change that retains CTS requirements, and is consistent with TSTF-241. Q3.2-6

3.2-10 Consistent with Traveler TSTF-110, this change moves requirements for increased surveillance frequencies in the event of inoperable alarms to licensee controlled documents. This change is acceptable because it removes requirements regarding alarms and alarm responses that are not necessary to be in the TS to protect public health and safety. Q3.2-3

3.2-11 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

3.2-12 ~~Consistent with CTS,~~ The required time for completion of a flux map 3.2.1.2. for determination of the heat flux hot channel factor is changed from 12 hours to 24 hours after achieving equilibrium Conditions. The proposed change affects SR 3.2.1.1 and SR ~~3.2.2.1.~~ Based on plant experience, The proposed time (24 hours) is a reasonable time period for the completion of the surveillance and does not allow for plant operation in an uncertain condition for a protracted time period. This change is consistent with the TS requirements of Specification 3.0.4 (and associated Bases) that allow 24 hours for the completion of a surveillance after prerequisite plant conditions are attained and for which an exception to Specification 4.0.4 was provided. Insert

3.2-13 This change retains the CTS for the performance of peaking factor determinations following plant shutdowns. The CTS, through the exemption to Specification 4.0.4, allows prerequisite plant conditions to be obtained prior to requiring that the surveillance be completed. Insert

3.2-14 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B). Q3.2-4

3.2-15 This change incorporates Traveler TSTF-109. ACTION A.2. would require the QPTR be determined rather than performing a specific surveillance because more than one surveillance can be used to determine QPTR. SR 3.2.4.1 was revised to retain allowance that SR 3.2.4.2 may be performed in lieu of SR 3.2.4.1. ~~The note for SR 3.2.4.2 is changed to require performance if one "or more" QPTR inputs are inoperable with RTP > 75 percent.~~ Q3.2-10 These changes are acceptable because they clarify the ITS regarding frequency and use of incore flux monitoring for QPTR measurement. The changes reflect that incore detectors provide an acceptable QPTR determination during all plant Conditions.

3.2-16 This change would require both transient and static F_Q measurements be determined when performed for Required ACTIONS 3.2.4 A.3 and A.6. The intent of the Required ACTIONS is to verify that $F_Q(Z)$ is within its limit. $F_Q(Z)$ is approximated by $F^C_Q(Z)$ (which is obtained via SR 3.2.1.1) and $F^W_Q(Z)$ (which is obtained via SR 3.2.1.2). Thus, both $F^C_Q(Z)$ and $F^W_Q(Z)$ must be established to verify $F_Q(Z)$. This change is consistent with Traveler WOG-105.

3.2-17 ~~Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).~~ Q3.2-3 Not used

3.2-18 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

3.2-19 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B). The note and frequency for SR 3.2.4.2 are revised consistent with typical presentation formats that provide for a period of time after establishing conditions. Q3.2-10

3.2-20 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B). Q3.2-1

	TECH SPEC CHANGE	APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.2-01	Substitute overpower CPSES N-16 for overpower ΔT reactor trip function.	No	Yes	No	No
3.2-02	Retain CTS requirement for performing F_Q after a 20% change in power (versus the 10% value specific in the ISTS).	Yes	Yes	No, CTS specifies 10%.	No, CTS specifies 10%.
3.2-03	Consistent with TSTF-98, Rev. 1, the factor by which the F_Q must be adjusted on increasing FQ measurements is moved to the COLR.	Yes	Yes	Yes	Yes
3.2-04	SR is applicable to plants using the Westinghouse Constant Axial Offset Control methodology. TU Electric uses the methodology described in RXE-90-006-P-A, described in CTS 6.9.1.6b, which ties the target flux difference surveillance frequency to the frequency at which the $F_Q^W(Z)$ peaking factor is verified.	No	Yes	No	No
3.2-05	Per a proposed revision to WOG-95, this change clarifies that the excore detectors are normalized to indicate zero tilt. ← Insert	Yes (Q 3.2-6)	Yes	Yes	Yes
3.2-06	Consistent with TSTF-95, increases time to reset Hi Flux setpoints from 8 to 72 hours.	Yes	Yes	Yes	Yes
3.2-07	Consistent with TSTF-97, clarifies the ACTIONS for increasing $F^W_Q(Z)$.	Yes	Yes	Yes	Yes
3.2-08	Consistent withTSTF-99, increases the allowance for AFD restrictions from 2 hours to 4 hours.	Yes	Yes	Yes	Yes
3.2-09	The breakpoints for the Applicability of the surveillances in the notes in ITS SR 3.2.4.1 and SR 3.2.4.2 are modified to be applicable at less than or equal to 75% RTP and greater than 75% RTP, respectively.	Yes	Yes	Yes	Yes
3.2-10	Consistent with TSTF-110, moves requirements for increased surveillance frequencies in the event of inoperable alarms to licensee controlled documents.	Yes, to ECG and FSAR.	Yes, to TRM.	Yes, to FSAR.	Yes, to FSAR.

INSERT for 3.2-05

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
3.2-05	Consistent with TSTF-241, ISTS 3.2.4, Quadrant Power Tilt Ratio, is revised to provide more appropriate Actions.	Yes	Yes	Yes	Yes

	TECH SPEC CHANGE	APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.2-11	This change rearranges the notes applicable to LCO 3.2.3. AFD (Constant Axial Offset Control plants only , CPSES specific) from three separate notes affecting different portions of the LCO into a common note block. This change is consistent with TSTF-164.	No	Yes	No	No
3.2-12	~~Consistent with CTS, the required time for completion of a flux map for determination of the heat flux hot channel factor is changed from 12 hours to 24 hours after achieving equilibrium conditions. The proposed change affects SR 3.2.1.1 and SR 3.2.2.1.~~ Insert	Yes	Yes Q3.2-3	~~No, see CN 3.2-17.~~ Yes	~~No, see CN 3.2-17.~~ Yes
3.2-13	This change retains the CTS for the performance of peaking factor determinations following plant shutdowns.	Yes	Yes	Yes	Yes
3.2-14	This change retains the Wolf Creek current licensing basis (License Amendment 61) for Required ACTION 3.2.2 A.2.	No	No	Yes	No
3.2-15	This change partially incorporates the industry traveler TSTF-109. ACTION A.2 now requires the QPTR be determined rather than performing a specific surveillance because more than 1 surveillance can be used to determine QPTR. ~~The note for SR 3.2.4.2 is changed to require performance if one "or more" QPTR inputs are inoperable.~~	Yes Q3.2-10	Yes	Yes	Yes
3.2-16	Requires both transient and static F_Q measurements when performed for Required ACTIONS 3.2.4 A.3 and A.6.	Yes	Yes	Yes	Yes
3.2-17	~~The frequency requirement for performing F_Q measurements has been revised to conform to the CTS which did not specify a Completion Time.~~	~~No, see CN 3.2-12~~ NA	~~No, see CN 3.2-12.~~ NA	~~Yes~~ NA	~~Yes~~ NA Q3.2-3
3.2-18	This change modifies the QPTR requirements in NUREG-1431, Rev. 1, for Wolf Creek, to retain ~~some~~ the CTS requirements ~~and to incorporate revisions to Required ACTIONS proposed by Traveler WOG-95.~~ approved in Amendment No. 116.	No	No	Yes Q3.2-6	No

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.2-10 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

ITS 3.2.4 Quadrant Power Tilt Ratio
CTS 3/4.2.4 Quadrant Power Tilt Ratio (All FLOG Plants)
JFD 3.2-15
ITS SR 3.2.4.2

Comment: JFD 3.2-15 justifies numerous changes to the STS one of which is unacceptable. JFD 3.2-15 is based upon TSTF-109, which has been rejected. The unacceptable STS change is: The modification of the note to SR 3.2.4.2, and in particular the addition of the 12 hour allowance in the Note to SR 3.2.4.2. Provide adequate justification for this change or adopt the STS version of the Note.

FLOG RESPONSE: The latest status report from the TSTF industry database, dated June 16, 1998, indicates that the NRC has approved TSTF-109. The FLOG continues to pursue the changes approved in TSTF-109.

JFD 3.2-15 is revised to delete the sentence: "The note for SR 3.2.4.2 is changed to require performance if one 'or more' QPTR inputs are inoperable." and added: "The note and Frequency for SR 3.2.4.2 are revised consistent with typical presentation formats that provide for a period of time after establishing conditions." NUREG-1431, Revision 1 currently has "or more" in the Note and TSTF-109 did not modify this wording.

ATTACHED PAGES:

Encl 6A	2
Encl 6B	2

3.2-08 Consistent with Traveler TSTF-99, the LCO 3.2.1 (F_Q Methodology), Required ACTION B.1. Completion Time for the reduction of the AFD limits if $F^W_Q(Z)$ is not within limits is increased from 2 hours to 4 hours. This makes it consistent with the Completion Time associated with Required ACTION A.2. of LCO 3.2.1 (F_{XY} methodology). The change is acceptable because it eliminates an inconsistency in the ITS.

3.2-09 For consistency with CTS 3.2.4 and ITS 3.3.1, Condition D, the breakpoints for the Applicability of the surveillances in the notes in ITS SR 3.2.4.1 and SR 3.2.4.2 are modified to be applicable at less than or equal to 75 percent RTP, and greater than 75 percent RTP, respectively. ~~This is an~~ administrative change that retains CTS requirements, and is consistent with TSTF-241. Q3.2-6

3.2-10 Consistent with Traveler TSTF-110, this change moves requirements for increased surveillance frequencies in the event of inoperable alarms to licensee controlled documents. This change is acceptable because it removes requirements regarding alarms and alarm responses that are not necessary to be in the TS to protect public health and safety. Q3.2-3

3.2-11 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

3.2-12 ~~Consistent with CTS,~~ The required time for completion of a flux map 3.2.1.2. for determination of the heat flux hot channel factor is changed from 12 hours to 24 hours after achieving equilibrium Conditions. The proposed change affects SR 3.2.1.1 and SR ~~3.2.2.1~~. Based on plant experience, The proposed time (24 hours) is a reasonable time period for the completion of the surveillance and does not allow for plant operation in an uncertain condition for a protracted time period. This change is consistent with the TS requirements of Specification 3.0.4 (and associated Bases) that allow 24 hours for the completion of a surveillance after prerequisite plant conditions are attained and for which an exception to Specification 4.0.4 was provided.

Insert

3.2-13 This change retains the CTS for the performance of peaking factor determinations following plant shutdowns. The CTS, through the exemption to Specification 4.0.4, allows prerequisite plant conditions to be obtained prior to requiring that the surveillance be completed. Insert

3.2-14 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B). Q3.2-4

3.2-15 This change incorporates Traveler TSTF-109. ACTION A.2. would require the QPTR be determined rather than performing a specific surveillance because more than one surveillance can be used to determine QPTR. SR 3.2.4.1 was revised to retain allowance that SR 3.2.4.2 may be performed in lieu of SR 3.2.4.1. ~~The note for SR 3.2.4.2 is changed to require performance if one "or more" QPTR inputs are inoperable with RTP > 75 percent.~~ Q3.2-10 These changes are acceptable because they clarify the ITS regarding frequency and use of incore flux monitoring for QPTR measurement. The changes reflect that incore detectors provide an acceptable QPTR determination during all plant Conditions.

3.2-16 This change would require both transient and static F_Q measurements be determined when performed for Required ACTIONS 3.2.4 A.3 and A.6. The intent of the Required ACTIONS is to verify that $F_Q(Z)$ is within its limit. $F_Q(Z)$ is approximated by $F^C_Q(Z)$ (which is obtained via SR 3.2.1.1) and $F^W_Q(Z)$ (which is obtained via SR 3.2.1.2). Thus, both $F^C_Q(Z)$ and $F^W_Q(Z)$ must be established to verify $F_Q(Z)$. This change is consistent with Traveler WOG-105.

3.2-17 ~~Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).~~ Q3.2-3

Not used

3.2-18 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

3.2-19 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B)

The note and frequency for SR 3.2.4.2 are revised consistent with typical presentation formats that provide for a period of time after establishing conditions. Q3.2-10

3.2-20 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B). Q 3.2-1

	TECH SPEC CHANGE	APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.2-11	This change rearranges the notes applicable to LCO 3.2.3. AFD (Constant Axial Offset Control plants only, CPSES specific) from three separate notes affecting different portions of the LCO into a common note block. This change is consistent with TSTF-164.	No	Yes	No	No
3.2-12	Consistent with CTS, the required time for completion of a flux map for determination of the heat flux hot channel factor is changed from 12 hours to 24 hours after achieving equilibrium conditions. The proposed change affects SR 3.2.1.1 and SR 3.2.2.1. Insert	Yes	Yes Q3.2-3	~~No, see CN 3.2-17.~~ Yes	~~No, see CN 3.2-17.~~ Yes
3.2-13	This change retains the CTS for the performance of peaking factor determinations following plant shutdowns.	Yes	Yes	Yes	Yes
3.2-14	This change retains the Wolf Creek current licensing basis (License Amendment 61) for Required ACTION 3.2.2 A.2.	No	No	Yes	No
3.2-15	This change partially incorporates the industry traveler TSTF-109. ACTION A.2 now requires the QPTR be determined rather than performing a specific surveillance because more than 1 surveillance can be used to determine QPTR. ~~The note for SR 3.2.4.2 is changed to require performance if one "or more" QPTR inputs are inoperable.~~	Yes Q3.2-10	Yes	Yes	Yes
3.2-16	Requires both transient and static F_Q measurements when performed for Required ACTIONS 3.2.4 A.3 and A.6.	Yes	Yes	Yes	Yes
3.2-17	~~The frequency requirement for performing F_Q measurements has been revised to conform to the CTS which did not specify a Completion Time.~~	~~No, see CN 3.2-12.~~ NA	~~No, see CN 3.2-12.~~ NA	~~Yes~~ NA	~~Yes~~ NA Q3.2-3
3.2-18	This change modifies the QPTR requirements in NUREG-1431, Rev. 1, for Wolf Creek, to retain ~~some~~ the CTS requirements ~~and to incorporate revisions to Required ACTIONS proposed by Traveler WOG-95.~~ approved in Amendment No. 116.	No	No	Yes Q3.2-6	No

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: CA 3.2-001 **APPLICABILITY:** DC, WC, CA

REQUEST:

Revise last sentence of Bases 3.2.1, Action B.1 to read: "Reducing both the positive and negative AFD limits by ...." This change makes it clear that both positive and negative limits must be reduced when $F_Q^W(Z)$ is not within limits.

ATTACHED PAGES:

Encl 5B B3.2-5

BASES

ACTIONS (continued)

A.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which $F^C_Q(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 ~~8~~ hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which $F^C_Q(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.4

Verification that $F^C_Q(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions. Inherent in this action is identification of the cause of the out of limit condition, and the correction of the cause, to the extent necessary to allow safe operation at the higher power level. The allowable power level is determined by extrapolating $F^C_Q(Z)$. SR 3.2.1.1 must be satisfied prior to increasing power above the extrapolated allowable power level or restoration of any reduced Reactor Trip System setpoints.

B.1

both the positive and negative AFD limits

CA 3.2-001

If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F^W_Q(Z)$, exceeds its specified limits, there exists a potential for $F^C_Q(Z)$ to become excessively high if a normal operational transient occurs. Reducing ~~the AFD~~ by ≥ 1% for each 1% by which $F^W_Q(Z)$ exceeds its limit within the allowed Completion Time of 4 ~~2~~ hours, restricts the axial flux

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: CA 3.2-002 **APPLICABILITY:** DC, WC, CA

REQUEST:

Insert in ITS Bases 3.2.1 the expression from CTS SR 4.2.2.2.f for how to calculate the percent by which both F_Q^C and F_Q^W exceed their limits. This provides more readily available information to Operations and Engineering personnel.

ATTACHED PAGES:

Encl 2	3/4 2-8
Encl 5B	B3.2-4

SURVEILLANCE REQUIREMENTS (Continued)

1) $F_Q^{W}(Z)$ shall be increased ~~over that specified in Specification 4.2.2.2.c~~ by ~~an~~ the appropriate factor specified in the COLR, or 02-01-LG

2) $F_Q^{W}(Z)$ shall be measured at least once per 7 EFPD until two successive maps indicate that

$$\underset{\text{over } Z}{\text{maximum}} \frac{F_Q^{CM}(Z)}{K(Z)} \text{ is not increasing.}$$

~~f. With the relationship specified in Specification 4.2.2.2.c above not being satisfied:~~

~~1) Calculate the percent $F_Q(Z)$~~ If $F_Q^{W}(Z)$ exceeds its limit ~~by the following expression:~~

02-01-LG

$$\left\{\left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^{M}(z) \times W(z)}{\frac{F_Q^{RTP}}{P} \times K(z)}\right]\right) - 1\right\} \times 100 \quad \text{for } P \geq 0.5$$

CA 3.2-002

$$\left\{\left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^{M}(z) \times W(z)}{\frac{F_Q^{RTP}}{0.5} \times K(z)}\right]\right) - 1\right\} \times 100 \quad \text{for } P < 0.5$$

2. ~~Either one of t~~The following actions shall be taken: 02-14-M

a) ~~Place the core in an equilibrium condition where the limit in Specification 4.2.2.2.c is satisfied. Power level may then be increased provided the~~ Within 4 hours, reduce AFD limits of Specification 3.2.1 ~~are reduced~~ at least 1% AFD for each percent $F_Q^{W}(Z)$ exceeds its limit, or be in Mode 2 within the next 6 hours. 02-03-M

BASES

LCO (continued)

The expression for $F^W_Q(Z)$ is:

$$F^W_Q(Z) = F^C_Q(Z)\ W(Z)$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR.

Insert → (CA 3.2-002)

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA. with a high level of probability (DC 3.2-001)

This LCO requires operation within the bounds assumed in the safety analyses. ~~Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits.~~ If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ may produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

If the power distribution measurements are performed at a power level less than 100% RTP, then the $F_Q^C(Z)$ and $F_Q^W(Z)$ values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with F_Q^{RTP} assures compliance with the LCO at all power levels.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F^C_Q(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F^C_Q(Z)$ is $F^M_Q(Z)$ multiplied by ~~a~~ factors ~~which~~ account~~ing~~ for manufacturing tolerances and measurement uncertainties. $F^M_Q(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

(continued)

Insert for Enclosure 5B, page B3.2-4:

Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{\left(\begin{matrix}\text{maximum}\\ \text{over z}\end{matrix}\left[\frac{F_Q^C(z)\,x\,W(z)}{\dfrac{F_Q^{RTP}}{P_C}\;x\;K(z)}\right]\right)-1\right\}\;x\;100 \qquad for\;\; P \geq 0.5$$

$$\left\{\left(\begin{matrix}\text{maximum}\\ \text{over z}\end{matrix}\left[\frac{F_Q^C(z)\,x\,W(z)}{\dfrac{F_Q^{RTP}}{0.5}\;x\;K(z)}\right]\right)-1\right\}\;x\;100 \qquad for\;\; P < 0.5$$

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: CP 3.2-001 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

The mark-up of CTS SR 4.2.5.3 (4.2.3.4 for DCPP) was revised as follows. This SR requires a CHANNEL CALIBRATION on the RCS loop flow rate once per 18 months (refueling interval for DCPP). The CTS SR is equivalent to ITS SR 3.3.1.10 and the Reactor Coolant Flow - Low functional unit. DOC 5-12-A was initiated to address that CTS SR 4.2.5.3 (4.2.3.4 for DCPP) is equivalent to ITS SR 3.3.1.10. For CPSES and DCPP, the strikeout is removed consistent with the FLOG markup methodology.

For CPSES, the CTS SR 4.2.5.3 statement "The channels shall be normalized based on the RCS flow rate determination of Surveillance Requirement." is struck through and DOC 5-03-LG applied. DOC 5-03-LG is revised in Enclosures 3A and 3B to indicate the DOC is applicable to CPSES only and that this information is moved to ITS Bases 3.4.1.

ATTACHED PAGES:

Encl 2	3/4.2-17
Encl 3A	6 & 7
Encl 3B	7 & 8

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify that $F^N_{\Delta H}$ is within limits ~~through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits,~~ or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 ~~2~~ hours. 03-07-A 03-01-LG 03-04-LS6

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2 ~~and/or b.~~ above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ ~~the combination of R and indicated RCS total flow rate are~~ is demonstrated, ~~through incore flux mapping and RCS total flow rate comparison,~~ to be within limits ~~the region of acceptable operation shown on Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2~~ prior to exceeding the following THERMAL POWER levels:# 03-05-M 03-07-A 03-01-LG

1. A nominal 50% of RATED THERMAL POWER. 03-06-A

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

(new) With the above required Actions and Completion Times not met, be in Mode 2 within the next 6 hours. 03-03-M

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ ~~The combination of indicated RCS total flow rate and R~~ shall be determined to be within limits ~~the region of acceptable operation of Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2:~~ 02-07-A 03-07-A

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and 05-10-A

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The ~~indicated~~ RCS total flow rate shall be verified to be within the region of acceptable operation of Table 3.2-1 ~~Figure 3.2-3a~~ for Unit 1 and Table 3.2-2 ~~Figure 3.2-3b~~ for Unit 2 at least once per 12 hours ~~when the value of R, obtained per Specification 4.2.3.2, is assumed to exist.~~ 03-08-A 03-07-A CP-3.2-001

Remove Strike-out

~~4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.~~ 05-03-LG 05-12-A

4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per ~~18 months~~.

REFUELING INTERVAL

DC-ALL-005

THERMAL POWER does not have to be reduced to comply with this Action. 03-06-A

CHANGE NUMBER	NSHC	DESCRIPTION
		violated; and (3) the ITS required ACTIONS prior to and subsequent to power ascension provide assurance that POWER OPERATION at or near RTP will be in accordance with the safety analyses, and therefore acceptable.
04-07	A	The statement that Specification 3.0.4 does not apply is no longer needed as revised ACTIONS permit continued operation for an unlimited period of time.
04-08		Not used. (WC 3.2-001)
04-09	A	~~Consistent with NUREG-1431, a note is added to permit 3 OPERABLE excore channels to be used to calculate QPTR when 1 channel is inoperable and power is ≤75 percent.~~ Insert (Q 3.2-6)
04-10	LS14	~~Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).~~
04-11	A	Not Used.
05-01	LG	The designation of how instrument uncertainties are treated (nominal, in the analysis or in the development of the TS limit) is moved to the Bases. The movement of this level of detail out of the specification is consistent with NUREG-1431 and is an example of removing unnecessary details from the TS in accordance with 10 CFR 50.36.
05-02	LS7	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-03	LG	~~Consistent with NUREG-1431, the requirement to perform a CHANNEL CALIBRATION on the RCS flow meters at least once per 18 months and the requirement to normalize the channels are moved to the Bases for the RCS flow low reactor trip function in ITS Section 3.3.1.~~ (CP 3.2-001)
05-04	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-05	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-06	LS8	In accordance with NUREG-1431, if any of the DNB related parameters of pressure, temperature, or RCS flow are found to be outside their limits, the time period required to perform a power reduction would be extended to 6 hours. The DNB related parameters of RCS average temperature, pressurizer pressure, and RCS flow rate are maintained within specified limits in order to ensure consistency with the assumed initial conditions of the accident analyses. The limits placed on the RCS temperature, pressure, and flow ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed. Compliance with the above limits is verified every 12 hours. If a parameter is found to be outside the required limit, 2 hours are allowed in order to restore the parameter to within the limit. If the parameter is not restored to compliance within the required time, the plant must be shut down. The revised Completion Time of 6 hours is acceptable to allow transition to the required plant Conditions in an orderly manner without unnecessarily initiating any undue plant transients and on the small likelihood of a severe event occurring during the extended time period.
05-07	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-08		Not used.
05-09	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-11	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). (WC 3.2-001)

CHANGE NUMBER	NSHC	DESCRIPTION
05-10	A	The CTS requirement to verify RCS flow rate within limits prior to operation above 75 percent RTP after each fuel loading and at least every 31 effective full power days (EFPDs) would be eliminated from the ITS SRs for DNB parameters. This is acceptable based on the requirement to verify $F^N_{\Delta H}$ within limits prior to operation above 75 percent RTP after each fuel loading and every 31 EFPDs. Since the LCO, ACTION and SRs for $F^N_{\Delta H}$ would address any RCS flow rate problems, the RCS flow rate SR can be considered to duplicate the requirements of the $F^N_{\Delta H}$ ITS SR.
05-11	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-12	A	The requirement to perform [24] month CHANNEL CALIBRATIONS of the RCS loop flowrate indicators is part of ITS SR 3.3.1.10 for Reactor Trip System Instrumentation Function 10 (Reactor Coolant Flow - Low). CP-3.2-001

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-01 LG	The designation of how instrument uncertainties are treated (nominal, in the analysis, or in the development of the CTS limit) is moved to the Bases.	Yes	Yes	Yes	Yes
05-02 LS7	The CPSES specific requirement to verify that the total RCS flow is within limits using the plant computer or elbow tap output voltage on a monthly basis is deleted.	No	Yes	No	No
05-03 LG	The requirement ~~to perform a CHANNEL CALIBRATION at least once per 18 months and the requirement~~ to normalize the ~~channels are~~ RCS loop flow rate indicators is moved to the Bases for ~~the SRs for the RCS flow - low reactor trip function in~~ ITS 3.4.1 ~~3.3.1.~~	~~Yes~~ No, not in CTS.	Yes	~~Yes~~ No, not in CTS.	~~Yes~~ No, not in CTS. (CP 3.2-001)
05-04 LG	Consistent with industry Traveler TSTF-105, the explicit requirements that the RCS flow be measured through the use of a precision heat balance measurement and that the instrumentation used in the performance of the calorimetric flow measurement be calibrated within a specified time period of performing the measurement is moved to the Bases.	No, requirement not in CTS.	Yes	Yes	Yes
05-05 LG	The Wolf Creek Required ACTIONS would be modified to move details regarding identification of the cause for low flow rate to the Bases.	No	No	Yes	No
05-06 LS8	The time to reduce power to less than 5% RTP would be revised from within 4 hours to within the next 6 hours.	Yes	Yes	Yes	Yes
05-07 M	This surveillance is modified for Callaway to require that it be performed within 7 days of achieving 95% RTP.	No, see ITS Section 3.4, CN 3.4-51.	No, see CN 5-11-A.	Yes	Yes
05-08	Not Used.	N/A	N/A	N/A	N/A
05-09 LG	The requirements for inspecting and cleaning the feedwater flow venturi would be moved to licensee-controlled documents.	No, requirement not in CTS.	No, requirement not in CTS.	Yes, to USAR, Chapter 16.	Yes, to FSAR, Chapter 16.

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-10 A	The CTS requirement to verify RCS flow rate within limits prior to operation above 75% RTP after each fuel loading and at least every 31 EFPDs would be eliminated from the SRs for departure from nucleate boiling (DNB) parameters.	Yes	No, requirement not in CTS.	No, requirement not in CTS.	Yes
05-11 A	The change is specific to Comanche Peak. An ACTION is added to clarify that the accident analyses support operation below 85% RTP with a reduced flow rate, and this Condition is not affected by the failure of a precision RCS flow measurement to verify that the required flow exists. In addition, the parameters to be verified per the LCO are clarified. Failure of the precision flow measurement when below 85% RTP following a refueling outage does not result in the violation of the LOC; it only prohibits POWER ASCENSION above 85% RTP.	No	Yes	No	No
05-12 A	The requirement to perform [24] month CHANNEL CALIBRATIONS of the RCS loop flowrate indicators is part of ITS SR 3.3.1.10 for Reactor Trip System Instrumentation Function 10 (Reactor Coolant Flow – Low).	Yes	Yes	Yes	Yes CP 3.2-001

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC 3.2-001 **APPLICABILITY:** DC

REQUEST:

Due to approval of LA 121/119 dated February 13, 1998 which addressed the use of best estimate LOCA, the ITS Bases are revised. Specifically, the appropriate sections of the Bases refer to "high level of probability."

ATTACHED PAGES:

Encl 5B B3.2-2, B3.2-4, B3.2-12, and B3.2-23

BASES

BACKGROUND (continued)

the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO's principal effect is to precludes core power distributions that ~~violate~~ could lead to violation of the following fuel design ~~criteria~~ criterion:

~~a.~~ During a large break loss of coolant accident (LOCA), there is a high level of probability that the peak cladding temperature ~~must~~ will not exceed 2200°F (Ref. 1)~~;~~.

DC 3.2-001

~~b. During a the Condition 2 partial loss of forced reactor coolant flow accident, 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 departure from nucleate boiling (DNB) condition;~~

~~c. During an ejected rod accident, the average fuel pellet enthalpy at the hot spot in irradiated energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); 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 (Ref. 3).~~

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the LOCA peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) ~~the NRC Policy Statement~~.

BASES

LCO (continued)

The expression for $F^w_Q(Z)$ is:

$$F^w_Q(Z) = F^c_Q(Z)\ W(Z)$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR.

Insert → CA 3.2-002

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA. with a high level of probability DC 3.2-001

This LCO requires operation within the bounds assumed in the safety analyses. ~~Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits.~~ If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ may produce~~s~~ unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

If the power distribution measurements are performed at a power level less than 100% RTP, then the $F_Q^c(Z)$ and $F_Q^w(Z)$ values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with F_Q^{RTP} assures compliance with the LCO at all power levels.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F^c_Q(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F^c_Q(Z)$ is $F^m_Q(Z)$ multiplied by a factor~~s~~ which account~~ing~~ for manufacturing tolerances and measurement uncertainties. $F^m_Q(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

(continued)

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During there is a high level of probability that a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) ~~must~~ will not exceed 2200°F; DC 3.2-001

c. During an ejected rod accident, the ~~energy deposition to the~~ average fuel pellet enthalpy at the hot spot in irradiated fuel must not exceed 280 cal/gm [Ref. 1]; and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

~~For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance.~~ The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion ~~of [1.3] using the [W3] CHF~~ applicable to a specific DNBR correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB condition.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This ~~functionality in~~ relationship between power and $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.~~7~~ 6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria: DC 3.2-001

a. During a large break loss of coolant accident (LOCA), there is a high level of probability that the peak cladding temperature ~~must~~ will not exceed 2200°F (Ref. 1);

b. During a ~~the~~ Condition II partial loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the ~~energy deposition to the~~ average fuel pellet enthalpy at the hot spot in irradiated fuel must not exceed 280 cal/gm (Ref. 2); 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 (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC 3.2-ED **APPLICABILITY:** DC

REQUEST:

Various editorial changes that do not impact the technical content of the submittal or other FLOG members.

Changes are noted with DC-3.2-ED in the right margin and noted below:

1. Enclosure 5B, page B3.2-25: Condition A.3.....adds "to verify peaking factors and that the incore quadrant power tilt and QPTR are consistent."
2. Enclosure 5B, page B3.2-26: Condition A.4....adds "incore."
3. Enclosure 5B, page B3.2-22, "for information only" Figure B3.2.3B-1 is deleted consistent with mark-up on page B3.2-20.
4. Enclosure 5B, several pages revised for clarification.

ATTACHED PAGES:

Encl 5B B3.2-13, B3.2-14, B3.2-16, B3.2-17, B3.2-18, B3.2-22, B3.2-25, B3.2-26

BASES

APPLICABLE SAFETY ANALYSES (continued)

this variable value of F^N_H in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an limiting initial $F^N_{\Delta H}$ as a function of power level defined by the ~~COLR~~ $F^N_{\Delta H}$ limit equation in the COLR. DC-3.2-ED 3.2.G-1 redline

The LOCA safety analysis ~~indirectly models~~ also uses $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature [Ref. 3].

The fuel is protected in part by compliance with Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.~~7~~ 6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit ~~identifies~~ is representative of the coolant flow channel with the maximum enthalpy rise. This channel has ~~the least heat removal capability and thus~~ the highest probability for a DNB condition.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses. DC 3.2-ED

A power multiplication factor in this equation includes an additional ~~margin~~ allowance for higher radial peaking factors from reduced

(continued)

BASES

LCO (continued)

thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase ~~0.3%~~ by a cycle-dependent factor, $PF_{\Delta H}$, specified in the COLR for ~~every a 1% RTP~~ reduction in THERMAL POWER.

If the power distribution measurements are performed at a power level less than 100% RTP, then the $F^N_{\Delta H}$ values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with $F^N_{\Delta H}$ assures compliance with the LCO at all power levels. DC 3.2-ED

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. ~~Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.~~

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely to be violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time. The restoration of the peaking factor to within its limits by power reduction or control rod movement does not restore compliance with the LCO. Thus, this condition can not be exited until a valid surveillance demonstrates compliance with the LCO.

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 nevertheless requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.

However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding

(continued)

BASES

ACTIONS

A.2 (continued)

is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

redline

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to ~~the $F^N_{\Delta H}$ exceeding its~~ the $F^N_{\Delta H}$ limit is identified, to the extent necessary, and corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP. SR 3.2.2.1 must be satisfied prior to increasing power above the extrapolated allowable power level or restoration of any reduced Reactor Trip System setpoints. When $F^N_{\Delta H}$ is measured at reduced power levels, the allowable power level is determined by evaluating $F^N_{\Delta H}$ for higher power levels.

redline

3.2.6-1

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

SR 3.2.2.1 is modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving MODE 1). The Note allows for power ascensions if the surveillances are not current. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. Equilibrium conditions are achieved when the core is sufficiently stable such that uncertainties associated with the measurement are valid.

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for

DC 3.2-ED

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

DC 3.2-ED

measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle. Performing this Surveillance in MODE 1 prior to exceeding 75% RTP, or at a reduced power level at any other time, and meeting the 100% RTP $F^N_{\Delta H}$ limit, provides assurance that the $F^N_{\Delta H}$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

REFERENCES

1. Regulatory Guide 1.77, Rev. ~~[0]~~ 0, May 1974.
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3~~B~~ AXIAL FLUX DIFFERENCE (AFD) ~~(Relaxed Axial Offset Control (RAOC) Methodology)~~

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

Relaxed Axial Offset Control (RAOC) is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

The AFD is monitored on an automatic basis using the ~~unit~~ plant process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message ~~immediately~~ if the AFD for two or more OPERABLE excore channels is outside its specified limits.

Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, ~~Constant Axial Offset Control (CAOC)~~, is used to control axial power distribution in day to day operation (Ref. 1). ~~CAOC~~ This requires that the AFD be controlled within a ~~narrow tolerance~~ band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers. DC 3.2-ED

~~The CAOC~~ This operating space is typically smaller and lies within the RAOC operating space. Control within ~~the CAOC~~ this operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

(continued)

D3.2-ED

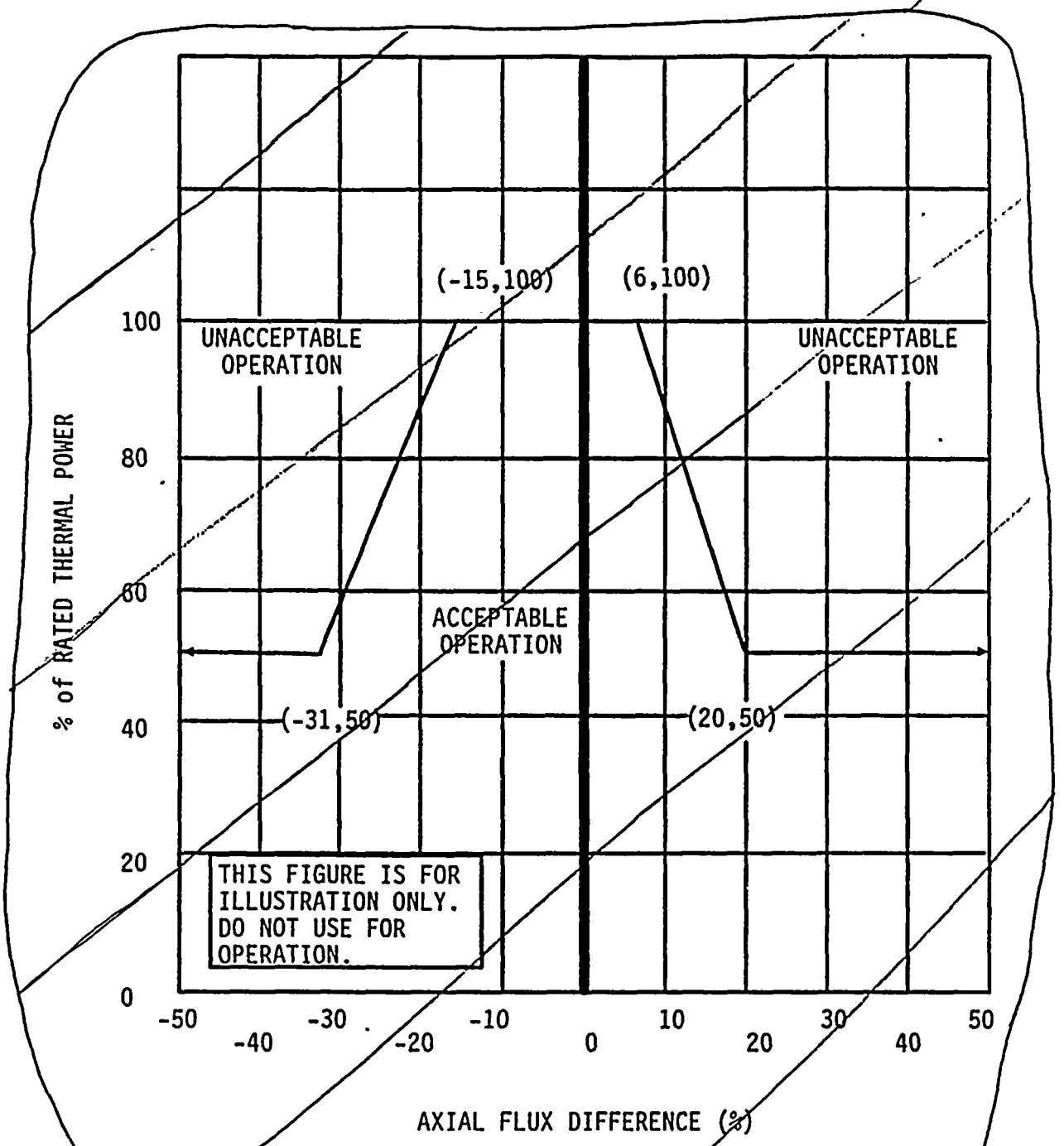


Figure B 3.2.3B-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
as a Function of RATED THERMAL POWER

(continued)

BASES

ACTIONS

A.1 (continued)

time to a identify the cause and correct the tilt. Note that ~~the~~ power reduction (insert) may cause a change in the tilted condition. Q3.2-6

(Insert)

A.2

After completion of Required Action A.1, the QPTR ~~alarm may still be in its alarmed state~~ may still exceed its limits. ~~As such, a~~Any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions to support flux mapping.

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit. Q3.2-6 DC 3.2-E3

to verify peaking factors and that the incore quadrant power tilt and QPTR are consistent

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded

(continued)

BASES

ACTIONS

A.4 (continued)

and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This ~~re-~~evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Actions A.1 ~~and A.2~~, the reactor core conditions are consistent with the assumptions in the safety analyses.

incore

DC 3.2-ED

Q3.2-6

A.5

If the QPTR ~~has exceeded~~ remains above the 1.02 limit and a ~~re-~~evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized ~~to eliminate the indicated tilt~~ recalibrated ~~to show a zero QPTR~~ prior to increasing THERMAL POWER to above the limit of Required Actions A.1 ~~and A.2~~. This is done to detect any subsequent significant changes in QPTR.

to restore QPTR to within limit

Q3.2-6

Required Action A.5 is modified by ~~a Note that~~ two notes. Note 1 states that the ~~indicated tilt is not eliminated~~ ~~QPT is not zeroed out~~ until after the ~~re-~~evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). ~~This Note is~~ These Notes are intended to prevent any ambiguity about the required sequence of actions.

excore detectors are not normalized to restore QPTR to within limit

Q3.2-6

Insert A

A.6

Once the excore detectors are normalized to ~~eliminate the indicated tilt~~ ~~flux tilt is zeroed out~~ (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the peaking factor verification cannot be performed within 24 hours due to ~~the~~ non-equilibrium core conditions, a maximum time of 48 hours is allowed for the completion of the verification.

restore QPTR to within limit

achieving equilibrium conditions

Insert B

Q3.2-6

DC 3.2-ED

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC ALL-005 **APPLICABILITY:** DC

REQUEST:

LAs 123/121 were issued February 27, 1998, and addressed CTS surveillance interval increases due to 24-month fuel cycles.

ATTACHED PAGES:

Encl 2 3/4 2-17

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify that $F^N_{\Delta H}$ is within limits ~~through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits.~~ or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 ~~2~~ hours. 03-07-A 03-01-LG 03-04-LS6

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2 ~~and/or b.~~, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ ~~the combination of R and indicated RCS total flow rate are~~ is demonstrated, ~~through incore flux mapping and RCS total flow rate comparison,~~ to be within limits ~~the region of acceptable operation shown on Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2~~ prior to exceeding the following THERMAL POWER levels:# 03-05-M 03-07-A 03-01-LG

1. A nominal 50% of RATED THERMAL POWER. 03-06-A

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

(new) With the above required Actions and Completion Times not met, be in Mode 2 within the next 6 hours. 03-03-M

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ ~~The combination of indicated RCS total flow rate and R~~ shall be determined to be within limits ~~the region of acceptable operation of Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2:~~ 02-07-A 03-07-A

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and 05-10-A

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The ~~indicated~~ RCS total flow rate shall be verified to be within the region of acceptable operation of Table 3.2-1 ~~Figure 3.2-3a~~ for Unit 1 and Table 3.2-2 ~~Figure 3.2-3b~~ for Unit 2 at least once per 12 hours ~~when the value of R, obtained per Specification 4.2.3.2, is assumed to exist.~~ 03-08-A 03-07-A CP-3.2-001

Remove strike-out

~~4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.~~ 05-03-LG 05-12-A

4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per ~~18 months~~.

REFUELING INTERVAL

DC-ALL-005

THERMAL POWER does not have to be reduced to comply with this Action. 03-06-A

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: TR 3.2-004 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

Revise Traveler Status page to reflect NRC approval and latest revision number of travelers TSTF-110 Revision 2 and TSTF-136. There are no changes involved to any CTS mark-ups, ITS mark-ups, DOCs, or JFDs.

ATTACHED PAGES:

Encl 5A traveler status page

Industry Travelers Applicable to Section 3.2

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-24	Not Incorporated	NA	Not NRC approved as of traveler cut-off date
TSTF-95	Incorporated	3.2-06	Approved by NRC.
TSTF-97	Incorporated	3.2-07	Approved by NRC.
TSTF-98, R1	Incorporated	3.2-03	
TSTF-99	Incorporated	3.2-08	Approved by NRC.
TSTF-109	Incorporated	3.2-15	Approved by NRC.
TSTF-110, ~~R1~~ R2	Incorporated	3.2-10	Approved by NRC. TR 3.2-004
TSTF-112, R1	Not Incorporated	~~NA~~ 3.2-20	~~Not NRC approved as of traveler cut-off date~~
TSTF-136	Incorporated	NA	Approved by NRC. TR 3.2-004
TSTF-164	Incorporated	3.2-11	Applicable to CAOC only. (CPSES)
~~WOG-95, proposed Rev. 2~~ TSTF-241	Incorporated	3.2-05 ~~3.2-18~~ 3.2-09	Q 3.2-6
WOG-105	Incorporated	3.2-16	

Approved by NRC. Applicable to CAOC plants (CPSES ONLY) Q 3.2-1

JLS CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS

CTS 3/4.5 - EMERGENCY CORE COOLING SYSTEMS
ITS 3.5 - EMERGENCY CORE COOLING SYSTEMS

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION AND LICENSEE INITIATED ADDITIONAL CHANGES

INDEX OF ADDITIONAL INFORMATION

ADDITIONAL INFORMATION NUMBER	APPLICABILITY	ENCLOSED
3.5.G-1	DC, CP, WC, CA	YES
3.5.1-1	CP	NA
3.5.1-2	WC	NA
3.5.1-3	CP	NA
3.5.1-4	DC	YES
3.5.1-5	DC	YES
3.5.1-6	DC, CP, WC, CA	YES
3.5.2-1	DC, CP, WC, CA	YES
3.5.2-2	DC, CP, WC, CA	YES
3.5.2-3	DC, CP, WC, CA	YES
3.5.2-4	DC, CP, WC, CA	YES
3.5.2-5	DC, CP, WC, CA	YES
3.5.2-6	DC, CP, WC, CA	YES
3.5.2-7	CP	NA
3.5.2-8	WC, CA	NA
3.5.2-9	DC	YES
3.5.3-1	DC, CP, WC, CA	YES
3.5.3-2	DC, CP, WC, CA	YES
3.5.3-3	DC, CP, WC, CA	YES
3.5.3-4	DC, CP, WC, CA	YES
3.5.3-5	DC, CP, WC, CA	YES
3.5.4-1	DC	YES
3.5.5-1	DC, CP	YES
3.5.5-2	WC, CA	NA
CA 3.5-001	DC, WC, CA	YES
CA 3.5-002	DC, CP, WC, CA	YES
CA 3.5-003	CA	NA
CP 3.5-002	CP	NA
CP 3.5-003	CP	NA
CP 3.5-004	CP	NA
DC 3.5-ED	DC	YES
DC ALL-002 (3.5 changes only)	DC	see DCL-98-003
DC 3.5-001	DC, WC, CA	YES

INDEX OF ADDITIONAL INFORMATION
(cont.)

ADDITIONAL INFORMATION NUMBER	APPLICABILITY	ENCLOSED
DC 3.5-002	DC	YES
DC 3.5-003	DC	YES
DC 3.5-005	DC	YES
DC 3.5-006	DC	YES
TR 3.5-001	DC, CP, WC, CA	YES
WC 3.5-ED	WC	NA
WC 3.5-001	WC	NA
WC 3.5-002	WC	NA
WC 3.5-003	WC	NA

JOINT LICENSING SUBCOMMITTEE METHODOLOGY FOR PROVIDING ADDITIONAL INFORMATION

The following methodology is followed for submitting additional information:

1. Each licensee is submitting a separate response for each section.

2. If an RAI does not apply to a licensee (i.e., does not actually impact the information that defines the technical specification change for that licensee), "NA" has been entered in the index column labeled "ENCLOSED" and no information is provided in the response for that licensee.

3. If a licensee initiated change does not apply, "NA" has been entered in the index column labeled "ENCLOSED" and no information is provided in the response for that licensee.

4. The common portions of the "Additional Information Cover Sheets" are identical, except for brackets, where applicable (using the same methodology used in enclosures 3A, 3B, 4, 6A and 6B of the conversion submittals). The list of attached pages will vary to match the licensee specific conversion submittals. A licensee's FLOG response may not address all applicable plants if there is insufficient similarity in the plant specific responses to justify their inclusion in each submittal. In those cases, the response will be prefaced with a heading such as "PLANT SPECIFIC DISCUSSION."

5. Changes are indicated using the redline/strikeout tool of WordPerfect or by using a hand markup that indicates insertions and deletions. If the area being revised is not clear, the affected portion of the page is circled. The markup techniques vary as necessary, based on the specifics of the area being changed and the complexity of the changes, to provide the clearest possible indication of the changes.

6. A marginal note (the Additional Information Number from the index) is added in the right margin of each page being changed, adjacent to the area being changed, to identify the source of each change.

7. Some changes are not applicable to one licensee but still require changes to the Tables provided in Enclosures 3A, 3B, 4, 6A, and 6B of the original license amendment request to reflect the changes being made by one or more of the other licensees. These changes are not included in the additional information for the licensee to which the change does not apply, as the changes are only for consistency, do not technically affect the request for that licensee, and are being provided in the additional information being provided by the licensees for which the change is applicable. The complete set of changes for the license amendment request will be provided in a licensing amendment request supplement to be provided later.

JOINT LICENSING SUBCOMMITTEE METHODOLOGY FOR PROVIDING ADDITIONAL INFORMATION (cont)

8. The item numbers are formatted as follows:

   [Source] [ITS Section]-[nnn]

   Source =
   - Q - NRC Question
   - CA - AmerenUE
   - DC - PG&E
   - WC - WCNOC
   - CP - TU Electric
   - TR - Traveler

   ITS Section = The ITS section associated with the item (e.g., 3.3). If all sections are potentially impacted by a broad change or set of changes, "ALL" is used for the section number.

   nnn = a three digit sequential number

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.5.G-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST:

ITS 3.5.x Bases

General

There have been a number of instances that the specific changes to the STS Bases are not properly identified with redline or strikeout marks.

Comment: Perform an audit of all STS Bases markups and identify instances where additions and/or deletions of Bases were not properly identified in the original submittal.

FLOG response: Some differences that were identified were in accordance with the markup methodologies (e.g., deletion of brackets and reviewer's notes). Most of the differences were editorial in nature and would not have affected the review. Examples of editorial changes are:

1) Capitalizing a letter with only a "redline" but not striking out the lower case letter that it replaced.
2) Changing a verb from singular to plural by adding an "s" without "redlining" the "s."
3) Deleting instead of striking-out the A, B, C, etc., following a specification title (e.g., SR3.6.6A.7).
4) Changing a bracketed reference (in the reference section) with only a "redline" for the new reference but failing to include the strike-out of the old reference.
5) In some instances, the brackets were retained (and struck-out) but the unchanged text within the brackets was not redlined.
6) Not redlining a title of a bracketed section. The methodology calls for the section title to be redlined when an entire section was bracketed.
7) Additional text not contained in the STS Bases was added to the ITS Bases by the lead FLOG member during the development of the submittal. Once it was determined to not be applicable, the text was then struck-out and remains in the ITS Bases mark-up.

Differences of the above editorial nature will not be provided as attachments to this response. The pages requiring changes that are more than editorial and are not consistent with the markup methodology are attached.

ATTACHED PAGES:

Encl. 5B B3.5-1, B3.5-2, B3.5-3, B3.5-7, B3.5-9, B3.5-10, B3.5-12, B3.5-13, B3.5-15, B3.5-18, B3.5-30, B3.5-33, B3.5-36, B3.5-37

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

the latter phase of blowdown to the beginning phase of reflood

BACKGROUND

The functions of the ECCS accumulators are to supply borated water to replace inventory in the reactor vessel during the ~~blowdown~~ ~~refill phase~~ of a loss of coolant accident (LOCA). ~~to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.~~ DC 3.5-001

The ECCS injection made following a large break LOCA consists of three phases; 1) blowdown, 2) refill, and 3) reflood.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ~~ejection~~ spill out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of ~~safety injection (SI)~~ ECCS water. DC 3.5-001 Insert DC 3.5-ED

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

3.5.G-1

order switched

redline

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by two check valves in series and by an open motor operated isolation valve (8808A, B, C, and D). ~~The motor operated isolation valves are interlocked by P-11 with the pressurizer pressure measurement channels to ensure that the valves will automatically open as RCS pressure increases to above permissive circuit P-11 setpoint. to receive an "open" signal when permissive circuit P-11 is cleared. However, before permissive circuit P-11 is reached, these valves are manually opened and their motor operator breakers are sealed open to satisfy SR 3.5.1.5. Therefore, in the event of a LOCA, accumulator actuation is passive (Ref. [6.1]).~~ CA3.5-001

BASES

BACKGROUND
(continued)

CA 3.5-001

~~This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. However, if these valves were closed, they would be automatically opened as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.~~

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE
SAFETY ANALYSES

1 CA3.5-001

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2 and 4). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

3

3.5.G-1

This paragraph was after next paragraph in NUREG-1431

DC-ALL-002

The limiting large break LOCA is a double ended guillotine break in the RCS piping ~~at the discharge of the reactor coolant pump~~. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow with no credit taken for ECCS pump flow until an effective delay has elapsed. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break. No operator action is assumed during the blowdown stage of a large break LOCA.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.~~

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, ~~the rate of blowdown is such that~~ the SI pumps begin RCS injection, however, the increase in fuel clad temperature is terminated ~~solely~~ primarily by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and the ECCS centrifugal charging and SI pumps ~~both~~ play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease. The accumulators do not discharge above the pressure of their nitrogen cover gas (595.5 to 647.5 psig.) At higher pressures the ECCS centrifugal charging pumps and SI pumps injection becomes solely responsible for terminating the temperature increase.

redline 3.5.G-1

DC3.5-ED

3.5.G-1

~~Until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.~~

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. ~~2~~ 2) CA3.5-001 that are applicable for the accumulators will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium-water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding cladding surrounding the plenum volume, were to react; and

DC3.5-001

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown and reflood phases of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46, though their water volume is credited as part of the long term cooling inventory.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase

SURVEILLANCE REQUIREMENTS (continued)

analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before 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

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as ~~stratification or~~ in-leakage. Sampling the affected accumulator within 6 hours after a solution volume increase ~~of 5.6 1%~~ (101 gallon) narrow range indicated level will identify whether in-leakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), and the RWST has not been diluted since verifying that its boron concentration satisfies SR 3.5.4.3, because the water contained in the RWST is within the accumulator boron concentration requirements as verified by SR 3.5.4.3. This is consistent with the recommendation of GL 93-05 (Ref. ~~8~~ 4).

3.5.G-1

~~of a 1% volume~~

CA3.5-001

SR 3.5.1.5

~~NUREG-1366~~ 3.5.G-1

Verification every 31 days that power is removed from each accumulator isolation valve operator (8808A, B, C, and D) when the ~~pressurizer~~ RCS pressure is ~~≥ 2000~~ greater than 1000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when ~~pressurizer~~ RCS pressure is ~~< 2000~~

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS – Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), non-isolable coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

3.5.G-1

order of these paragraphs and next paragraph on page B3.5-10 (marked with *) is changed from NUREG-1431

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the Refueling Water Storage Tank (RWST) are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS components are divided into two trains, A and B. The following are the train assignments for the ECCS pumps.

Train A:	RHR Pump 2 SI Pump 1 Centrifugal Charging Pump (CCP) 1	Train B:	RHR Pump 1 SI Pump 2 Centrifugal Charging Pump (CCP) 2

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the ~~centrifugal charging pumps~~, CCPs, the RHR pumps, heat exchangers, and the SI pumps. Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the

redline

BASES

ability to utilize components from opposite trains to achieve the required 100% flow to the core.

BACKGROUND (continued) *

3.5.G-1

There are three phases of ECCS operation following a LOCA: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the RWST and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment recirculation sump has enough water to supply the required net positive suction head to the ~~ECCS~~ RHR pumps, suction is switched to the containment recirculation sump for cold leg recirculation. After ~~approximately 24~~ several hours, the ECCS ~~flow~~ operation is shifted to the hot leg recirculation phase to provide ~~a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.~~ reverse flow through the core to backflush out the high boron concentration that could result from core boiling after a cold leg break.

redline

3.5.G-1

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. The RWST header supplies separate piping ~~supplies~~ for each subsystem. ~~and each train within the subsystem.~~ The discharge from the CCPs ~~centrifugal charging pumps~~ combines in a common header and then divides again into four supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the SI and RHR pumps divides and feeds an injection line to each of the RCS cold legs. ~~Control~~ Throttle/runout valves are set to balance the flow to the RCS. The throttle/runout valves also protect the SI and CCPs from exceeding their runout flow limits. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

redline

~~prior to entering the boron injection tank (BIT) (if the plant utilizes a BIT)~~

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the CCPs ~~centrifugal charging pumps~~ supply water until the RCS pressure decreases below the SI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment recirculation sump. The RHR pumps then supply the other ECCS pumps. Initially, recirculation discharge is through the same paths as the injection phase to the cold legs. Subsequently, recirculation ~~alternates~~ provides injection ~~between~~ to both the hot and cold legs.

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions

BASES

e. Adequate long term core cooling capability is maintained.

APPLICABLE SAFETY SAFETY ANALYSES (continued)

The LCO also limits the potential for a post-trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken ~~for~~ credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement ~~to limit~~ runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in the injection phase for mitigation of a small break LOCA event. This event establishes the flow and discharge head for the design point of the CCPs ~~centrifugal charging pumps~~. The SGTR and MSLB events also credit the CCPs ~~centrifugal charging pumps~~. The OPERABILITY requirements for the ECCS are based on the 3.5.6-1 following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite ~~both~~ power and a single failure disabling one RHR pump ~~(all~~ EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large break LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small break LOCA to maintain core subcriticality. For smaller break LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36(c)(2)(ii).

BASES

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. NO PARAGRAPH BREAK

Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal. ~~and initiating semi-automatic switchover of suction having its~~ 3.5.G-1 During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold legs. The ECCS suction is manually transferred to the containment recirculation sump to place the system in the recirculation mode of operation to supply its flow to the RCS hot and cold legs. During the recirculation operation, the RHR pumps provide suction to the charging and SI pumps. redline

~~During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment recirculation sump and to supply its flow to the RCS hot and cold legs.~~

3.5.G-1 redline During recirculation operation, the flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

DC-ALL-002

As indicated in Note 1, the SI flow paths ~~are~~ may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room and a single active failure (Ref. 7) is not assumed coincident with this testing. Therefore the ECCS trains are considered OPERABLE during this isolation.

move to Applicability section

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

BASES

a single OPERABLE ECCS train remains available. (i.e. minimum of one OPERABLE CCP, SI, and RHR pump and applicable flow paths capable of drawing from the RWST and injecting into the RCS cold legs). This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

The intent of this Condition to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available applies to both the injection mode and the recirculation mode.

ACTIONS (continued)

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 6 describes situations in which one component, such as an RHR ~~crossover~~ cross-tie valve can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

Opening the containment recirculation sump access hatch in MODES 1 through 4 is considered to be a condition which is outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered. (Ref. 9)

B.1 and B.2

If the inoperable trains cannot be returned 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 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.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

3.5.4-1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Valve position is the concern and not indicated position in the control room. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power ~~by a control board switch in the correct position~~ ensures that they cannot change position as a result of an active failure or be ~~or by key locking the control in the correct position~~

(continued)

BASES

REQUIREMENTS (continued)

~~Section XI of~~ the ASME Code. (Ref. 8). This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is ~~greater than or equal to~~ within the performance assumed in the plant safety analysis. SRs are specified in ~~Technical Requirements Manual and in~~ the applicable portions of the Inservice Testing Program, which encompasses ~~Section XI~~ Part 6 of the ASME Code for Operation and Maintenance of Nuclear Power Plants. (Ref. 8). ~~Section XI~~ This section of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.

The following ECCS pumps are required to develop the indicated differential pressure when tested on recirculation flow:
CCP ≥ 2400 psid
SI pump ≥ 1455 psid
RHR pump ≥ 165 psid.

DC 3.5-003

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

24 DC-ALL-001 24

SR 3.5.2.7

3.5.6-1

The correct ~~Realignment~~ position of throttle/runout valves in the ECCS flow paths ~~on an SI signal~~ is necessary for proper ECCS performance. These manual throttle/runout valves are positioned during flow balancing and have mechanical locks and seals ~~stops~~ to ~~allow~~ ensure that the proper positioning for restricted flow to a ruptured cold leg ~~ensuring~~ is maintained. The verification of proper position of a throttle/runout valve can be accomplished by confirming the seals ~~and lock~~ have not been altered since the last performance of the flow balance test. Restricting the flow to a ruptured cold leg ensures ~~and~~ that the other cold legs receive at least the required minimum flow. ~~This Surveillance is not required for~~

redline

BASES

APPLICABLE SAFETY ANALYSES (continued)

which increases the differential pressure provided by the downcomer head (this phenomena is sometimes referred to as steam binding). Thus, a higher downcomer mixture level is required to maintain the same reflood rate as before. The additional time required to establish the downcomer head translates into a reduction in the reflood rate in the core. When the downcomer has completely filled, the equilibrium reflood rate for the low containment pressure case would be less than that calculated for a high containment pressure case. This reduction in reflood rate results in a reduction in heat transfer and ultimately an increase in the calculated PCT. Thus, the regulations require that a low containment pressure be calculated in the large-break LOCA analysis.

When calculating containment back pressure for LOCA peak clad temperature

DC 3.5-005

~~In the ECCS~~ analysis, the CS temperature is assumed to be equal to the RWST minimum temperature limit of 35°F. If the minimum temperature limit is violated, the CS ~~containment spray~~ further reduces containment pressure, which decreases the core reflood as explained in the preceding paragraph. For the containment response following a MSLB, the lower limit on boron concentration is used to maximize the total energy release to containment.

redline 3.5.6-1

Insert 1

DC 3.5-005

Steam Line and Feedwater Line Breaks

Volume

RWST volume is not an explicit assumption in other than LOCA events since the required volume for those events is much less than that required by LOCA.

redline 3.5.6-1

Boration

The minimum RWST solution boron concentration is an explicit assumption in the MSLB analysis to ensure the required shutdown capability. Since DCPP no longer uses the boron injection tank, the minimum boron concentration limit is an important assumption in ensuring the required shutdown capability. For the containment response following an MSLB, the lower limit on boron concentration is used to maximize the total energy release to containment.

Feedwater line break results in high temperature/high pressure in the RCS. There is very little RWST water injected due to the high pressure. Also, the analysis results are not affected by the negative reactivity provided by RWST water. Therefore, RWST boron concentration is not a consideration for the feedwater line break.

Insert 2

DC 3.5-005

BASES

ACTIONS (continued)

the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains and that borated water volume can be restored more rapidly than boron concentration or temperature.

C.1 and C.2

If the RWST cannot be returned 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 to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be ~~within the limits~~ above the minimum assumed in the accident analyses ~~band.~~ This Frequency is sufficient to identify a temperature change that would approach ~~either~~ the limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures ~~are within the operating limits of~~ is above the minimum temperature for the RWST. With ambient air temperature ~~within the band,~~ above the minimum temperature, the RWST temperature should not exceed the limit.

3.5.6-1

SR 3.5.4.2

~~injection and to support continued ECCS and containment Spray System pump operation~~

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for ECCS injection and CS pump operation (and to support continued ECCS) on recirculation. Since the RWST volume is normally stable and the contained volume required is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

redline

~~The required RWST water volume of 95% (428,237 gallons, 91% plus 4%measurement uncertainty) is surveilled by the control board indication.~~

BASES

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the ~~injection points~~ cold legs (Ref. ~~2~~ 1). This is accomplished by limiting the line resistance in the RCP seal injection lines to a value consistent with the assumptions in the accident analysis.

The 40 gpm identified in the LCO is not ~~strictly a~~ an absolute flow limit, but rather a flow limit through the RCP seal injection line that is assumed in the accident analyses initial conditions when the ECCS systems are aligned in the injection mode following a LOCA. This flow value correlates to a line resistance in the seal injection flow path that is used in the accident analyses ECCS performance. Thus, the line resistance is the parameter which is controlled to ensure that the ECCS alignment is maintained consistent with the accident analysis assumptions. Charging flow control valve, FCV-128 full open is a test condition and is not indicative of normal operation. Consequently, during normal plant operation, it is possible to have the indicated total seal injection flow greater than 40 gpm while still being within the LCO because during normal plant operation, the ECCS system is not in post accident alignment. ~~based on flow line resistance with the CCPs aligned for safety injection.~~

In order to establish the proper flow line resistance, the seal injection flow path differential pressure and flow are measured. The line resistance is then determined with the RCS pressure within normal limits and the CCP flow control valve fully open. ~~a pressure and flow must be known. The flow line resistance is determined by assuming that the RCS pressure is at normal operating pressure and that the CCP discharge pressure is in this LCO. The CCP discharge header pressure remains essentially constant through all the applicable MODES of this LCO.~~ A reduction in RCS pressure, with no concurrent decrease in CCP discharge header pressure, would result in more flow being ~~diverted to~~ discharged through the RCP seal injection line than at normal RCS operating pressure. The RCP seal injection valve settings established at the prescribed ~~CCP discharge header~~ RCS pressure result in a conservative valve position should RCS pressure decrease. The additional modifier of this LCO, the charging flow control valve ~~(charging flow for four loop units and air operated seal injection for three loop units)~~ being full open, is ~~required since~~ consistent with the air operated valve ~~is designed~~ assumed to fail open for the accident condition.

3.5.G-1

~~greater than or equal to the value specified~~

redline

(continued)

BASES

LCO (continued)

With the ~~discharge~~ RCS pressure and control valve position as specified by the LCO, a flow limit is established which assures that the seal injection line resistance is consistent with the analysis assumptions. ~~a limit is established. It is this line limit resistance that is used in the accident analyses.~~ This limit assures that when the RCS depressurizes following a LOCA and the flow to the pump seals increases, the resulting flow to the seals will be less than the limit assumed in the accident analysis. ~~flow~~

redline

~~The limit on seal injection flow, combined with the CCP discharge header pressure limits and an open wide condition of the charging flow control valve, must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses.~~

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in ~~this~~ these MODE 4. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

3.5.6-1

ACTIONS

A.1 and A.2

With the seal injection flow exceeding its limit, the amount of charging flow available for ECCS injection to the RCS may be reduced. Under this Condition, action must be taken to restore the seal injection flow to below its limit. Required Action A.1 ensures that within 4 hours the remaining available ECCS charging flow (without assuming an additional failure) is ≥ 100% of the assumed post-LOCA charging flow. 100% flow capability may be verified by assuring both CCPs are OPERABLE. Required Action A.2 then allows the operator ~~has 4~~ 72 hours from the time the flow is known to be above the limit but still allowing 100% of the assumed post-LOCA ECCS charging flow, to correctly position the manual valves and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limits. This time is ~~conservative~~ consistent with ~~respect to~~ the Completion Times for ECCS in 3.5.2, ACTION A. ~~of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel.~~

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.1-4 **APPLICABILITY:** DC

REQUEST:

DOC 1-04 LS-8
CTS 3.5.1 Action a
ITS 3.5.1 Action A

The referenced DOC describes the change to the CTS but does not provide any justification for making the change other than that it is consistent with the STS.

Comment: The NSHC for this change appears to provide the needed justification. Therefore, please incorporate the information contained in the NSHC into the subject DOC.

FLOG RESPONSE: The information from NSHC LS-8 has been incorporated into DOC 1-04 LS-8. The revised justification is based on the continued OPERABILITY of the remaining accumulators and avoiding an unnecessary plant transient and shutdown. The replacement of the required ACTION of going to HOT SHUTDOWN with a required reduction in pressure is consistent with the Applicability.

ATTACHED PAGES:

Encl. 3A 1

DESCRIPTION OF CHANGES TO CURRENT TS (CTS) SECTION 3/4.5

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[ ]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER	NSHC	DESCRIPTION
01-01	M	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-02	A	~~Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).~~ Not used (P3.5.1-1)
01-03	A	Replaces reference to the "pressurizer pressure" with a reference to the "RCS pressure." ACTIONS A. and B. require reducing pressurizer pressure to less than 1000 psig. However, pressurizer pressure instrumentation does not have the range to read that pressure. Consequently, reactor coolant system (RCS) pressure instrumentation is used. For the purposes of this limiting condition of operation (LCO), the use of RCS pressure is equivalent.
01-04	LS 8	The ACTION statement is restructured in agreement with NUREG-1431. (Insert C) The CTS ACTION time (within 1 hour) for restoration of accumulator OPERABILITY for conditions other than a closed isolation valve has been replaced. (P3.5.1-4, P3.5.1-5) The replacement ACTION requires that an accumulator inoperable due to boron concentration not within limits must be restored within 72 hours. (Insert A) Also, the requirement to go to HOT SHUTDOWN within the following 6 hours is replaced by the ACTION to reduce RCS pressure to less than or equal to 1000 psig within the following 6 hours. These changes are considered to be relaxations. (Insert B)
01-05	LS 9	The ACTION statement is restructured in agreement with NUREG-1431. The CTS ACTION time (immediately) for restoration of accumulator OPERABILITY due to the isolation valve closed is replaced. (P3.5.1-4, P3.5.1-5) (Insert C) The replacement ACTION requires that for reasons other than boron concentration, the accumulator must be restored within 1 hour. (Insert D) Also, the requirement to go to HOT SHUTDOWN within the following 6 hours is replaced by the ACTION to reduce RCS pressure to less than or equal to 1000 psig. These changes are considered to be relaxations. (Insert E)
01-06	A	The words "with three accumulators OPERABLE and" are added to both ACTION statements to make entry into LCO 3.0.3 mandatory with two or more accumulators inoperable. This change is consistent with NUREG-1431 and is considered administrative in nature since it reflects current plant practice, i.e., current ACTION Statements A. and B. are not entered at the same time on different accumulators.

Insert A for 01-04 LS-8

A 1-hour ACTION to initiate plant shutdown compromises the opportunity for verification, diagnosis, and restoration of the condition of the accumulator to within limits. Changes in boron concentration are slow and the ACTION change provides a more reasonable time in which to restore limits. Increasing the AOT from 1 hour to 72 hours could avoid unnecessary plant transients and plant shutdowns if OPERABILITY cannot be restored within 1 hour but could be restored within 72 hours, thus improving plant safety and increasing plant availability.

The boron in the accumulator water contributes to the analysis assumption that the combined ECCS water in the partially recovered core during the early reflood phase of a large break LOCA is sufficient to keep that portion of the core subcritical. Core cooling during the ECCS injection mode is not affected by the accumulator water boron concentration being outside limits because the accumulator water remains available for injection throughout the duration of the AOT.

The condition of 1 accumulator not within the boron concentration limits does not affect the other 3 accumulators, as the accumulators are mutually isolated. One accumulator below the minimum boron concentration will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Consequently, although the accumulator may be inoperable in accordance with TS requirements due to boron concentration outside of limits, it still retains capability to contribute in satisfying its safety function. If 2 or more accumulators are inoperable for any reason, ITS 3.5.1, ACTION D requires immediate entry into TS 3.0.3.

Insert B for 01-04 LS-8

Replacing the ACTION requirement to go to HOT SHUTDOWN within the following 6 hours with the requirement to reduce RCS pressure to less than or equal to 1000 psig within the following 6 hours makes the ACTION consistent with the Applicability. Reducing the RCS pressure to less than or equal to 1000 psig results in the accumulators no longer being required, and TS Applicability can be exited. At pressures less than or equal to 1000 psig, the rate of RCS blowdown is reduced and the ECCS pumps can provide adequate injection to ensure the peak clad temperature remains below the acceptance limit of 10 CFR 50.46.

Insert C for 01-04 LS-8 & 01-05 LS-9

and is consistent with NUREG-1024

Insert D for 01-05 LS-9

Accumulator OPERABILITY is important to satisfy the analyses assumption that the contents of 3 of the 4 accumulators reach the core following a LOCA. However, an immediate ACTION to initiate plant shutdown denies the opportunity for verification, diagnosis, and restoration of the condition of the accumulator to within limits. The additional time proposed will reduce the probability of unnecessary plant transients and plant shutdowns, thus improving plant safety and increasing plant availability. If 2 or more accumulators are inoperable for any reason, ITS 3.5.1 ACTION D requires immediate entry into TS 3.0.3.

Insert E for 01-05 LS-9

Replacing the ACTION requirement to go to HOT SHUTDOWN within the following 6 hours with the requirement to reduce RCS pressure to less than or equal to 1000 psig within the following 6 hours makes the ACTION consistent with the Applicability. Reducing the RCS pressure to less than or equal to 1000 psig results in the accumulators no longer being required for TS Applicability. At pressures less than or equal to 1000 psig, the rate of RCS blowdown is reduced and the ECCS pumps can provide adequate injection to ensure the peak clad temperature remains below the acceptance limit of 10 CFR 50.46.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.1-5 **APPLICABILITY:** DC

REQUEST:

DOC 1-05 LS-9
CTS 3.5.1 Action b
ITS 3.5.1 Action B

The referenced DOC describes the change to the CTS but does not provide any justification for making the change other than that it is consistent with the STS.

Comment: The NSHC for this change appears to provide the needed justification. Therefore, please incorporate the information contained in the NSHC into the subject DOC.

FLOG RESPONSE: The information from NSHC LS-9 has been incorporated into DOC 1-05 LS-9. The revised justification is based on the continued OPERABILITY of the remaining accumulators and avoiding an unnecessary plant transient and shutdown. The replacement of the required ACTION of going to HOT SHUTDOWN with a required reduction in pressure is consistent with the Applicability.

ATTACHED PAGES:

Encl. 3A 1

DESCRIPTION OF CHANGES TO CURRENT TS (CTS) SECTION 3/4.5

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[ ]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER	NSHC	DESCRIPTION
01-01	M	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-02	A	~~Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).~~ Not used (P3.5.1-1)
01-03	A	Replaces reference to the "pressurizer pressure" with a reference to the "RCS pressure." ACTIONS A. and B. require reducing pressurizer pressure to less than 1000 psig. However, pressurizer pressure instrumentation does not have the range to read that pressure. Consequently, reactor coolant system (RCS) pressure instrumentation is used. For the purposes of this limiting condition of operation (LCO), the use of RCS pressure is equivalent.
01-04	LS 8	The ACTION statement is restructured in agreement with NUREG-1431. (Insert C) The CTS ACTION time (within 1 hour) for restoration of accumulator OPERABILITY for conditions other than a closed isolation valve has been replaced. (P3.5.1-4) (P3.5.1-5) The replacement ACTION requires that an accumulator inoperable due to boron concentration not within limits must be restored within 72 hours. (Insert A) Also, the requirement to go to HOT SHUTDOWN within the following 6 hours is replaced by the ACTION to reduce RCS pressure to less than or equal to 1000 psig within the following 6 hours. These changes are considered to be relaxations. (Insert B)
01-05	LS 9	The ACTION statement is restructured in agreement with NUREG-1431. The CTS ACTION time (immediately) for restoration of accumulator OPERABILITY due to the isolation valve closed is replaced. (P3.5.1-4) (P3.5.1-5) (Insert C) The replacement ACTION requires that for reasons other than boron concentration, the accumulator must be restored within 1 hour. (Insert D) Also, the requirement to go to HOT SHUTDOWN within the following 6 hours is replaced by the ACTION to reduce RCS pressure to less than or equal to 1000 psig. (Insert E) These changes are considered to be relaxations.
01-06	A	The words "with three accumulators OPERABLE and" are added to both ACTION statements to make entry into LCO 3.0.3 mandatory with two or more accumulators inoperable. This change is consistent with NUREG-1431 and is considered administrative in nature since it reflects current plant practice, i.e., current ACTION Statements A. and B. are not entered at the same time on different accumulators.

Insert A for 01-04 LS-8

A 1-hour ACTION to initiate plant shutdown compromises the opportunity for verification, diagnosis, and restoration of the condition of the accumulator to within limits. Changes in boron concentration are slow and the ACTION change provides a more reasonable time in which to restore limits. Increasing the AOT from 1 hour to 72 hours could avoid unnecessary plant transients and plant shutdowns if OPERABILITY cannot be restored within 1 hour but could be restored within 72 hours, thus improving plant safety and increasing plant availability.

The boron in the accumulator water contributes to the analysis assumption that the combined ECCS water in the partially recovered core during the early reflood phase of a large break LOCA is sufficient to keep that portion of the core subcritical. Core cooling during the ECCS injection mode is not affected by the accumulator water boron concentration being outside limits because the accumulator water remains available for injection throughout the duration of the AOT.

The condition of 1 accumulator not within the boron concentration limits does not affect the other 3 accumulators, as the accumulators are mutually isolated. One accumulator below the minimum boron concentration will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Consequently, although the accumulator may be inoperable in accordance with TS requirements due to boron concentration outside of limits, it still retains capability to contribute in satisfying its safety function. If 2 or more accumulators are inoperable for any reason, ITS 3.5.1, ACTION D requires immediate entry into TS 3.0.3.

Insert B for 01-04 LS-8

Replacing the ACTION requirement to go to HOT SHUTDOWN within the following 6 hours with the requirement to reduce RCS pressure to less than or equal to 1000 psig within the following 6 hours makes the ACTION consistent with the Applicability. Reducing the RCS pressure to less than or equal to 1000 psig results in the accumulators no longer being required, and TS Applicability can be exited. At pressures less than or equal to 1000 psig, the rate of RCS blowdown is reduced and the ECCS pumps can provide adequate injection to ensure the peak clad temperature remains below the acceptance limit of 10 CFR 50.46.

Insert C for 01-04 LS-8 & 01-05 LS-9

and is consistent with NUREG-1024

Insert D for 01-05 LS-9

Accumulator OPERABILITY is important to satisfy the analyses assumption that the contents of 3 of the 4 accumulators reach the core following a LOCA. However, an immediate ACTION to initiate plant shutdown denies the opportunity for verification, diagnosis, and restoration of the condition of the accumulator to within limits. The additional time proposed will reduce the probability of unnecessary plant transients and plant shutdowns, thus improving plant safety and increasing plant availability. If 2 or more accumulators are inoperable for any reason, ITS 3.5.1 ACTION D requires immediate entry into TS 3.0.3.

Insert E for 01-05 LS-9

Replacing the ACTION requirement to go to HOT SHUTDOWN within the following 6 hours with the requirement to reduce RCS pressure to less than or equal to 1000 psig within the following 6 hours makes the ACTION consistent with the Applicability. Reducing the RCS pressure to less than or equal to 1000 psig results in the accumulators no longer being required for TS Applicability. At pressures less than or equal to 1000 psig, the rate of RCS blowdown is reduced and the ECCS pumps can provide adequate injection to ensure the peak clad temperature remains below the acceptance limit of 10 CFR 50.46.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.1-6 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

DOC 1-07 LG
CTS 4.5.1.1.b (DC, CA, WC)
CTS 4.5.1.b (CP)
ITS SR 3.5.1.4

The referenced DOC describes the change to the CTS but does not provide any justification for making the change other than that it is consistent with the STS.

Comment: Please revise the DOC to include additional justification as to why this detail is not necessary in the ITS.

FLOG RESPONSE: DOC 1-07 LG has been revised to provide additional justification for the proposed change by adding the following information:

"The RWST has its own LCO and SRs to verify OPERABILITY and cross-references to other specifications are generally inconsistent with the ITS format and are not required to impose OPERABILITY on the referenced equipment. The RWST boron concentration is maintained between [2300] ppm and [2500] ppm which is higher than the minimum boron concentration required to be maintained in the accumulators. If there were reason to doubt the RWST boron concentration, ITS 3.5.4 Condition A would be entered with its 8 hour Completion Time. In addition, ITS SR 3.5.4.3 verifies the boron concentration of the RWST every 7 days. Therefore, it is unlikely that the boron concentration being added to the accumulators would be below [2300] ppm. Additionally, plant procedures implementing the SR 3.5.1.4 Bases specify that if the RWST has been diluted since its last boron concentration sample per SR 3.5.4.3, the boron concentration in the accumulators must be verified within 6 hours after adding [101] gallons or more to the accumulators from the RWST. The moving of this detail to the ITS Bases maintains consistency with NUREG-1431 and is not necessary to adequately protect the health and safety of the public. Details for performing surveillance requirements are more appropriately specified in the plant procedures required by ITS 5.4.1 and the ITS Bases. Control of the plant conditions appropriate to perform a surveillance test is an issue for procedures and scheduling and has been previously determined by the NRC to be unnecessary as a TS restriction. As indicated in Generic Letter 91-04, allowing this licensee control is consistent with the vast majority of other surveillance requirements that do not dictate plant conditions for surveillances. Any change to this detail will be made in accordance with the Bases Control Program described in ITS Section 5.5.14."

ATTACHED PAGES:

Encl. 3A 2

CHANGE NUMBER	NSHC	DESCRIPTION
01-07	LG	The Surveillance Requirement (SR) currently requires a 6 hour surveillance if the makeup source is the refueling water storage tank (RWST) and the RWST has not been diluted since verifying its boron concentration per the RWST LCO. The proposed change would move the statement "and the RWST has not been diluted since verifying ..." from the accumulator SR to the ITS SR 3.5.1.4 Bases. This level of detail is not included in the ISTS and is consistent with the kind of information contained in the Bases. Insert Q3.5.1-6
01-08	A	In accordance with NUREG-1431, the LCO accumulator contained solution volume expressed in cubic feet is replaced by percent (%). Also, the surveillance solution volume increase is revised from 1 percent tank volume (101 gallons) to 5.6 percent of narrow range indicated level, equivalent of 1 percent tank volume. This value can be read from control room indicators. This change in terms is administrative and does not result in a change in the measured volume.
02-01	LG	Insert Q3.5.2-1 Consistent with NUREG-1431, the LCO and ACTION a. are revised to replace the word "subsystem" with the word "train." The descriptive information in the LCO is moved to the Bases. Whereas there is no technical change associated with the replacement of the term "subsystem," "train" better describes that all parts of the required system (e.g., piping, instruments, controls, etc.) must be OPERABLE to support the required safety function.
02-02	LS 1	Consistent with NUREG-1431, a Note with respect to RCS pressure isolation valve (PIV) testing is added to the LCO. Plant design requires closure of certain valves in the safety injection (SI) paths to perform PIV testing. Isolation of the injection paths in MODE 3 is currently prohibited as it would constitute entering TS 3.0.3 since both SI trains would be made administratively inoperable. The flow paths are readily restorable from the control room and a single active failure is not likely in the short term (2 hours). The new Note will allow closing these valves for testing without declaring either SI train inoperable. This change is consistent with Industry Traveler TSTF-153.
02-03	LS 2	This change revises ACTION a. to allow for increased flexibility in plant operations under circumstances where components in opposite trains are inoperable, but at least 100 percent of the emergency core cooling system (ECCS) flow equivalent to a single OPERABLE ECCS train is available. Due to the design of the ECCS subsystems, the inoperable condition of one or more components in each train does not necessarily render the ECCS inoperable for performing its safety function. The allowed outage time (AOT) of 72 hours is unchanged; but it is to be contingent on being capable of providing 100 percent of the ECCS flow equivalent to a single OPERABLE ECCS train. This change is consistent with NUREG-1431.

Insert for 1-07 LG

The RWST has its own LCO and SRs to verify OPERABILITY and cross-references to other specifications are generally inconsistent with the ITS format and are not required to impose OPERABILITY on the referenced equipment. The RWST boron concentration is maintained between [2300] ppm and [2500] ppm which is higher than the minimum boron concentration required to be maintained in the accumulators. If there were reason to doubt the RWST boron concentration, ITS 3.5.4 Condition A would be entered with its 8 hour Completion Time. In addition, ITS SR 3.5.4.3 verifies the boron concentration of the RWST every 7 days. Therefore, it is unlikely that the boron concentration being added to the accumulators would be below [2300] ppm. Additionally, plant procedures implementing the SR 3.5.1.4 Bases specify that if the RWST has been diluted since its last boron concentration sample per SR 3.5.4.3, the boron concentration in the accumulators must be verified within 6 hours after adding [101] gallons or more to the accumulators from the RWST. The moving of this detail to the ITS Bases maintains consistency with NUREG-1431 and is not necessary to adequately protect the health and safety of the public. Details for performing surveillance requirements are more appropriately specified in the plant procedures required by ITS 5.4.1 and the ITS Bases. Control of the plant conditions appropriate to perform a surveillance test is an issue for procedures and scheduling and has been previously determined by NRC to be unnecessary as a TS restriction. As indicated in Generic Letter 91-04, allowing this licensee control is consistent with the vast majority of other surveillance requirements that do not dictate plant conditions for surveillances. Any change to this detail will be made in accordance with the Bases Control Program described in ITS Section 5.5.14.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.2-1 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

DOC 2-01 LG
CTS 3.5.2 LCO
ITS 3.5.2 LCO

The referenced DOC describes the change to the CTS but does not provide any justification for making the change other than that it is consistent with the STS.

Comment: Please revise the DOC to include additional justification as to why this detail is not necessary in the ITS.

FLOG RESPONSE: DOC 2-01-LG has been revised to provide additional justification for the proposed change by adding the following information:

"The proposed change is consistent with NUMARC 93-03, "Writer's Guide for the Restructured Technical Specifications," and the philosophy of NUREG-1431 in which the LCO describes as simply as possible the lowest functional capability of the system and relegates the details of what constitutes an OPERABLE system to the Bases. Therefore, the details of what constitutes an OPERABLE subsystem (train) such as required pumps, heat exchangers, and flow paths, are more appropriately discussed in the Bases than in the LCO. These details are not necessary to ensure ECCS OPERABILITY or that the ECCS can perform its intended safety function. Therefore, the proposed change moves to the Bases details that are not necessary to provide operational safety while retaining in the technical specifications the basic requirements for maintaining OPERABILITY."

ATTACHED PAGES:

Encl. 3A 2

CHANGE NUMBER	NSHC	DESCRIPTION
01-07	LG	The Surveillance Requirement (SR) currently requires a 6 hour surveillance if the makeup source is the refueling water storage tank (RWST) and the RWST has not been diluted since verifying its boron concentration per the RWST LCO. The proposed change would move the statement "and the RWST has not been diluted since verifying ..." from the accumulator SR to the ITS SR 3.5.1.4 Bases. This level of detail is not included in the ISTS and is consistent with the kind of information contained in the Bases. Insert Q3.5.1-6
01-08	A	In accordance with NUREG-1431, the LCO accumulator contained solution volume expressed in cubic feet is replaced by percent (%). Also, the surveillance solution volume increase is revised from 1 percent tank volume (101 gallons) to 5.6 percent of narrow range indicated level, equivalent of 1 percent tank volume. This value can be read from control room indicators. This change in terms is administrative and does not result in a change in the measured volume.
02-01	LG	Consistent with NUREG-1431, the LCO and ACTION a. are revised to replace the word "subsystem" with the word "train." The descriptive information in the LCO is moved to the Bases. Insert Q3.5.2-1 Whereas there is no technical change associated with the replacement of the term "subsystem," "train" better describes that all parts of the required system (e.g., piping, instruments, controls, etc.) must be OPERABLE to support the required safety function.
02-02	LS 1	Consistent with NUREG-1431, a Note with respect to RCS pressure isolation valve (PIV) testing is added to the LCO. Plant design requires closure of certain valves in the safety injection (SI) paths to perform PIV testing. Isolation of the injection paths in MODE 3 is currently prohibited as it would constitute entering TS 3.0.3 since both SI trains would be made administratively inoperable. The flow paths are readily restorable from the control room and a single active failure is not likely in the short term (2 hours). The new Note will allow closing these valves for testing without declaring either SI train inoperable. This change is consistent with Industry Traveler TSTF-153.
02-03	LS 2	This change revises ACTION a. to allow for increased flexibility in plant operations under circumstances where components in opposite trains are inoperable, but at least 100 percent of the emergency core cooling system (ECCS) flow equivalent to a single OPERABLE ECCS train is available. Due to the design of the ECCS subsystems, the inoperable condition of one or more components in each train does not necessarily render the ECCS inoperable for performing its safety function. The allowed outage time (AOT) of 72 hours is unchanged; but it is to be contingent on being capable of providing 100 percent of the ECCS flow equivalent to a single OPERABLE ECCS train. This change is consistent with NUREG-1431.

Insert for 2-01-LG

The proposed change is consistent with NUMARC 93-03, "Writer's Guide for the Restructured Technical Specifications" and the philosophy of NUREG-1431 in which the LCO describes as simply as possible the lowest functional capability of the system and relegates the details of what constitutes an OPERABLE system to the Bases. Therefore, the details of what constitutes an OPERABLE subsystem (train) such as required pumps, heat exchangers and flow paths, are more appropriately discussed in the bases than in the LCO. These details are not necessary to ensure ECCS OPERABILITY or that the ECCS can perform its intended safety function. Therefore, the proposed change moves to the Bases details that are not necessary to provide operational safety while retaining in the technical specifications the basic requirements for maintaining OPERABILITY.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.2-2 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

DOC 2-09 LG
CTS 4.5.2.c

The referenced DOC describes the change to the CTS but does not provide any justification for making the change other than that it is consistent with the STS.

Comment: Please revise the DOC to include additional justification as to why this surveillance is not necessary in the ITS.

FLOG RESPONSE: DOC 2-09 LG has been revised to provide additional justification for the proposed change by adding the following information:

"CTS SR 4.5.2.c requires a visual inspection to verify that no loose debris is present in the containment which could be transported to the containment sump and cause restriction to the pump suction during LOCA conditions at the frequency specified. This ensures that during the process of performing maintenance or other work inside containment that debris is appropriately discarded. Existing procedures restrict containment entries and assure accountability of items entering containment such that they are removed at the completion of the containment entry. ITS SR 3.5.2.8 continues to require a visual inspection every 18 months on each of the ECCS train containment sump suction inlets to ensure that the sump suction inlet is not restricted by debris. Therefore, this detail is not required to be in the technical specifications and moving this requirement maintains consistency with NUREG-1431."

ATTACHED PAGES:

Encl. 3A 3

CHANGE NUMBER	NSHC	DESCRIPTION
02-04	TR 2	Consistent with NUREG-1431, the requirement to submit a special report within 90 days of an ECCS actuation and injection event is deleted. This change is acceptable because the requirement to submit a report is sufficiently addressed by the reporting requirements contained in 10 CFR 50.73.
02-05	LS 3	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-06		Not used.
02-07	A	Consistent with NUREG-1431, this change revises the surveillance to make it clear that "listed" valve position is the concern and not indicated position in the control room. The surveillance can be satisfied using indicated position in the control room but may also be satisfied using local observation. This is an administrative change since the surveillance acceptance criteria are not changed.
02-08	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-09	LG	The visual inspection surveillance performed when establishing containment integrity is moved to a licensee controlled document. ~~consistent with NUREG-1431~~. Insert Q3.5.2-2
02-10	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-11	TR 1	Consistent with NUREG-1431, the ECCS pump and valve actuation SR is changed to allow the use of an actual signal, if and when one occurs, to satisfy SRs. The specific signals used to actuate the pumps and valves have been moved to the Bases. Insert Q3.5.2-3
02-12	LG	The ECCS pump performance is revised to be consistent with NUREG-1431. The test method and specific data required to verify pump performance is relocated to the Bases. CTS 4.0.5 no longer exists in the ITS. However, the requirement for an Inservice Testing (IST) Program is moved to Section 5.5.8 of the ITS. The IST Program is referenced directly for the frequency of testing. Insert Q3.5.2-4
02-13	TR 3	The CTS allowance, which permits the ECCS throttle valves to be declared OPERABLE without verifying ECCS throttle valve position for four hours following stroke valve testing or maintenance is deleted from the CTS consistent with NUREG-1431. The ECCS throttle valves are manual valves and plant procedures governing the restoration of equipment after maintenance specify verification of correct throttle position prior to declaring the valves OPERABLE. This requirement is inherent to post-maintenance OPERABILITY requirements and removal from the specifications does not affect TS requirements for testing scope or frequency.

Insert for 2-09 LG

CTS SR 4.5.2.c requires a visual inspection to verify that no loose debris is present in the containment which could be transported to the containment sump and cause restriction to the pump suction during LOCA conditions at the frequency specified. This ensures that during the process of performing maintenance or other work inside containment that debris is appropriately discarded. Existing procedures restrict containment entries and assure accountability of items entering containment such that they are removed at the completion of the containment entry. ITS SR 3.5.2.8 continues to require a visual inspection every 18 months on each of the ECCS train containment sump suction inlets to ensure that the sump suction inlet is not restricted by debris. Therefore, this detail is not required to be in the technical specifications and moving this requirement maintains consistency with NUREG-1431.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.2-3 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

DOC 2-11 TR-1
CTS 4.5.2.e
ITS SR 3.5.2.5 & SR 3.5.2.6

The referenced DOC describes the change to the CTS but does not provide any justification for making the change other than that it is consistent with the STS.

Comment: The NSHC for this change appears to provide the needed justification. Therefore, please incorporate the information contained in the NSHC into the subject DOC.

FLOG RESPONSE: The CTS requires the use of a test signal for initiation of valid tests. The unintentional result was to require the performance of the verification even if an actual signal has already verified proper operation of equipment. TR-1 allows either an actual or test signal. DOC 2-11 TR-1 has been revised to provide additional discussion to allow the use of an actual signal to meet this surveillance requirement.

ATTACHED PAGES:

Encl. 3A 3

CHANGE NUMBER	NSHC	DESCRIPTION
02-04	TR 2	Consistent with NUREG-1431, the requirement to submit a special report within 90 days of an ECCS actuation and injection event is deleted. This change is acceptable because the requirement to submit a report is sufficiently addressed by the reporting requirements contained in 10 CFR 50.73.
02-05	LS 3	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-06		Not used.
02-07	A	Consistent with NUREG-1431, this change revises the surveillance to make it clear that "listed" valve position is the concern and not indicated position in the control room. The surveillance can be satisfied using indicated position in the control room but may also be satisfied using local observation. This is an administrative change since the surveillance acceptance criteria are not changed.
02-08	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-09	LG	The visual inspection surveillance performed when establishing containment integrity is moved to a licensee controlled document. ~~consistent with NUREG-1431~~. Insert Q3.5.2-2
02-10	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-11	TR 1	Consistent with NUREG-1431, the ECCS pump and valve actuation SR is changed to allow the use of an actual signal, if and when one occurs, to satisfy SRs. The specific signals used to actuate the pumps and valves have been moved to the Bases. Insert Q3.5.2-3
02-12	LG	The ECCS pump performance is revised to be consistent with NUREG-1431. The test method and specific data required to verify pump performance is relocated to the Bases. CTS 4.0.5 no longer exists in the ITS. However, the requirement for an Inservice Testing (IST) Program is moved to Section 5.5.8 of the ITS. The IST Program is referenced directly for the frequency of testing. Insert Q3.5.2-4
02-13	TR 3	The CTS allowance, which permits the ECCS throttle valves to be declared OPERABLE without verifying ECCS throttle valve position for four hours following stroke valve testing or maintenance is deleted from the CTS consistent with NUREG-1431. The ECCS throttle valves are manual valves and plant procedures governing the restoration of equipment after maintenance specify verification of correct throttle position prior to declaring the valves OPERABLE. This requirement is inherent to post-maintenance OPERABILITY requirements and removal from the specifications does not affect TS requirements for testing scope or frequency.

Insert for 2-11 TR-1:

In several specifications throughout the CTS, OPERABILITY of certain equipment is demonstrated by ensuring that the equipment performs its safety function upon receipt of a simulated test signal. The intent of a 'simulated' signal was to be able to perform the required testing without the occurrence (or without causing) an actual signal generating event. However, the unintended effect was to require the performance of the surveillance (using a test signal) even if an actual signal had previously verified the operation of the equipment. This change allows credit to be taken for actual events when the required equipment actuates successfully.

While the occurrence of events that cause actuation of accident mitigation equipment is undesirable, the actuation of mitigation equipment on an actual signal is a better demonstration of its OPERABILITY than an actuation using a test signal. Thus, the change does not reduce the reliability of the equipment tested. The change also improves plant safety by reducing the amount of time the equipment is taken out of service for testing, and thereby increasing its availability during an actual event, and by reducing the wear of the equipment caused by unnecessary testing.

CHANGE NUMBER	NSHC	DESCRIPTION
02-04	TR 2	Consistent with NUREG-1431, the requirement to submit a special report within 90 days of an ECCS actuation and injection event is deleted. This change is acceptable because the requirement to submit a report is sufficiently addressed by the reporting requirements contained in 10 CFR 50.73.
02-05	LS 3	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-06		Not used.
02-07	A	Consistent with NUREG-1431, this change revises the surveillance to make it clear that "listed" valve position is the concern and not indicated position in the control room. The surveillance can be satisfied using indicated position in the control room but may also be satisfied using local observation. This is an administrative change since the surveillance acceptance criteria are not changed.
02-08	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-09	LG	The visual inspection surveillance performed when establishing containment integrity is moved to a licensee controlled document. ~~consistent with NUREG-1431~~. Insert Q3.5.2-2
02-10	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-11	TR 1	Consistent with NUREG-1431, the ECCS pump and valve actuation SR is changed to allow the use of an actual signal, if and when one occurs, to satisfy SRs. The specific signals used to actuate the pumps and valves have been moved to the Bases. Insert Q3.5.2-3
02-12	LG	The ECCS pump performance is revised to be consistent with NUREG-1431. The test method and specific data required to verify pump performance is relocated to the Bases. CTS 4.0.5 no longer exists in the ITS. However, the requirement for an Inservice Testing (IST) Program is moved to Section 5.5.8 of the ITS. The IST Program is referenced directly for the frequency of testing. Insert Q3.5.2-4
02-13	TR 3	The CTS allowance, which permits the ECCS throttle valves to be declared OPERABLE without verifying ECCS throttle valve position for four hours following stroke valve testing or maintenance is deleted from the CTS consistent with NUREG-1431. The ECCS throttle valves are manual valves and plant procedures governing the restoration of equipment after maintenance specify verification of correct throttle position prior to declaring the valves OPERABLE. This requirement is inherent to post-maintenance OPERABILITY requirements and removal from the specifications does not affect TS requirements for testing scope or frequency.

Insert for 2-12 LG

ITS SR 3.5.2.4 retains the SR requirement and references the Inservice Testing Program (IST), discussed in ITS 5.5.8, for the surveillance Frequency. The specific SR acceptance criteria for the pumps have been moved to the ITS SR 3.5.2.4 Bases. Although this may make the ECCS pump performance testing more flexible in the future, in regard to licensee control over the numerical values of the acceptance criteria, this testing must continue to conform to the IST program requirements. Revisions to the acceptance criteria will have to meet the requirements of the Bases Control Program discussed in ITS 5.5.14. Details for performing surveillance requirements are more appropriately specified in the plant procedures required by ITS 5.4.1 and the ITS Bases. Control of the acceptance criteria for a surveillance test is an issue for the IST procedures and has been previously determined by NRC to be unnecessary as a TS restriction. As indicated in Generic Letter 91-04, allowing this licensee control is consistent with the vast majority of other surveillance requirements that do not dictate plant conditions for surveillances.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.2-5 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

DOC 2-15 LG
CTS 4.5.2.h

The referenced DOC describes the change to the CTS but does not provide any justification for making the change other than that it is consistent with the STS.

Comment: Please revise the DOC to include additional justification as to why this surveillance is not necessary in the ITS.

FLOG RESPONSE: DOC 2-15 LG has been revised to include additional justification as to why this surveillance is not necessary in the ITS.

ATTACHED PAGES:

Encl. 3A 4

CHANGE NUMBER	NSHC	DESCRIPTION
02-14	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-15	LG	The SR for the flow balance test following ECCS modifications is moved to a licensee controlled document. This requirement is not included in NUREG-1431. Insert Q3.5.2-5
02-16	LG	The specific means by which the ECCS piping is assured to be full of water is moved to the Bases. This level of detail is not included in the ISTS and is consistent with the kind of information contained in the Bases. Insert Q3.5.2-6
02-17	A	This change adds the phrase "that is not locked, sealed, or otherwise secured in position" with regard to which valves require actuation testing. This change is merely a clarification. Valves that are secured in place, are secured in the position required to meet their safety function. The actuation testing ensures that valves can move to the position that meets their safety function. If the valves are secured in the position that meets their safety function, no testing is necessary.
02-18	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-19	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-01	LG	Consistent with NUREG-1431 Insert Q3.5.3-1 the LCO is revised to replace the word "subsystem" with the word "train" and the descriptive information in the LCO is moved to the Bases. Whereas, there is no technical change associated with the replacement of the term "subsystem," "train" better describes that all parts of the required system (e.g., piping, instruments, controls, etc.) must be OPERABLE to support the required safety functions.
03-02	LS 4	Consistent with NUREG-1431, the LTOP limitation on ECCS pumps and related surveillances are moved to ITS 3.4.12. The prescriptive wording related to pump OPERABILITY is changed to wording specifically addressing the pump's capability to inject into the RCS. This change is less restrictive on the configuration of the centrifugal charging pump (CPP) and SI pumps but is acceptable because it is consistent with the cold overpressure analysis requirements and still precludes flow to the RCS.
03-03	LS 5	~~Consistent with NUREG-1431, CTS 3.5.3 ACTION a. terminology is revised and the descriptive information moved to the Bases.~~ The ACTION a. Completion Time for COLD SHUTDOWN due to CCP inoperability is increased by 4 hours, from 20 hours to 24 hours. This time is reasonable based on operating experience to reach MODE 5 in an orderly manner, without challenging plant systems or operators, and is consistent with other shutdown ACTION Completion Times to reach MODE 5 from MODE 4. Insert Q3.5.3-2

Insert for 2-15 LG

Plant procedures governing the restoration of equipment after maintenance specify the requirements for determining the appropriate post-maintenance testing. Any time the Operability of a system or component has been affected by repair, maintenance, or replacement of a component, post-maintenance testing is required to demonstrate Operability of the system or component. As such, the requirement to perform a flow balance test after modifications that alter ECCS subsystem flow characteristics is not required to be in the TS to provide adequate protection of the public health and safety. This requirement has been moved to the [F]SAR (for Callaway, Diablo Canyon, and Wolf Creek) or TRM (Comanche Peak). These licensee controlled documents containing the moved requirements will be maintained using the provisions of 10 CFR 50.59. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.2-6 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

DOC 2-16 LG
CTS 4.5.2.i
ITS SR 3.5.2.3

The referenced DOC describes the change to the CTS but does not provide any justification for making the change other than that it is consistent with the STS.

Comment: Please revise the DOC to include additional justification as to why this detail is not necessary in the ITS.

FLOG RESPONSE: DOC 2-16 LG has been revised to include additional justification as to why this detail is not necessary in the ITS.

ATTACHED PAGES:

Encl. 3A 4

CHANGE NUMBER	NSHC	DESCRIPTION
02-14	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-15	LG	The SR for the flow balance test following ECCS modifications is moved to a licensee controlled document. This requirement is not included in NUREG-1431. Insert Q3.5.2-5
02-16	LG	The specific means by which the ECCS piping is assured to be full of water is moved to the Bases. This level of detail is not included in the ISTS and is consistent with the kind of information contained in the Bases. Insert Q3.5.2-6
02-17	A	This change adds the phrase "that is not locked, sealed, or otherwise secured in position" with regard to which valves require actuation testing. This change is merely a clarification. Valves that are secured in place, are secured in the position required to meet their safety function. The actuation testing ensures that valves can move to the position that meets their safety function. If the valves are secured in the position that meets their safety function, no testing is necessary.
02-18	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-19	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-01	LG	Consistent with NUREG-1431 Insert Q3.5.3-1 the LCO is revised to replace the word "subsystem" with the word "train" and the descriptive information in the LCO is moved to the Bases. Whereas, there is no technical change associated with the replacement of the term "subsystem," "train" better describes that all parts of the required system (e.g., piping, instruments, controls, etc.) must be OPERABLE to support the required safety functions.
03-02	LS 4	Consistent with NUREG-1431, the LTOP limitation on ECCS pumps and related surveillances are moved to ITS 3.4.12. The prescriptive wording related to pump OPERABILITY is changed to wording specifically addressing the pump's capability to inject into the RCS. This change is less restrictive on the configuration of the centrifugal charging pump (CPP) and SI pumps but is acceptable because it is consistent with the cold overpressure analysis requirements and still precludes flow to the RCS.
03-03	LS 5	~~Consistent with NUREG-1431, CTS 3.5.3 ACTION a. terminology is revised and the descriptive information moved to the Bases.~~ The ACTION a. Completion Time for COLD SHUTDOWN due to CCP inoperability is increased by 4 hours, from 20 hours to 24 hours. This time is reasonable based on operating experience to reach MODE 5 in an orderly manner, without challenging plant systems or operators, and is consistent with other shutdown ACTION Completion Times to reach MODE 5 from MODE 4. Insert Q3.5.3-2

Insert for 2-16 LG

The requirements of ITS LCO 3.5.2 and the associated Surveillance Requirements are adequate to ensure the ECCS are maintained OPERABLE. As a result, the methods of performing Surveillances are not necessary to ensure the ECCS can perform their intended safety function and the details are not required to be in the TS to provide adequate protection of the public health and safety. The ITS Bases containing the moved requirements will be maintained using the provisions of 10 CFR 50.59, as required by Bases Control Program described in ITS Section 5.5.14. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.2-9 **APPLICABILITY:** DC

REQUEST:

ITS B3.5.2 Bases SR 3.5.2.3

The Diablo Canyon STS Bases markup appears to be a markup of the Comanche Peak markup of the STS Bases, rather than a direct markup of the STS Bases. See the referenced section for an example.

Comment: Please confirm that the STS Bases are the starting point for the 3.5 Bases markup and *not* the Comanche Peak ITS Bases.

FLOG RESPONSE: As part of the response to the General Comment Q3.5.G-1, PG&E performed a review of the ITS Bases markup compared to NUREG-1431, Revision 1. Any deviations beyond the criteria stated in the response to the General Comment Q3.5.G-1 is identified. In several areas, text was inserted from the Comanche Peak (lead FLOG plant for that ITS section) mark-up of the STS but was not used as indicated with the strike-out feature. If the added text was used, then it was highlighted with the red-line feature (or indicated as such).

ATTACHED PAGES:

See attached pages to General Comment Q3.5.G-1.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.3-1 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

DOC 3-01 LG
CTS LCO 3.5.3
ITS LCO 3.5.3

The referenced DOC describes the change to the CTS but does not provide any justification for making the change other than that it is consistent with the STS.

Comment: Please revise the DOC to include additional justification as to why this detail is not necessary in the ITS.

FLOG RESPONSE: DOC 3-01-LG has been revised to provide additional justification for the proposed change by adding the following information:

"The proposed change is consistent with NUMARC 93-03, "Writer's Guide for the Restructured Technical Specifications," and the philosophy of NUREG-1431 in which the LCO describes as simply as possible the lowest functional capability of the system and relegates the details of what constitutes an OPERABLE system to the Bases. Therefore, the details of what constitutes an OPERABLE subsystem (train) such as required pumps, heat exchangers, and flow paths, are more appropriately discussed in the bases than in the LCO. These details are not necessary to ensure ECCS OPERABILITY or that the ECCS can perform its intended safety function. Therefore, the proposed change moves to the Bases details that are not necessary to provide operational safety while retaining in the technical specifications the basic requirements for maintaining OPERABILITY."

ATTACHED PAGES:

Encl. 3A 4

CHANGE NUMBER	NSHC	DESCRIPTION
02-14	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-15	LG	The SR for the flow balance test following ECCS modifications is moved to a licensee controlled document. This requirement is not included in NUREG-1431. Insert Q3.5.2-5
02-16	LG	The specific means by which the ECCS piping is assured to be full of water is moved to the Bases. This level of detail is not included in the ISTS and is consistent with the kind of information contained in the Bases. Insert Q3.5.2-6
02-17	A	This change adds the phrase "that is not locked, sealed, or otherwise secured in position" with regard to which valves require actuation testing. This change is merely a clarification. Valves that are secured in place, are secured in the position required to meet their safety function. The actuation testing ensures that valves can move to the position that meets their safety function. If the valves are secured in the position that meets their safety function, no testing is necessary.
02-18	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-19	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-01	LG	Consistent with NUREG-1431 Insert Q3.5.3-1 the LCO is revised to replace the word "subsystem" with the word "train" and the descriptive information in the LCO is moved to the Bases. Whereas, there is no technical change associated with the replacement of the term "subsystem," "train" better describes that all parts of the required system (e.g., piping, instruments, controls, etc.) must be OPERABLE to support the required safety functions.
03-02	LS 4	Consistent with NUREG-1431, the LTOP limitation on ECCS pumps and related surveillances are moved to ITS 3.4.12. The prescriptive wording related to pump OPERABILITY is changed to wording specifically addressing the pump's capability to inject into the RCS. This change is less restrictive on the configuration of the centrifugal charging pump (CPP) and SI pumps but is acceptable because it is consistent with the cold overpressure analysis requirements and still precludes flow to the RCS.
03-03	LS 5	~~Consistent with NUREG-1431, CTS 3.5.3 ACTION a. terminology is revised and the descriptive information moved to the Bases.~~ The ACTION a. Completion Time for COLD SHUTDOWN due to CCP inoperability is increased by 4 hours, from 20 hours to 24 hours. This time is reasonable based on operating experience to reach MODE 5 in an orderly manner, without challenging plant systems or operators, and is consistent with other shutdown ACTION Completion Times to reach MODE 5 from MODE 4. Insert Q3.5.3-2

Insert for 3-01-LG

The proposed change is consistent with NUMARC 93-03, "Writer's Guide for the Restructured Technical Specifications" and the philosophy of NUREG-1431 in which the LCO describes as simply as possible the lowest functional capability of the system and relegates the details of what constitutes an OPERABLE system to the Bases. Therefore, the details of what constitutes an OPERABLE subsystem (train) such as required pumps, heat exchangers and flow paths, are more appropriately discussed in the bases than in the LCO. These details are not necessary to ensure ECCS OPERABILITY or that the ECCS can perform its intended safety function. Therefore, the proposed change moves to the Bases details that are not necessary to provide operational safety while retaining in the technical specifications the basic requirements for maintaining OPERABILITY.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.3-2 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

DOC 3-03 LS-5
CTS 3.5.3 Action a
ITS 3.5.3 Actions A & C

DOC 3-03 LS-5 discussed two distinct changes. The first change involves movement of the descriptive information to the Bases. The second change is an increase in the completion time to reach Mode 5 from 20 to 24 hours.

Comment: The first change, movement of the descriptive information to the Bases, should be separated out and justified as an "LG" change, consistent with other similar changes in this section. The increase in the completion time to reach Mode 5 from 20 to 24 hours is correctly justified as an "LS" change and the justification provided in DOC 3-03 LS-5 is acceptable.

FLOG RESPONSE: DOC 3-03 LS-5 has been separated into two DOCs (DOC 3-03-LS-5 and DOC 3-13-LG). DOC 3-03 LS-5 has been revised to address only the increase in completion time. DOC 3-03 LS-5 has been enhanced to include additional justification by noting that due to the stable conditions associated with MODE 4 operation, the probability of a Design Basis Accident is low. New DOC 3-13 LG has been created to address movement of information to the Bases.

ATTACHED PAGES:

Encl. 2	3/4 5-7
Encl. 3A	4 and 6
Encl. 3B	4 and 6

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 ~~As a minimum, one~~ One ECCS ~~subsystem comprised of the following train~~ shall be OPERABLE.*# 03-01-LG 03-06-A

~~a. One OPERABLE centrifugal charging pump.*~~ 03-02-LS4

~~b. One OPERABLE Residual Heat Removal heat exchanger.~~

~~c. One OPERABLE Residual Heat Removal pump, and~~

~~d. An OPERABLE flow path capable of taking suction from the Refueling Water Storage Tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.~~

APPLICABILITY: MODE 4.

ACTION:

a. With ~~no~~ the required ECCS centrifugal charging pump (CCP) subsystem ~~OPERABLE~~ inoperable ~~because of the inoperability of either the centrifugal charging pump or the flow path from the Refueling Water Storage Tank.~~ restore ~~at least one~~ required ECCS CCP subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next ~~20~~ 24 hours. 03-03-LS5 03-13-LG @3.5.3-2

b. With ~~no~~ the required ECCS Residual Heat Removal (RHR) subsystem ~~OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump.~~ inoperable, restore ~~at least one~~ required ECCS RHR subsystem to OPERABLE status immediately. ~~or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.~~ 03-04-LG 03-14-A @3.5.3-3

~~c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~ 03-05-TR2

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*A maximum of one centrifugal charging pump shall be ~~OPERABLE~~ capable of injecting into the RCS whenever the temperature of one or more of the RCS cold legs is ~~less than or equal to 270°F.~~ below the temperature where LTOP is required. 03-02-LS4 03-11-LG

CHANGE NUMBER	NSHC	DESCRIPTION
02-14	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-15	LG	The SR for the flow balance test following ECCS modifications is moved to a licensee controlled document. This requirement is not included in NUREG-1431. Insert Q3.5.2-5
02-16	LG	The specific means by which the ECCS piping is assured to be full of water is moved to the Bases. This level of detail is not included in the ISTS and is consistent with the kind of information contained in the Bases. Insert Q3.5.2-6
02-17	A	This change adds the phrase "that is not locked, sealed, or otherwise secured in position" with regard to which valves require actuation testing. This change is merely a clarification. Valves that are secured in place, are secured in the position required to meet their safety function. The actuation testing ensures that valves can move to the position that meets their safety function. If the valves are secured in the position that meets their safety function, no testing is necessary.
02-18	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-19	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-01	LG	Consistent with NUREG-1431 Insert Q3.5.3-1 the LCO is revised to replace the word "subsystem" with the word "train" and the descriptive information in the LCO is moved to the Bases. Whereas, there is no technical change associated with the replacement of the term "subsystem," "train" better describes that all parts of the required system (e.g., piping, instruments, controls, etc.) must be OPERABLE to support the required safety functions.
03-02	LS 4	Consistent with NUREG-1431, the LTOP limitation on ECCS pumps and related surveillances are moved to ITS 3.4.12. The prescriptive wording related to pump OPERABILITY is changed to wording specifically addressing the pump's capability to inject into the RCS. This change is less restrictive on the configuration of the centrifugal charging pump (CPP) and SI pumps but is acceptable because it is consistent with the cold overpressure analysis requirements and still precludes flow to the RCS.
03-03	LS 5	~~Consistent with NUREG-1431, CTS 3.5.3 ACTION a. terminology is revised and the descriptive information moved to the Bases.~~ The ACTION a. Completion Time for COLD SHUTDOWN due to CCP inoperability is increased by 4 hours, from 20 hours to 24 hours. This time is reasonable based on operating experience to reach MODE 5 in an orderly manner, without challenging plant systems or operators, and is consistent with other shutdown ACTION Completion Times to reach MODE 5 from MODE 4. Insert Q3.5.3-2

Insert for 3-03 LS-5

Due to stable conditions associated with operation in MODE 4, the probability of occurrence of a Design Basis Accident is low. As a result, the ECCS operational requirements are reduced with only one train of the ECCS CCP Subsystem required to be OPERABLE. The required action if the CCP Subsystem is inoperable is to proceed to cold shutdown.

the PTLR.

CHANGE NUMBER	NSHC	DESCRIPTION
03-12	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-13	LG	Insert Q3.5.3-2
03-14	A	Insert Q3.5.3-3
04-01	LS 4	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-03	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-04	A	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-05	M	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-06	A	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-01	LS 7	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-02	LS 10	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-03	A	This change converts the RWST volume requirement from gallons to the equivalent percent of tank water level, in accordance with NUREG-1431. This change in method of indicating tank volume is administrative and does not result in a change in volume of the tank contents.
05-04	LS 12	This change modifies 3.5.5 ACTION a. to include the requirement for RWST borated water temperature to be above the minimum required temperature and ACTION b. to reference ACTION a. The ACTION for water temperature was in ACTION b with a Completion Time of 1 hour. With the water temperature included in ACTION a. the Completion Time is 8 hours. This change is consistent with NUREG-1431 and NUREG-1024. This change from 1 to 8 hours is a relaxation. Insert Q3.5.4-1
05-05	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). CP3.5-002

Enclosure 3A, page 6

Insert for 3-13 LG

03-13	LG	CTS LCO 3.5.3 Action a. terminology is revised and the descriptive information is moved to the ITS Bases. These details are not necessary to ensure the ECCS Required Actions are met. The requirements of ITS 3.5.3 LCO and Conditons are adequate for ensuring the ECCS are OPERABLE. These details are not necessary to ensure the ECCS can perform their intended safety function. As such, these details are not required to be in the TS to provide adequate protection of the public health and safety. Moving these details maintains consistency with NUREG-1431. Any change to these details will be made in accordance with 10CFR50.59 and the Bases Control Program described in ITS Section 5.5.14.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.5

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-17 A	This change adds the phrase "that is not locked, sealed, or otherwise secured in position" with regard to which valves require actuation testing.	Yes	Yes	Yes	Yes
02-18 LG	The CPSES requirement for venting the ECCS pump casing and piping following maintenance or activity which drains portions of the system is moved out of the TS.	No, not in CTS.	Yes	No, not in CTS.	No, not in CTS.
02-19 LG	This change moves the requirement that the 18 month verification of automatic ECCS valve actuation and ECCS pump actuation be performed during shutdown to the Bases.	No, DCPP does not have this restriction.	No, CPSES does not have this restriction.	Yes	Yes
03-01 LG	The LCO is revised from subsystem to train and the descriptive information moved to the Bases.	Yes	Yes	Yes	Yes
03-02 LS 4	The LTOP limitation on ECCS pumps and related surveillances are moved to Section 3.4.12 in the ITS. The prescriptive wording related to pump OPERABILITY is changed to wording specifically addressing the pump's capability to inject into the RCS.	Yes	Yes	Yes	Yes
03-03 LS 5	~~LCO 3.5.3 ACTION a. descriptive information is moved to the Bases.~~ The Completion Time for COLD SHUTDOWN due to CCP inoperability is increased from 20 to 24 hours.	Yes Q3.5.3-2	Yes	Yes	Yes
03-04 LG	The LCO ACTION terminology is revised and the descriptive information moved to the Bases.	Yes	Yes	Yes	Yes

Insert Q3.5.3-3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-11 LG	The minimum RCS temperature limit for DCPP below which the CCPs and the SI pumps must be demonstrated not capable of injecting into the RCS is replaced by the statement "below the temperature where LTOP is required as specified in the PTLR."	Yes	No	No	No
03-12 A	The SR to verify that no more than one CCP and no SI pumps are capable of injecting into the RCS and the SR exception for 4 hours after entering MODE 4 from MODE 3 or until the temperature of one or more RCS cold legs decreases below 325°F, whichever comes first are moved to ITS SR 3.4.12.1, SR 3.4.12.2, and LCO 3.4.12, Note 2.	No, not in CTS.	No, see Change No. 04-06-A.	Yes	Yes
04-01 LS 4	The requirement for having ECCS pump injection sources in excess of that allowed by cold overpressure analysis assumptions be rendered inoperable, has been revised to preclude those pumps from injecting into the RCS.	No, DCPP does not have this TS.	Yes	Yes	Yes
04-02 M	The ACTION required if ECCS pumps in violation of the cold overpressure analyses are capable of injecting into the RCS has been changed to require immediate ACTION initiation. Otherwise, if precluded from compliance, to depressurize the RCS and establish the necessary vent path within 8 hours.	No, DCPP does not have this TS.	Yes	Yes	Yes
04-03 M	This change requires the verification that the disallowed ECCS pumps are not capable of injecting to the RCS on a 12 hour frequency. Previously a 31 day verification on breaker position was required.	No, DCPP does not have this TS.	Yes	Yes	Yes
03-13 LG	Insert Q3.5.3-2				
03-14 A	Insert Q3.5.3-3				

Enclosure3B INSERT

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
3-13 LG	The CTS LCO 3.5.3 Action a. terminology is revised and descriptive information moved to the ITS Bases. These details are not necessary to ensure the ECCS Required Actions are met.	Yes	Yes	Yes	Yes
3-14 A	CTS LCO 3.5.3 Action b. provides, with no ECCS RHR subsystem OPERABLE, the option to either restore at least one ECCS RHR subsystem to OPERABLE status or to maintain the RCS T_{avg} < 350°F by use of alternate heat removal methods. The CTS Action is revised to immediately initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status to ensure that prompt action is taken to restore the required cooling capacity.	Yes	Yes	Yes	Yes

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.3-3 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

DOC 3-04 LG
CTS 3.5.3 Action b
ITS 3.5.3 Action B

DOC 3-04 LG discussed two distinct changes. The first change involves a change in the wording of the Action requirement. The second change is movement of the instructions to maintain temperature using alternate heat removal methods to the Bases.

Comment: The first change to the wording of the Action requirement should be separated out and justified as an "A" change. The movement of the instructions to the Bases is correctly justified as an "LG" change, but the justification provided in DOC 3-03 LS-5 is not adequate.

Please revise the DOC to include additional justification as to why this detail is not necessary in the ITS.

FLOG RESPONSE: DOC 3-04 LG has been separated into two DOCs (DOC 3-04 LG and DOC 3-14-A). DOC 3-04 LG now contains only the information related to the movement of information to the Bases. DOC 3-04 LG has been enhanced to provide the requested additional justification. New DOC 3-14-A was created to address the change to the wording in the Action requirement.

ATTACHED PAGES:

Encl. 2	3/4 5-7
Encl. 3A	5 and 6
Encl. 3B	4 and 6

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 ~~As a minimum, one~~ One ECCS ~~subsystem comprised of the following train~~ shall be OPERABLE.*# 03-01-LG 03-06-A

~~a. One OPERABLE centrifugal charging pump.*~~ 03-02-LS4

~~b. One OPERABLE Residual Heat Removal heat exchanger.~~

~~c. One OPERABLE Residual Heat Removal pump, and~~

~~d. An OPERABLE flow path capable of taking suction from the Refueling Water Storage Tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.~~

APPLICABILITY: MODE 4.

ACTION:

a. With ~~no~~ the required ECCS centrifugal charging pump (CCP) subsystem ~~OPERABLE~~ inoperable ~~because of the inoperability of either the centrifugal charging pump or the flow path from the Refueling Water Storage Tank,~~ restore ~~at least one~~ required ECCS CCP subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next ~~20~~ 24 hours. 03-03-LS5 03-13-LG @3.5.3-2

b. With ~~no~~ the required ECCS Residual Heat Removal (RHR) subsystem ~~OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump,~~ inoperable, restore ~~at least one~~ required ECCS RHR subsystem to OPERABLE status immediately. ~~or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.~~ 03-04-LG 03-14-A @3.5.3-3

~~c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~ 03-05-TR2

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*A maximum of one centrifugal charging pump shall be ~~OPERABLE~~ capable of injecting into the RCS whenever the temperature of one or more of the RCS cold legs is ~~less than or equal to 270°F.~~ below the temperature where LTOP is required. 03-02-LS4 03-11-LG

CHANGE NUMBER	NSHC	DESCRIPTION
03-04	LG	Consistent with NUREG-1431, the ACTION b terminology is revised. The requirement to restore at least one ECCS subsystem is revised to "immediately initiate action to restore" a residual heat removal (RHR) subsystem. With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, when the only available heat removal system is the RHR. Therefore, the appropriate ACTION is to initiate measures to restore one ECCS RHR subsystem and to continue the ACTIONS until the subsystem is restored to OPERABLE status. Also, the alternate requirement (if RHR cannot be restored) to maintain T_{avg} <350°F by use of alternate heat removal methods is descriptive information and is moved to the Bases. The transition to MODE 3 is already prohibited in this scenario by the ECCS specification for MODES 1, 2, and 3. Insert Q3.5.3-3
03-05	TR 2	Consistent with NUREG-1431, the requirement to submit a special report within 90 days of an ECCS actuation and injection event is deleted. This change is acceptable because the requirement to submit a report is sufficiently addressed by the reporting requirements contained in 10 CFR 50.73.
03-06	A	Consistent with TSTF-90, a Note is added to the LCO that clarifies an RHR train's ECCS function is operable if it is capable of being manually realigned to the ECCS mode of operation. This is an administrative change to provide clarification.
03-07	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-08	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-09	M	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-10	LS 6 DC-ALL-001	Consistent with NUREG-1431, the requirement to demonstrate ECCS train OPERABILITY in MODE 4 in SR [4.5.3.1] has been revised to delete the 31 day surveillance to verify the correct position of each valve in the ECCS flow path which is not already locked in place, and the 18 [24] month surveillance to verify automatic actuation of ECCS pumps and automatic valves. Insert Q3.5.3-5
03-11	LG	The minimum RCS temperature limit below which the CCP and SI pumps must be demonstrated not capable of injecting into the RCS is replaced by the statement "below the temperature where LTOP is required as specified in the pressure temperature limits report (PTLR)." The minimum temperature is a plant specific requirement based on the reactor vessel material characteristics documented in the PTLR and is periodically reviewed and adjusted as required. Referring to the PTLR for the current value is consistent with the relocation of the pressure temperature limits from ITS Section 3.4.3 to

Insert for 3-04 LG

CTS 3.5.3 Action b provides an alternate requirement (if RHR cannot be restored) to maintain $T_{avg} < 350°F$ by use of alternate heat removal methods. These details are moved to the ITS Bases. These details are not necessary to ensure the ECCS are OPERABLE. The requirements of ITS 3.5.3 LCO and Conditions are adequate for ensuring that the ECCS are OPERABLE. These details are not necessary to ensure the ECCS can perform their intended safety function. As such, these details are not required to be in the TS to provide adequate protection of the public health and safety. Moving these details maintains consistency with NUREG-1431. Any change to these details will be made in accordance with 10CFR50.59 and the Bases Control Program described in ITS Section 5.5.14.

the PTLR.

CHANGE NUMBER	NSHC	DESCRIPTION
03-12	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-13	LG	Insert Q3.5.3-2
03-14	A	Insert Q3.5.3-3
04-01	LS 4	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-03	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-04	A	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-05	M	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-06	A	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-01	LS 7	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-02	LS 10	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-03	A	This change converts the RWST volume requirement from gallons to the equivalent percent of tank water level, in accordance with NUREG-1431. This change in method of indicating tank volume is administrative and does not result in a change in volume of the tank contents.
05-04	LS 12	This change modifies 3.5.5 ACTION a. to include the requirement for RWST borated water temperature to be above the minimum required temperature and ACTION b. to reference ACTION a. The ACTION for water temperature was in ACTION b with a Completion Time of 1 hour. With the water temperature included in ACTION a. the Completion Time is 8 hours. This change is consistent with NUREG-1431 and NUREG-1024. This change from 1 to 8 hours is a relaxation. Insert Q3.5.4-1
05-05	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). CP3.5-002

Insert for 3-14 A

03-14 A

CTS LCO 3.5.3 Action b. provides, with no ECCS RHR subsystem OPERABLE, the option to either restore at least one ECCS RHR subsystem to OPERABLE status or to maintain the RCS T_{avg} < 350°F by use of alternate heat removal methods. Condition A of ITS LCO 3.5.3 requires that with no ECCS RHR subsystems OPERABLE that immediate action be initiated to restore an ECCS RHR subsystem to OPERABLE status. While the CTS does not specify a time frame to initiate action to restore one ECCS RHR subsystem, the current operational philosophy is that this action is initiated immediately. The Completion Time of "immediately" to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Revising the CTS Action to immediately initiate action is considered an administrative change and is consistent with NUREG-1431.

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-17 A	This change adds the phrase "that is not locked, sealed, or otherwise secured in position" with regard to which valves require actuation testing.	Yes	Yes	Yes	Yes
02-18 LG	The CPSES requirement for venting the ECCS pump casing and piping following maintenance or activity which drains portions of the system is moved out of the TS.	No, not in CTS.	Yes	No, not in CTS.	No, not in CTS.
02-19 LG	This change moves the requirement that the 18 month verification of automatic ECCS valve actuation and ECCS pump actuation be performed during shutdown to the Bases.	No, DCPP does not have this restriction.	No, CPSES does not have this restriction.	Yes	Yes
03-01 LG	The LCO is revised from subsystem to train and the descriptive information moved to the Bases.	Yes	Yes	Yes	Yes
03-02 LS 4	The LTOP limitation on ECCS pumps and related surveillances are moved to Section 3.4.12 in the ITS. The prescriptive wording related to pump OPERABILITY is changed to wording specifically addressing the pump's capability to inject into the RCS.	Yes	Yes	Yes	Yes
03-03 LS 5	~~LCO 3.5.3 ACTION a. descriptive information is moved to the Bases.~~ The Completion Time for COLD SHUTDOWN due to CCP inoperability is increased from 20 to 24 hours.	Yes Q3.5.3-2	Yes	Yes	Yes
03-04 LG	The LCO ACTION terminology is revised and the descriptive information moved to the Bases.	Yes	Yes	Yes	Yes

Insert Q3.5.3-3

Insert for 03-04-LG:

CTS 3.5.3 Action b provides an alternate requirement (if RHR cannot be restored) to maintain $T_{avg} < 350$ F by use of alternate heat removal methods. These details are moved to the ITS Bases.

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-11 LG	The minimum RCS temperature limit for DCPP below which the CCPs and the SI pumps must be demonstrated not capable of injecting into the RCS is replaced by the statement "below the temperature where LTOP is required as specified in the PTLR."	Yes	No	No	No
03-12 A	The SR to verify that no more than one CCP and no SI pumps are capable of injecting into the RCS and the SR exception for 4 hours after entering MODE 4 from MODE 3 or until the temperature of one or more RCS cold legs decreases below 325°F, whichever comes first are moved to ITS SR 3.4.12.1, SR 3.4.12.2, and LCO 3.4.12, Note 2.	No, not in CTS.	No, see Change No. 04-06-A.	Yes	Yes
04-01 LS 4	The requirement for having ECCS pump injection sources in excess of that allowed by cold overpressure analysis assumptions be rendered inoperable, has been revised to preclude those pumps from injecting into the RCS.	No, DCPP does not have this TS.	Yes	Yes	Yes
04-02 M	The ACTION required if ECCS pumps in violation of the cold overpressure analyses are capable of injecting into the RCS has been changed to require immediate ACTION initiation. Otherwise, if precluded from compliance, to depressurize the RCS and establish the necessary vent path within 8 hours.	No, DCPP does not have this TS.	Yes	Yes	Yes
04-03 M	This change requires the verification that the disallowed ECCS pumps are not capable of injecting to the RCS on a 12 hour frequency. Previously a 31 day verification on breaker position was required.	No, DCPP does not have this TS.	Yes	Yes	Yes
03-13 LG	Insert Q3.5.3-2				
03-14 A	Insert Q3.5.3-3				

Enclosure3B INSERT

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
3-13 LG	The CTS LCO 3.5.3 Action a. terminology is revised and descriptive information moved to the ITS Bases. These details are not necessary to ensure the ECCS Required Actions are met.	Yes	Yes	Yes	Yes
3-14 A	CTS LCO 3.5.3 Action b. provides, with no ECCS RHR subsystem OPERABLE, the option to either restore at least one ECCS RHR subsystem to OPERABLE status or to maintain the RCS T_{avg} < 350°F by use of alternate heat removal methods. The CTS Action is revised to immediately initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status to ensure that prompt action is taken to restore the required cooling capacity.	Yes	Yes	Yes	Yes

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.3-4 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

DOC 3-06 A
CTS LCO 3.5.3
ITS LCO 3.5.3 Note

This change is categorized as an administrative change even though it provides an exception to the LCO requirements that does not exist in the CTS. The DOC states that the note is only to "provide clarification."

Comment: Despite licensees' individual interpretations of the CTS, the CTS themselves do not contain the allowance provided in the ITS Note. Therefore, this change should be reclassified as a less restrictive change and an appropriate justification provided.

FLOG RESPONSE: As discussed during a telecon with NRC Staff on June 25, 1998, the FLOG takes exception to this RAI. The NRC accepted the same change at Vogtle as an administrative change, as discussed in Section 3.1.3.5 item (4) of the Vogtle SER wherein it was stated that this Note "is a necessary clarification when using the RHR system for cooling the RCS, when transitioning between MODES 4 and 5. Because this clarification constitutes existing operating practices, this change is administrative and is acceptable." In addition, the wording of CTS LCO 3.5.3.d, which refers to the RWST flow path "being manually realigned," supports the position that the new LCO Note, moved from the SR to the LCO per NRC-approved TSTF-90, Revision 1, is an administrative change.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.3-5 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

DOC 3-10 LS-6
CTS 4.5.3.1.1 (DC), CTS 4.5.3.1 (All others)
ITS 3.5.3

The referenced DOC describes the change to the CTS but does not provide any justification for making the change other than that it is consistent with the STS.

Comment: The NSHC for this change appears to provide the needed justification. Therefore, please incorporate the information contained in the NSHC into the subject DOC.

FLOG RESPONSE: DOC 3-10 LS-6 has been revised to provide additional justification for the proposed change by adding the following information:

"This change is acceptable because the ECCS operational requirements can be reduced due to the stable conditions associated with operation in MODE 4 and the decreased probability of occurrence of a Design Basis Accident (DBA). ECCS operational requirement reductions mean that certain automatic safety injection (SI) actuation signals are not available. However, in MODE 4 sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA."

ATTACHED PAGES:

Encl. 3A 5

CHANGE NUMBER	NSHC	DESCRIPTION
03-04	LG	Consistent with NUREG-1431, the ACTION b terminology is revised. The requirement to restore at least one ECCS subsystem is revised to "immediately initiate action to restore" a residual heat removal (RHR) subsystem. With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, when the only available heat removal system is the RHR. Therefore, the appropriate ACTION is to initiate measures to restore one ECCS RHR subsystem and to continue the ACTIONS until the subsystem is restored to OPERABLE status. Also, the alternate requirement (if RHR cannot be restored) to maintain T_{avg} <350°F by use of alternate heat removal methods is descriptive information and is moved to the Bases. The transition to MODE 3 is already prohibited in this scenario by the ECCS specification for MODES 1, 2, and 3. Insert @3.5.3-3
03-05	TR 2	Consistent with NUREG-1431, the requirement to submit a special report within 90 days of an ECCS actuation and injection event is deleted. This change is acceptable because the requirement to submit a report is sufficiently addressed by the reporting requirements contained in 10 CFR 50.73.
03-06	A	Consistent with TSTF-90, a Note is added to the LCO that clarifies an RHR train's ECCS function is operable if it is capable of being manually realigned to the ECCS mode of operation. This is an administrative change to provide clarification.
03-07	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-08	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-09	M	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-10	LS 6 DC-ALL-001 [24]	Consistent with NUREG-1431, the requirement to demonstrate ECCS train OPERABILITY in MODE 4 in SR [4.5.3.1] has been revised to delete the 31 day surveillance to verify the correct position of each valve in the ECCS flow path which is not already locked in place, and the 18 month surveillance to verify automatic actuation of ECCS pumps and automatic valves. Insert @3.5.3-5
03-11	LG	The minimum RCS temperature limit below which the CCP and SI pumps must be demonstrated not capable of injecting into the RCS is replaced by the statement "below the temperature where LTOP is required as specified in the pressure temperature limits report (PTLR)." The minimum temperature is a plant specific requirement based on the reactor vessel material characteristics documented in the PTLR and is periodically reviewed and adjusted as required. Referring to the PTLR for the current value is consistent with the relocation of the pressure temperature limits from ITS Section 3.4.3 to

Insert for 3-10 LS-6

This change is acceptable because the ECCS operational requirements can be reduced due to the stable conditions associated with operation in MODE 4 and the decreased probability of occurrence of a Design Basis Accident (DBA). ECCS operational requirement reductions mean that certain automatic safety injection (SI) actuation signals are not available. However, in MODE 4, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.4-1 **APPLICABILITY:** DC

REQUEST:

DOC 5-04 LS-12
CTS 3.5.5 Actions a & b
ITS 3.5.4 Action A

The referenced DOC describes the change to the CTS but does not provide any justification for making the change other than that it is consistent with the STS.

Comment: The NSHC for this change appears to provide the needed justification. Therefore, please incorporate the information contained in the NSHC into the subject DOC.

FLOG RESPONSE: The information from NSHC LS-12 has been incorporated into DOC 5-04 LS-12. The revised justification is based on the fact that RWST temperature changes would be slow to occur and recognizes that the deviation from the analysis temperature assumption would probably be minor due to the moderate site temperature and that the contents of the tank are still available for injection. The additional time proposed will allow time to restore conditions and reduce the possibility of unnecessary plant transients and plant shutdowns, thus improving plant safety and increasing plant availability.

ATTACHED PAGES:

Encl. 3A 6

the PTLR.

CHANGE NUMBER	NSHC	DESCRIPTION
03-12	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-13	LG	Insert Q3.5.3-2
03-14	A	Insert Q3.5.3-3
04-01	LS 4	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-03	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-04	A	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-05	M	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-06	A	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-01	LS 7	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-02	LS 10	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-03	A	This change converts the RWST volume requirement from gallons to the equivalent percent of tank water level, in accordance with NUREG-1431. This change in method of indicating tank volume is administrative and does not result in a change in volume of the tank contents.
05-04	LS 12	This change modifies 3.5.5 ACTION a. to include the requirement for RWST borated water temperature to be above the minimum required temperature and ACTION b. to reference ACTION a. The ACTION for water temperature was in ACTION b with a Completion Time of 1 hour. With the water temperature included in ACTION a. the Completion Time is 8 hours. This change is consistent with NUREG-1431 and NUREG-1024. This change from 1 to 8 hours is a relaxation. Insert Q3.5.4-1
05-05	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). CP3.5-002

Insert for 05-04 LS-12

Changes in water temperature are slow and the change provides a more reasonable time in which to restore limits. The increase in allowed time to restore conditions to within limits prior to requiring a unit shutdown recognizes that the deviation from the analysis temperature assumption would probably be minor (if they ever occur) due to the moderate site temperature and that the contents of the tank are still available for injection. Also, the occurrence of an event requiring injection of the RWST contents during the increased Completion Time is extremely unlikely. The additional time proposed will reduce the possibility of unnecessary plant transients and plant shutdowns, thus improving plant safety and increasing plant availability.

The contents of the tank are still available for injection during the Completion Time and the extent of temperature outside the limit is not likely to be significant to safety.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.5-1 **APPLICABILITY:** DC, CP

REQUEST:

Section 3.4 DOC 6-21 LS-35
Section 3.5 JFD 3.5-4
CTS 3.4.5.2 Action b (CP)
CTS 3.4.6.2 Action b (DC)
ITS 3.5.5 Action A

This change is a change to both the CTS and the STS and is beyond the scope of the conversion review and is generic. DOC 6-21 states that this change is consistent with WOG-84.

Comment: Please provide the current status of WOG-84. If WOG-84 is not approved by the TSTF, then this change should be withdrawn from the conversion submittal at the time of the TSTF rejection. If WOG-84 has not been acted on by the TSTF, or is approved by the TSTF but not approved by the NRC by the time the draft safety evaluation is being prepared, then it should be withdrawn from the conversion submittal at that time. This change will not be reviewed on a plant-specific basis.

FLOG RESPONSE: DCPP and CPSES will continue to pursue the revisions proposed by this change. WOG-84 is now TSTF-236 which was approved by the TSTF on February 5, 1998. The NRC has requested that the WOG provide additional justification to support the extended Completion Time and changes to Required Action A. The WOG is preparing that information in addition to proposed changes to the 3.5.5 LCO and SRs. The revised traveler will be issued in the near future to the NRC.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: CA 3.5-001 **APPLICABILITY:** DC, WC, CA

REQUEST:

Revise the ITS 3.5.1 Bases to address Westinghouse NSAL 97-003 with regard to the relationship of permissive P-11 to the accumulator isolation valves. The discussion of P-11 is not relevant to this LCO which is applicable above 1000 psig. Nor is the IEEE 279-1971 "operating bypass" discussion relevant or correct per the current licensing basis.

ATTACHED PAGES:

Encl. 5B B 3.5-1 thru B 3.5-5, B 3.5-7, and B 3.5-8

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

the latter phase of blowdown to the beginning phase of reflood

BACKGROUND

The functions of the ECCS accumulators are to supply borated water to replace inventory in the reactor vessel during the ~~blowdown~~ ~~refill phase~~ of a loss of coolant accident (LOCA). ~~to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.~~

DC 3.5-001

The ECCS injection mode following a large break LOCA consists of three phases; 1) blowdown, 2) refill, and 3) reflood.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ~~ejection~~ spill out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of ~~safety injection (SI)~~ ECCS water.

DC 3.5-001 Insert

DC 3.5-ED

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

3.5.G-1

redline

order switched

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by two check valves in series and by an open motor operated isolation valve (8808A, B, C, and D). ~~The motor operated isolation valves are interlocked by P-11 with the pressurizer pressure measurement channels to ensure that the valves will automatically open as RCS pressure increases to above permissive circuit P-11 setpoint. to receive an "open" signal when permissive circuit P-11 is cleared. However, before permissive circuit P-11 is reached, these valves are manually opened and their motor operator breakers are sealed open to satisfy SR 3.5.1.5. Therefore, in the event of a LOCA, accumulator actuation is passive (Ref. [6]).~~

CA3.5-001

BASES

BACKGROUND
(continued)

CA 3.5-001

~~This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. However, if these valves were closed, they would be automatically opened as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.~~

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE
SAFETY ANALYSES

1 CA3.5-001

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2 and 4). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

3

DC-ALL-002

3.5.G-1

This paragraph was after next paragraph in NUREG-1431

The limiting large break LOCA is a double ended guillotine break in the RCS piping ~~at the discharge of the reactor coolant pump~~. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow with no credit taken for ECCS pump flow until an effective delay has elapsed. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break. No operator action is assumed during the blowdown stage of a large break LOCA.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.~~

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, ~~the rate of blowdown is such that~~ the SI pumps begin RCS injection, however, the increase in fuel clad temperature is terminated ~~solely~~ primarily by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and the ECCS centrifugal charging and SI pumps ~~both~~ play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease. The accumulators do not discharge above the pressure of their nitrogen cover gas (595.5 to 647.5 psig). At higher pressures the ECCS centrifugal charging pumps and SI pumps injection becomes solely responsible for terminating the temperature increase.

redline 3.5.G-1

DC3.5-ED

3.5.G-1

~~until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.~~

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. ~~2~~ 2) CA3.5-001 that are applicable for the accumulators will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium-water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

DC3.5-001

Since the accumulators discharge during the blowdown and reflood phases of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46, though their water volume is credited as part of the long term cooling inventory.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase

BASES

APPLICABLE SAFETY ANALYSES (continued)

in water volume is a peak clad temperature penalty. ~~For large breaks~~ Depending on the NRC-approved methodology used to analyze large breaks, an increase in water volume ~~can~~ may ~~be~~ result in either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The safety analysis assumes values of ~~≥[6468]~~ 60.8% (836 cubic feet) ~~gallons~~ and ~~≤[6879]~~ 72.6% (864 cubic feet) ~~gallons~~ as read on narrow range level instruments, not including instrument uncertainty. ~~To allow for instrument inaccuracy values of [6520] gallons and [6820] gallons are specified.~~

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction ~~in~~ below the accumulator LCO minimum boron concentration would produce a subsequent reduction in the available containment recirculation sump boron concentration for post LOCA shutdown and an increase in the ~~maximum~~ sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure (~~603 psia~~) (595.5 psig), since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit (~~693 psia~~) (647.5 psig) ~~prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.~~ provides margin to assure inadvertent relief valve actuation does not occur.

These analysis-assumed pressures are specified in the SRs. Volumes are shown on the control board indicators as % readings on accumulator narrow range level instruments. Adjustments to the analysis parameters for instrument inaccuracies or other reasons are applied to determine the acceptance criteria used in the plant surveillance procedures. These adjustments assure the assumed analyses parameters are maintained.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 4). CA35-001

The accumulators satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. ~~8~~ 2) could be violated. CA3.5-001

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above ~~[2000]~~ a nominal RCS pressure of 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at RCS pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. ~~8~~ 2) limit of 2200°F. CA3.5-001

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are normally closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

Insert CA3.5-001

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, the ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in

Enclosure 5B Page B3.5-5 (Applicability)

Accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. This condition is in agreement with the TS 3.4.12 LCO requirement.

SURVEILLANCE REQUIREMENTS (continued)

analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before 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

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as ~~stratification or~~ in-leakage. Sampling the affected accumulator within 6 hours after a solution volume increase ~~of 5.6 1%~~ (101 gallon) narrow range indicated level will identify whether in-leakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), and the RWST has not been diluted since verifying that its boron concentration satisfies SR 3.5.4.3, because the water contained in the RWST is within the accumulator boron concentration requirements as verified by SR 3.5.4.3. This is consistent with the recommendation of GL 93-05 (Ref. ~~8~~ 4).

3.5.G-1

~~of a 1% volume~~

CA3.5-001

~~NUREG-1366~~ 3.5.G-1

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator (8808A, B, C, and D) when the ~~pressurizer~~ RCS pressure is ~~≥ 2000~~ greater than 1000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when ~~pressurizer~~ RCS pressure is ~~< 2000~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

less than or equal to 1000 psig, thus allowing ~~operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.~~ the valves to be closed to enable plant shutdown without discharging the accumulators into the RCS. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock (P-11) associated with the valves.

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

CA3.5-001

REFERENCES

~~1. IEEE Standard 279-1971.~~

CA3.5-001

1 ~~2.~~ FSAR, Chapter ~~[6]~~.

2 ~~3.~~ 10 CFR 50.46.

3 ~~4.~~ FSAR, Chapter ~~[15]~~.

4 ~~5.~~ GL 93-05, Item 7.1.

5 ~~6.~~ [DCM S 38A]

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: CA 3.5-002 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

Revise ITS 3.5.4 Bases to indicate that the RWST LCO, by virtue of its temperature, volume, and boron concentration limits, also satisfies Criterion 2 (initial conditions of accident analyses).

ATTACHED PAGES:

Enclosure 5B B 3.5-31

BASES

APPLICABLE SAFETY ANALYSES (continued)

Steam Generator Tube Rupture (SGTR)

Volume

The RWST volume needed in response to a SGTR is not an explicit assumption since the required volume is much less than that required by a LOCA.

Boration

Borated RWST water will be injected into the RCS for a SGTR event. The insertion of the control rods and the negative reactivity provided by the injected RWST solution provides sufficient SDM during the initial recovery operations. One of the initial operator recovery actions for this event is to equalize the RCS pressure and the faulted steam generator pressure to minimize or stop the primary-to-secondary tube rupture flow and terminate safety injection. Further RCS boration will be initiated by the operator by manual makeup to the RCS.

The RWST satisfies ~~Criterion~~ Criteria 2 and 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). CA 3.5-002

Insert 3 DC 3.5-005

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment recirculation sump to support ECCS ~~and Containment Spray System~~ pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and CS ~~Containment Spray~~ System OPERABILITY requirements. Since both the ECCS and the CS ~~Containment Spray~~ Systems must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and LCO 3.9.6, "Residual DC 3.5-ED

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC 3.5-ED **APPLICABILITY:** DC

REQUEST:

Various editorial changes that do not impact the technical content of the submittal or other FLOG members.

Changes are noted with DC-3.6-ED in the right margin and noted below:

1) Enclosure 5B, page B3.5-1, safety injection (SI) changed to ECCS to be consistent with FSAR language
2) Enclosure 5A, page 3.5-9, JFD 3.5-8 does not apply to Diablo Canyon. "B-PS" should be used.
3) Enclosure 5B, page B3.5-31, singular versus plural use.
4) Enclosure 5B, page B3.5-3, the word "both" is deleted since three items are listed.
5) Enclosure 5B, page B3.5-14, paragraph on Notes is revised to be consistent with Enclosure 5A mark-up.

ATTACHED PAGES:

Encl. 5A	3.5-9
Encl. 5B	B3.5-1, B3.5-3, B3.5-14, B3.5-31

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.5.4.1	NOTE Only required to be performed when ambient air temperature is ~~< [35]°F or > [100]°F.~~ Verify RWST borated water temperature is ≥~~[35]~~°F ~~and ≤ [100]°F.~~	24 hours	B-PS ~~3.5-8~~ B-PS DC 3.5-ED
SR 3.5.4.2	Verify RWST borated water volume is ≥ ~~[466,200~~ 400,000 gallons (81.5% indicated level) ~~]~~	7 days	B-PS
SR 3.5.4.3	Verify RWST boron concentration is ≥ ~~[2000~~ 2300~~]~~ ppm and ≤ ~~[2200 2500]~~ ppm.	7 days	B-PS

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

the latter phase of blowdown to the beginning phase of reflood

BACKGROUND

The functions of the ECCS accumulators are to supply borated water to replace inventory in the reactor vessel during the ~~blowdown~~ ~~refill phase~~ of a loss of coolant accident (LOCA). ~~to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.~~ DC3.5-001

The ECCS injection mode following a large break LOCA consists of three phases; 1) blowdown, 2) refill, and 3) reflood.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ~~ejection~~ spill out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of ~~safety injection (SI)~~ ECCS water. DC3.5-001 Insert DC3.5-ED

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

3.5.G-1

order switched

redline

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by two check valves in series and by an open motor operated isolation valve (8808A, B, C, and D). ~~The motor operated isolation valves are interlocked by P-11 with the pressurizer pressure measurement channels to ensure that the valves will automatically open as RCS pressure increases to above permissive circuit P-11 setpoint. to receive an "open" signal when permissive circuit P-11 is cleared. However, before permissive circuit P-11 is reached, these valves are manually opened and their motor operator breakers are sealed open to satisfy SR 3.5.1.5. Therefore, in the event of a LOCA, accumulator actuation is passive (Ref. [6]).~~ CA3.5-001

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.~~

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, ~~the rate of blowdown is such that~~ the SI pumps begin RCS injection, however, the increase in fuel clad temperature is terminated ~~solely~~ primarily by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and the ECCS centrifugal charging and SI pumps ~~both~~ play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease. The accumulators do not discharge above the pressure of their nitrogen cover gas (595.5 to 647.5 psig). At higher pressures the ECCS centrifugal charging pumps and SI pumps injection becomes solely responsible for terminating the temperature increase.

redline 3.5.G-1

DC3.5-ED

3.5.G-1

~~Until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.~~

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. ~~1~~ 2) CA3.5-001 that are applicable for the accumulators will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium-water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding cladding surrounding the plenum volume, were to react; and

DC3.5-001

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown and reflood phases of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46, though their water volume is credited as part of the long term cooling inventory.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase

BASES

APPLICABILITY (continued)

This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS – Shutdown." DC 3.5-ED

un-strike

~~As indicated in Note 2, the flow path may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room. As indicated in Note 2, operation in MODE 3 with ECCS trains declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered inoperable at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status.~~

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available (capable of injection into the RCS, if actuated), the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their ~~design~~ safety function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to

BASES

APPLICABLE SAFETY ANALYSES (continued)

Steam Generator Tube Rupture (SGTR)

Volume

The RWST volume needed in response to a SGTR is not an explicit assumption since the required volume is much less than that required by a LOCA.

Boration

Borated RWST water will be injected into the RCS for a SGTR event. The insertion of the control rods and the negative reactivity provided by the injected RWST solution provides sufficient SDM during the initial recovery operations. One of the initial operator recovery actions for this event is to equalize the RCS pressure and the faulted steam generator pressure to minimize or stop the primary-to-secondary tube rupture flow and terminate safety injection. Further RCS boration will be initiated by the operator by manual makeup to the RCS.

The RWST satisfies ~~Criterion~~ Criteria 2 and 3 of ~~the NRC Policy Statement~~ ~~10 CFR 50.36(c)(2)(ii).~~ CA 3.5-002

Insert 3 DC 3.5-005

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment recirculation sump to support ECCS ~~and Containment Spray System~~ pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and CS ~~Containment Spray~~ System OPERABILITY requirements. Since both the ECCS and the CS ~~Containment Spray~~ Systems must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual DC 3.5-ED

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC ALL-002 **APPLICABILITY:** DC

REQUEST:

An errata to LAR 97-09 was submitted to the NRC January 8, 1998 in DCL-98-003. Errata changes on pages affected by NRC comment numbers are indicated with "DC-ALL-002." Errata changes that dealt with issuance of LAs 119/117 and 118/116 (issued 7/13/97) that addressed CTS surveillance interval increases due to 24-month fuel cycles are indicated with "DC-ALL-001."

ATTACHED PAGES:

See notations on applicable pages for each comment number.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC 3.5-001 **APPLICABILITY:** DC, WC, CA

REQUEST:

Revise the Bases 3.5.1, Background, to discuss the three phases for large break LOCA (blowdown, refill, and reflood) as discussed in FSAR Chapter 15. The revision clarifies that reflood is accomplished initially by accumulator discharge and by ECCS pump flow.

ATTACHED PAGES:

Encl. 5B B 3.5-1 and B 3.5-3

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

the latter phase of blowdown to the beginning phase of reflood

BACKGROUND

The functions of the ECCS accumulators are to supply borated water to replace inventory in the reactor vessel during the ~~blowdown~~ ~~refill phase~~ of a loss of coolant accident (LOCA). ~~to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.~~ DC 3.5-001

The ECCS injection mode following a large break LOCA consists of three phases; 1) blowdown, 2) refill, and 3) reflood.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ~~ejection~~ spill out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of ~~safety injection (SI)~~ ECCS water. DC 3.5-001 Insert DC 3.5-ED

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

3.5.6-1

order switched

redline

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by two check valves in series and by an open motor operated isolation valve (8808A, B, C, and D). ~~The motor operated isolation valves are interlocked by P-11 with the pressurizer pressure measurement channels to ensure that the valves will automatically open as RCS pressure increases to above permissive circuit P-11 setpoint. to receive an "open" signal when permissive circuit P-11 is cleared. However, before permissive circuit P-11 is reached, these valves are manually opened and their motor operator breakers are sealed open to satisfy SR 3.5.1.5. Therefore, in the event of a LOCA, accumulator actuation is passive (Ref. [6]).~~ CA3.5-001

Insert in Enclosure 5B, page B3.5-1

The refill phase is complete when the injection of ECCS water has filled the reactor vessel downcomer and the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods.

The reflood phase follows the refill phase and continues until the reactor vessel has been filled to the extent that core temperature rise has been terminated.

The accumulators function in the later stage of blowdown to the beginning of reflood to fill the downcomer and lower plenum. The injection of the ECCS pumps aid during refill. Reflood and the following long term heat removal is accomplished by water pumped into the core by the ECCS pumps.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.~~

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, ~~the rate of blowdown is such that~~ the SI pumps begin RCS injection, however, the increase in fuel clad temperature is terminated ~~solely~~ primarily by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and the ECCS centrifugal charging and SI pumps ~~both~~ play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease. The accumulators do not discharge above the pressure of their nitrogen cover gas (595.5 to 647.5 psig.) At higher pressures the ECCS centrifugal charging pumps and SI pumps injection becomes solely responsible for terminating the temperature increase.

redline 3.5.G-1

DC3.5-ED

3.5.G-1

~~until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.~~

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. ~~3~~ 2) CA3.5-001 that are applicable for the accumulators will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium-water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

DC3.5-001

Since the accumulators discharge during the blowdown and reflood phases of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46, though their water volume is credited as part of the long term cooling inventory.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC 3.5-002 **APPLICABILITY:** DC

REQUEST:

Revise ITS SR 3.5.5.1 by adding a second note that states: "The provisions of specification SR 3.0.4 are not applicable for entry into MODE 3." This note is equivalent to the current technical specification 4.4.6.2.1 c. note except that it does not apply to MODE 4 entry since ITS 3.5.5 does not apply to MODE 4.

ATTACHED PAGES

Encl. 5A	3.5-11
Encl. 5B	B 3.5-38

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.5.1 ----------NOTES---------- 1. Not required to be performed until 4 hours after the Reactor Coolant System pressure stabilizes at ≥ ~~[~~2215 psig and ≤ 2255 psig.~~]~~ B 2. The provisions of SR 3.0.4 are not applicable for entry into MODE 3. DC 3.5-002 ---------- Verify manual seal injection throttle valves are adjusted to give a flow within limit with ~~[centrifugal charging pump discharge header]~~ RCS pressure ≥ ~~[~~ ~~2480~~ 2215 psig and ≤ 2255~~]~~ psig and the ~~[~~ charging flow~~]~~ control valve full open.	31 days B 3.5-5

BASES

ACTIONS (continued)

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown must be initiated. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown begun in Required Action B.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Verification every 31 days that the manual seal injection throttle valves are adjusted to give a flow ~~within~~ below the limit ensures ~~that~~ proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained. The Frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve Surveillance Frequencies. The Frequency has proven to be acceptable through operating experience.

As noted, the Surveillance is ~~not required~~ to be ~~performed until~~ completed within 4 hours after the RCS pressure has stabilized within ~~a ± 20 psig range of normal operating~~ the specified pressure limits. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the Surveillance is timely.

DC 3.5-002

REFERENCES

1. FSAR, Chapter 6 and Chapter 15.
2. 10 CFR 50.46.

This surveillance is further modified by Note 2 that states that the provisions of SR 3.0.4 are not applicable for entry into MODE 3.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC 3.5-003 **APPLICABILITY:** DC

REQUEST:

Revise the Bases of SR 3.5.2.4 to include information on the differential pressure that was inadvertently left out. Specifically, the following text is added: "The following ECCS pumps are required to develop the indicated differential pressure on recirculation flow: 1) CCP ≥ 2400 psid, 2) SI pump ≥ 1455 psid, and 3) RHR pump ≥ 165 psid."

ATTACHED PAGES

Encl. 5B B 3.5-18

BASES

REQUIREMENTS (continued)

~~Section XI of~~ the ASME Code. (Ref. 8). This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is ~~greater than or equal to~~ within the performance assumed in the plant safety analysis. SRs are specified in ~~Technical Requirements Manual and in~~ the applicable portions of the Inservice Testing Program, which encompasses ~~Section XI~~ Part 6 of the ASME Code for Operation and Maintenance of Nuclear Power Plants. (Ref. 8). ~~Section XI~~ This section of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.

The following ECCS pumps are required to develop the indicated differential pressure when tested on recirculation flow:
CCP ≥ 2400 psid
SI pump ≥ 1455 psid
RHR pump ≥ 165 psid.

DC 3.5-003

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

24 DC-ALL-001 24

SR 3.5.2.7

3.5.6-1

The correct ~~Realignment~~ position of throttle/runout valves in the ECCS flow paths ~~on an SI signal~~ is necessary for proper ECCS performance. These manual throttle/runout valves are positioned during flow balancing and have mechanical locks and seals ~~stops~~ to ~~allow~~ ensure that the proper positioning for restricted flow to a ruptured cold leg ~~ensuring~~ is maintained. The verification of proper position of a throttle/runout valve can be accomplished by confirming the seals ~~and lock~~ have not been altered since the last performance of the flow balance test. Restricting the flow to a ruptured cold leg ensures ~~and~~ that the other cold legs receive at least the required minimum flow. ~~This Surveillance is not required for~~

redline

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC 3.5-005 **APPLICABILITY:** DC

REQUEST:

Revise Bases 3.5.4 to include additional information on RWST temperature requirements. The ITS Bases for ITS 3.5.4 currently discusses the basis of the RWST temperature surveillance. However, the temperature is not discussed in the Applicable Safety Analysis section in the same level of detail as other RWST parameters. This difference in the level of detail resulted from a previous PG&E request to delete the RWST temperature requirement. The deletion was not approved. Therefore, this enhancement will make information on RWST temperature more consistent with the description of other RWST parameters.

ATTACHED PAGES

Encl. 5B B 3.5-30 and B 3.5-31

BASES

APPLICABLE SAFETY ANALYSES (continued)

which increases the differential pressure provided by the downcomer head (this phenomena is sometimes referred to as steam binding). Thus, a higher downcomer mixture level is required to maintain the same reflood rate as before. The additional time required to establish the downcomer head translates into a reduction in the reflood rate in the core. When the downcomer has completely filled, the equilibrium reflood rate for the low containment pressure case would be less than that calculated for a high containment pressure case. This reduction in reflood rate results in a reduction in heat transfer and ultimately an increase in the calculated PCT. Thus, the regulations require that a low containment pressure be calculated in the large-break LOCA analysis.

DC 3.5-005

When calculating containment back pressure for LOCA peak clad temperature

~~In the ECCS~~ analysis, the CS temperature is assumed to be equal to the RWST minimum temperature limit of 35°F. If the minimum temperature limit is violated, the CS ~~containment spray~~ further reduces containment pressure, which decreases the core reflood as explained in the preceding paragraph. For the containment response following a MSLB, the lower limit on boron concentration is used to maximize the total energy release to containment.

redline 3.5.6-1

Insert 1

DC 3.5-005

Steam Line and Feedwater Line Breaks

Volume

RWST volume is not an explicit assumption in other than LOCA events since the required volume for those events is much less than that required by LOCA.

redline 3.5.6-1

Boration

The minimum RWST solution boron concentration is an explicit assumption in the MSLB analysis to ensure the required shutdown capability. Since DCPP no longer uses the boron injection tank, the minimum boron concentration limit is an important assumption in ensuring the required shutdown capability. For the containment response following an MSLB, the lower limit on boron concentration is used to maximize the total energy release to containment.

Feedwater line break results in high temperature/high pressure in the RCS. There is very little RWST water injected due to the high pressure. Also, the analysis results are not affected by the negative reactivity provided by RWST water. Therefore, RWST boron concentration is not a consideration for the feedwater line break.

Insert 2

DC 3.5-005

BASES

APPLICABLE SAFETY ANALYSES (continued)

Steam Generator Tube Rupture (SGTR)

Volume

The RWST volume needed in response to a SGTR is not an explicit assumption since the required volume is much less than that required by a LOCA.

Boration

Borated RWST water will be injected into the RCS for a SGTR event. The insertion of the control rods and the negative reactivity provided by the injected RWST solution provides sufficient SDM during the initial recovery operations. One of the initial operator recovery actions for this event is to equalize the RCS pressure and the faulted steam generator pressure to minimize or stop the primary-to-secondary tube rupture flow and terminate safety injection. Further RCS boration will be initiated by the operator by manual makeup to the RCS.

The RWST satisfies ~~Criterion~~ Criteria 2 and 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). CA 3.5-002

Insert 3 DC 3.5-005

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment recirculation sump to support ECCS ~~and Containment Spray System~~ pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and CS ~~Containment Spray~~ System OPERABILITY requirements. Since both the ECCS and the CS ~~Containment Spray~~ Systems must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and LCO 3.9.6, "Residual DC 3.5-ED

Inserts in Enclosure 5B, pages 30 and 31

Insert 1: Temperature

The primary reason for the TS minimum RWST temperature is to ensure the water will be above freezing. In addition, the LOCA analysis SATAN code assumes the containment spray temperature to be equal to the RWST TS temperature limit of 35 degrees F. Low water temperature can affect the analysis model of containment spray to result in a reduction of containment pressure, which affects core reflood and increases peak clad temperature.

Insert 2: Temperature

Minimum temperature is assumed in the MSLB core response analysis. Assuming minimum temperature for the MSLB is conservative as a MSLB causes substantial RCS cooling due to uncontrolled steam release and increases core reactivity. Cold water adds positive reactivity, however this effect is covered by the negative reactivity provided by the boron in the RWST water.

Minimum RWST temperature is not assumed for the feedwater line break, since warmer RWST temperatures are more limiting. However, since RCS pressure remains high during this event, there is very little RWST water injected and the temperature does not have a significant effect.

Insert 3: Temperature

Minimum RWST water temperature is not a factor in SGTR. The heat capacity of RWST water injected into the RCS is small relative to the RCS inventory and heat sources.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC 3.5-006 **APPLICABILITY:** DC

REQUEST:

Revise the SR Bases 3.5.2.3 to clarify what is required to verify that the ECCS piping is full of water.

ATTACHED PAGES

Encl. 5B B 3.5-17

BASES

REQUIREMENTS (continued)

for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

With the exception of the operating CCP ~~centrifugal charging pump~~, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of non-condensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.

gas bindings

The intent of the SR is to assure the ECCS piping is full of water. Different means of verification, as alternates to venting the accessible system high points, can be employed to provide this assurance. DC 3.5-006

such as verifying full the vent lines of the ECCS pump casings (for non-running pumps) and accessible high point vents.

~~Venting of the accessible ECCS high points prior to entering MODE 3 ensures the system is full of water and will perform properly, injecting its full capacity into the RCS on demand.~~

~~The CCP design and attached piping configuration allow the CCP to vent the accumulated gases via the attached suction and discharge piping. Continuous venting of the suction piping to the Volume Control tank (VCT) and manual venting of the discharge piping high points satisfies the pump casing venting requirements for the CCPs.~~

SR 3.5.2.4

SURVEILLANCE

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by

ACRONYM LIST

ADV	Atmospheric Dump Valve		RCS	Reactor Coolant System
CFCU	Containment Fan Cooling Unit		RG	Regulatory Guide
CIV	Containment isolation Valve		RHR	Residual Heat Removal
CPSES	Comanche Peak Steam Electric Station		SER	Safety Evaluation Report
CTS	Current Tech Spec		SR	Surveillance Requirement
DBA	Design Basis Accident		STS	Standard Tech Spec
DLAP	Department Level Administrative Procedure		TRM	Technical Requirements Manual
DOC	Description of Change		TSTF	Tech Spec Task Force
ECCS	Emergency Core Cooling System		USAR	Updated Safety Analysis Report
ECG	Equipment Control Guideline		WCGS	Wolf Creek Generating Station
EQ	Environment Qualification		WOG	Westinghouse Owners Group
FLOG	Four Loop Owner's Group			
FSAR	Final Safety Analysis Report			
GDC	Generic Design Criteria			
IEEE	Institute of Electrical and Electronic Engineers			
IST	In-service Test			
ITS	Improved Tech Spec			
JFD	Justification For Differences			
LA	License Amendment			
LAR	License Amendment Request			
LCO	Limiting Condition of Operation			
LER	Licensee Event Report			
LG	Less Restrictive			
LOCA	Loss of Coolant Access			
MFIV	Main Feedwater Isolation Valve			
MFRV	Main Feedwater Regulating Valves			
MSSV	Main Steam Safety Valve			
PAMS	Post Accident Monitoring System			
RA	Required Actions			
RAI	Request for additional information			

This letter and the Enclosures are not a supplement to Reference 3, and thus have not been reviewed and approved by DCPP's Plant Staff Review Committee. A supplement to Reference 3 will be provided at a later date. Any deviations from the responses provided in this letter will be discussed in the supplement.

Sincerely,

Gregory M. Rueger

Subscribed and sworn to before me this 5th day of August 1998
State of California
County of San Luis Obispo

Notary Public

cc: Edgar Bailey, DHS
Steven D. Bloom
Dennis F. Kirsch
Ellis W. Merschoff
David L. Proulx
Howard J. Wong
Diablo Distribution

DANA LOPEZ
COMM. #1096594
NOTARY PUBLIC CALIFORNIA
SAN LUIS OBISPO COUNTY
My Comm. Expires May 5, 2000

Enclosures
MRZ/

JLS CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS

CTS 3/4.1 - REACTIVITY CONTROL SYSTEMS
ITS 3.1 - REACTIVITY CONTROL SYSTEMS

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION AND LICENSEE INITIATED ADDITIONAL CHANGES

INDEX OF ADDITIONAL INFORMATION

ADDITIONAL INFORMATION NUMBER	APPLICABILITY	ENCLOSED
3.1.G-1	DC, CP, WC, CA	YES
3.1-1	WC, CA	NA
3.1-2	DC, CP	YES
3.1-3	DC, CP, WC, CA	YES
3.1-4	DC, CP	YES
3.1-5	CP	NA
3.1-6	DC, CP	YES
3.1-7	DC, CP	YES
3.1-8	DC, CP	YES
3.1-9	DC	YES
3.1-10	DC, CP	YES
3.1-11	DC, CP	YES
3.1-12	DC, CP	YES
3.1-13	DC, CP, WC, CA	YES
3.1-14	WC	NA
3.1-15	DC, CP, WC, CA	YES
3.1-16	DC, CP, WC, CA	YES
3.1-17	CP	NA
3.1-18	CP	NA
3.1-19	WC, CA	NA
3.1-20	DC, CP	YES
3.1-21	DC, CP	YES
3.1-22	DC	YES
3.1-23	WC	NA
3.1-24	DC, CP, WC, CA	YES
3.1-25	DC, CP, WC, CA	YES
3.1-26	CP	NA
3.1-27	DC, CP, WC, CA	YES
3.1-28	DC, CP, WC, CA	YES
CA 3.1-001	WC, CA	NA
CA 3.1-003	CA	NA
CA 3.1-004	CA	NA
CP 3.1-ED	CP	NA
CP 3.1-002	CP	NA
CP 3.1-003	CP	NA
DC 3.1-ED	DC	YES
DC 3.1-001	DC, CP	YES
DC-ALL-001 (3.1 changes only)	DC	see DCL-98-003

INDEX OF ADDITIONAL INFORMATION
(cont.)

ADDITIONAL INFORMATION NUMBER	APPLICABILITY	ENCLOSED
DC ALL-002 (3.1 changes only)	DC	see DCL-98-003
TR 3.1-001	DC, CP, WC, CA	YES
TR 3.1-003	DC, CP, WC, CA	YES
TR 3.1-004	DC, CP, WC, CA	YES
TR 3.1-005	DC, CP, WC, CA	YES
TR 3.1-006	DC, CP, WC, CA	YES
WC 3.1-ED	WC	NA

JOINT LICENSING SUBCOMMITTEE METHODOLOGY FOR PROVIDING ADDITIONAL INFORMATION

The following methodology is followed for submitting additional information:

1. Each licensee is submitting a separate response for each section.

2. If an RAI does not apply to a licensee (i.e., does not actually impact the information that defines the technical specification change for that licensee), "NA" has been entered in the index column labeled "ENCLOSED" and no information is provided in the response for that licensee.

3. If a licensee initiated change does not apply, "NA" has been entered in the index column labeled "ENCLOSED" and no information is provided in the response for that licensee.

4. The common portions of the "Additional Information Cover Sheets" are identical, except for brackets, where applicable (using the same methodology used in enclosures 3A, 3B, 4, 6A and 6B of the conversion submittals). The list of attached pages will vary to match the licensee specific conversion submittals. A licensee's FLOG response may not address all applicable plants if there is insufficient similarity in the plant specific responses to justify their inclusion in each submittal. In those cases, the response will be prefaced with a heading such as "PLANT SPECIFIC DISCUSSION."

5. Changes are indicated using the redline/strikeout tool of WordPerfect or by using a hand markup that indicates insertions and deletions. If the area being revised is not clear, the affected portion of the page is circled. The markup techniques vary as necessary, based on the specifics of the area being changed and the complexity of the changes, to provide the clearest possible indication of the changes.

6. A marginal note (the Additional Information Number from the index) is added in the right margin of each page being changed, adjacent to the area being changed, to identify the source of each change.

7. Some changes are not applicable to one licensee but still require changes to the Tables provided in Enclosures 3A, 3B, 4, 6A, and 6B of the original license amendment request to reflect the changes being made by one or more of the other licensees. These changes are not included in the additional information for the licensee to which the change does not apply, as the changes are only for consistency, do not technically affect the request for that licensee, and are being provided in the additional information being provided by the licensees for which the change is applicable. The complete set of changes for the license amendment request will be provided in a licensing amendment request supplement to be provided later.

JOINT LICENSING SUBCOMMITTEE METHODOLOGY FOR PROVIDING ADDITIONAL INFORMATION (cont.)

8. The item numbers are formatted as follows:

   [Source] [ITS Section]-[nnn]

   Source =
   - Q - NRC Question
   - CA - AmerenUE
   - DC - PG&E
   - WC - WCNOC
   - CP - TU Electric
   - TR - Traveler

   ITS Section = The ITS section associated with the item (e.g., 3.3). If all sections are potentially impacted by a broad change or set of changes, "ALL" is used for the section number.

   nnn = a three digit sequential number

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.1.G-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST:

ITS 3.1.x Bases

General

There have been a number of instances that the specific changes to the STS Bases are not properly identified with redline or strikeout marks.

Comment: Perform an audit of all STS Bases markups and identify instances where additions and/or deletions of Bases were not properly identified in the original submittal.

FLOG response: The submitted ITS Bases markups for Section 3.1 have been compared to the STS Bases. Some differences that were identified were in accordance with the markup methodologies (e.g., deletion of brackets and reviewer's notes). Most of the differences were editorial in nature and would not have affected the review. Examples of editorial changes are:

1) Capitalizing a letter with only a "redline" but not striking out the lower case letter that it replaced.
2) Changing a verb from singular to plural by adding an "s" without "redlining" the "s."
3) Deleting instead of striking-out the A, B, C, etc., following a specification title (e.g., SR3.6.6A.7).
4) Changing a bracketed reference (in the reference section) with only a "redline" for the new reference but failing to include the strike-out of the old reference.
5) In some instances, the brackets were retained (and struck-out) but the unchanged text within the brackets was not redlined.
6) Not redlining a title of a bracketed section. The methodology calls for the section title to be redlined when an entire section was bracketed.
7) Additional text not contained in the STS Bases was added to the ITS Bases by the lead FLOG member during the development of the submittal. Once it was determined to not be applicable, the text was then struck-out and remains in the ITS Bases mark-up.

Differences of the above editorial nature will not be provided as attachments to this response. The pages requiring changes that are more than editorial and are not consistent with the markup methodology are attached.

ATTACHED PAGES:

Encl. 5B B 3.1-1, B 3.1-2, B 3.1-2a, B 3.1-3, B 3.1-3a, B 3.1-7, B 3.1-11, B 3.1-13a, B 3.1-15, B 3.1-16a, B 3.1-17, B 3.1-18a, B 3.1-19, B 3.1-19a, B 3.1-23, B 3.1-24a, B 3.1-27, B 3.1-29

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.5.2-4 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

DOC 2-12 LG
CTS 4.5.2.f
ITS SR 3.5.2.4

The referenced DOC describes the change to the CTS but does not provide any justification for making the change other than that it is consistent with the STS.

Comment: Please revise the DOC to include additional justification as to why this detail is not necessary in the ITS.

FLOG RESPONSE: DOC 2-12-LG has been revised to provide additional justification for the proposed change by adding the following information:

"ITS SR 3.5.2.4 retains the SR requirement and references the Inservice Testing (IST) Program, discussed in ITS 5.5.8, for the surveillance Frequency. The specific SR acceptance criteria for the pumps have been moved to the ITS SR 3.5.2.4 Bases. Although this may make the ECCS pump performance testing more flexible in the future, in regard to licensee control over the numerical values of the acceptance criteria, this testing must continue to conform to the IST Program requirements. Revisions to the acceptance criteria will have to meet the requirements of the Bases Control Program discussed in ITS 5.5.14. Details for performing surveillance requirements are more appropriately specified in the plant procedures required by ITS 5.4.1 and the ITS Bases. Control of the acceptance criteria for a surveillance test is an issue for the IST procedures and has been previously determined by the NRC to be unnecessary as a TS restriction. As indicated in Generic Letter 91-04, allowing this licensee control is consistent with the vast majority of other surveillance requirements that do not dictate plant conditions for surveillances."

ATTACHED PAGES:

Encl. 3A 3

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: TR 3.5-001 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

Revise Traveler Status Sheet to: 1) reflect NRC approval of three travelers: TSTF-90, Revision 1; TSTF-117; and TSTF-153; 2) delete reference to TSTF-155, which was rejected by TSTF and not incorporated by the FLOG, and 3) change WOG-84 to TSTF-236. There are no changes involved to any CTS mark-ups, ITS mark-ups, DOCs, or JFDs.

ATTACHED PAGES:

Encl. 5A	Traveler Status Sheet
Encl. 6A	1

Industry Travelers Applicable to Section 3.5

TR 3.5-001

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-90, Rev. 1	Incorporated	3.5-6	Approved by NRC.
TSTF-117, Rev. 2	Incorporated	3.5-1	Approved by NRC.
TSTF-153	Incorporated	3.5-8	Approved by NRC.
~~TSTF-155~~	~~Not Incorporated~~	~~N/A~~	~~Not NRC approved as of traveler cut-off date.~~
~~WOG-84~~ TSTF-236	Incorporated	3.5-4	DCPP and CPSES only.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431
NUREG-1431 Section 3.5

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[ ]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER	JUSTIFICATION
3.5-1	This change replaces reference to the "pressurizer pressure" with a reference to the "RCS pressure" in the APPLICABILITY, Required Action C.2, and SR 3.5.1.5. Required ACTION C.2 requires reducing pressurizer pressure to less than 1000 psig. However, pressurizer pressure instrumentation does not have the range to read that pressure. Consequently, RCS pressure instrumentation is used. For the purposes of this LCO, the use of RCS pressure is equivalent. This is consistent with Industry Traveler 117.
3.5-2	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 6B).
3.5-3	This change adds the word "mechanical" with regard to throttle valve position stop, consistent with the CTS. These valves have mechanical stops that maintain the valves in position for proper ECCS performance.
3.5-4	This change increases the RCP seal injection flow Completion Time from 4 to 72 hours, with a new added verification that at least 100 percent of the assumed charging flow remains available. The Bases for seal injection flow relate the limit to ensuring adequate charging flow during post-LOCA injection. The revised ACTIONS continue to assure this basis is adequately addressed by providing an ECCS-like Required Action. ITS 3.5.2 allows a 72 hour Completion Time for 1 or more ECCS subsystems inoperable if at least 100 percent of the assumed ECCS flow is available. The seal injection flow ACTIONS have been modified so that if the remaining charging flow (with some inoperability in the charging system) is greater than or equal to 100 percent of the assumed post-LOCA charging flow, 72 hours is allowed to restore OPERABILITY. This change is consistent with industry Traveler ~~WOG-84~~ TSTF-236 TR 3.1-001
3.5-5	This change deleted reference to CCP discharge header pressure from the LCO and ACTION A to reflect CTS [3.4.6.2.]. A description is added to the Bases which provides the methodology for adjusting the seal injection throttle valves consistent with plant-specific analyses.

JLS CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS

CTS 3/4.9 -REFUELING OPERATIONS
ITS 3.9 - REFUELING OPERATIONS

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION AND LICENSEE INITIATED ADDITIONAL CHANGES

INDEX OF ADDITIONAL INFORMATION

ADDITIONAL INFORMATION NUMBER	APPLICABILITY	ENCLOSED
3.9.G-1	DC, CP, WC, CA	YES
3.9-i	DC, CP, WC, CA	YES
3.9-1a	DC, CP, WC, CA	YES
3.9-1b	CA	NA
3.9-2	CP	NA
3.9-3	CP, WC, CA	NA
3.9-4	DC, CP, WC, CA	YES
3.9-5	DC, CP, WC, CA	YES
3.9-6	CA	NA
3.9-7	DC, CP, WC, CA	YES
3.9-8	DC, CP, WC, CA	YES
3.9-9	DC, CP	YES
3.9-10	DC, CP	YES
3.9-11	DC, CP	YES
3.9-12	DC, CP, WC, CA	YES
3.9-13	DC, CP, WC	YES
3.9-14	DC, WC, CA	YES
3.9-15	DC, WC, CA	YES
3.9-16	DC, WC, CA	YES
3.9-17	DC, CP	YES
3.9-18	DC, CP	YES
3.9-19	CA	NA
3.9-20	WC	NA
3.9-21	DC, CP, WC, CA	YES
3.9-22	DC, CP, WC, CA	YES
3.9-23	DC, CP, WC, CA	YES
3.9-24	DC, CP, WC, CA	YES
3.9-25	DC	YES
CP 3.9-001	CP	NA
CP 3.9-002	CP	NA
CP 3.9-003	CP	NA
CP 3.9-004	CP	NA
DC 3.9-ED	DC	YES
DC ALL-001 (3.9 changes only)	DC	see DCL-98-003
DC ALL-003 (3.9 changes only)	DC	YES
TR 3.9-001	DC, CP, WC, CA	YES
TR 3.9-002	DC, CP, WC, CA	YES
TR 3.9-003	DC	YES

INDEX OF ADDITIONAL INFORMATION
(cont.)

WC 3.9-ED	WC	NA
WC 3.9-001	WC	NA
WC 3.9-002	WC	NA
WC 3.9-003	WC	NA
WC 3.9-004	WC	NA
WC 3.9-006	CP, WC	NA

JOINT LICENSING SUBCOMMITTEE METHODOLOGY FOR PROVIDING ADDITIONAL INFORMATION

The following methodology is followed for submitting additional information:

1. Each licensee is submitting a separate response for each section.

2. If an RAI does not apply to a licensee (i.e., does not actually impact the information that defines the technical specification change for that licensee), "NA" has been entered in the index column labeled "ENCLOSED" and no information is provided in the response for that licensee.

3. If a licensee initiated change does not apply, "NA" has been entered in the index column labeled "ENCLOSED" and no information is provided in the response for that licensee.

4. The common portions of the "Additional Information Cover Sheets" are identical, except for brackets, where applicable (using the same methodology used in enclosures 3A, 3B, 4, 6A and 6B of the conversion submittals). The list of attached pages will vary to match the licensee specific conversion submittals. A licensee's FLOG response may not address all applicable plants if there is insufficient similarity in the plant specific responses to justify their inclusion in each submittal. In those cases, the response will be prefaced with a heading such as "PLANT SPECIFIC DISCUSSION."

5. Changes are indicated using the redline/strikeout tool of WordPerfect or by using a hand markup that indicates insertions and deletions. If the area being revised is not clear, the affected portion of the page is circled. The markup techniques vary as necessary, based on the specifics of the area being changed and the complexity of the changes, to provide the clearest possible indication of the changes.

6. A marginal note (the Additional Information Number from the index) is added in the right margin of each page being changed, adjacent to the area being changed, to identify the source of each change.

7. Some changes are not applicable to one licensee but still require changes to the Tables provided in Enclosures 3A, 3B, 4, 6A, and 6B of the original license amendment request to reflect the changes being made by one or more of the other licensees. These changes are not included in the additional information for the licensee to which the change does not apply, as the changes are only for consistency, do not technically affect the request for that licensee, and are being provided in the additional information being provided by the licensees for which the change is applicable. The complete set of changes for the license amendment request will be provided in a licensing amendment request supplement to be provided later.

JOINT LICENSING SUBCOMMITTEE METHODOLOGY FOR PROVIDING ADDITIONAL INFORMATION
(cont)

8. The item numbers are formatted as follows:

   [Source] [ITS Section]-[nnn]

   Source =
   - Q - NRC Question
   - CA - AmerenUE
   - DC - PG&E
   - WC - WCNOC
   - CP - TU Electric
   - TR - Traveler

   ITS Section = The ITS section associated with the item (e.g., 3.3). If all sections are potentially impacted by a broad change or set of changes, "ALL" is used for the section number.

   nnn = a three digit sequential number

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.9.G-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST:

ITS 3.9.x Bases

General

There have been a number of instances that the specific changes to the STS Bases are not properly identified with redline or strikeout marks.

Comment: Perform an audit of all STS Bases markups and identify instances where additions and/or deletions of Bases were not properly identified in the original submittal.

FLOG response: T he submitted ITS Bases markups for Section 3.9 have been compared to the STS Bases. Some differences that were identified were in accordance with the markup methodologies (e.g., deletion of brackets and reviewer's notes). Most of the differences were editorial in nature and would not have affected the review. Examples of editorial changes are:

1) Capitalizing a letter with only a "redline" but not striking out the lower case letter that it replaced.
2) Changing a verb from singular to plural by adding an "s" without "redlining" the "s."
3) Deleting instead of striking-out the A, B, C, etc., following a specification title (e.g., SR3.6.6A.7).
4) Changing a bracketed reference (in the reference section) with only a "redline" for the new reference but failing to include the strike-out of the old reference.
5) In some instances, the brackets were retained (and struck-out) but the unchanged text within the brackets was not redlined.
6) Not redlining a title of a bracketed section. The methodology calls for the section title to be redlined when an entire section was bracketed.
7) Additional text not contained in the STS Bases was added to the ITS Bases by the lead FLOG member during the development of the submittal. Once it was determined to not be applicable, the text was then struck-out and remains in the ITS Bases mark-up.

Differences of the above editorial nature will not be provided as attachments to this response. The pages requiring changes that are more than editorial and are not consistent with the markup methodology are attached.

ATTACHED PAGES:

Encl. 5B B 3.9-1, B 3.9-11, B 3.9-17, B 3.9-18
NUREG-1431, Rev. 1 B 3.9-5 to B 3.9-7

STRIKEOUT

~~PLANT PROCEDURES ENSURE THE SPECIFIED BORON CONCENTRATION IS ORDER TO MAINTAIN THE OVERALL CORE REACTIVITY OF $K_{EFF} \leq 0.95$ DURING FUEL HANDLING, WITH CONTROL RODS AND FUEL ASSEMBLIES ASSUMED TO BE IN THE MOST ADVERSE CONFIGURATION (LEAST NEGATIVE REACTIVITY) ALLOWED BY PLANT PROCEDURES~~

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and when connected, the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling. Q3.9-1

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Q3.9.6-1 The refueling boron concentration is sufficient to maintain shutdown margin (SDM) with the most adverse conditions of fuel assembly and control rod position allowed by plant procedures. The boron concentration that is maintained in Mode 6 is sufficient to maintain $k_{eff} \leq 0.95$ with the most reactive rod control assembly completely removed from its fuel assembly.

RED LINE

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the principle system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with refueling grade borated water from the liquid hold up tanks or the refueling water storage tank ~~through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps~~.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level") to provide forced circulation cooling in the RCS and assist in maintaining the boron concentrations uniformity in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

(Continued)

BASES

DELETE

SURVEILLANCE REQUIREMENTS (continued)

performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is ~~(continued)~~ in accordance with the In service Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment. Q 3.9.G-1

REFERENCES

1. ~~GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.~~
   Design Criteria Memorandum T-16, Containment Functions.
2. FSAR, Section ~~[15.4.5]~~.
3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

BASES

An operable RHR loop must be capable of being realigned to provide an operable flow path.

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. ~~One or both RHR pumps maybe aligned to the RWST to support filling the refueling cavity or for performance of required testing (Ref. 2)~~.

Q3.9.6-1 Redline→

TR 3.9-001

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level." A Note is added to the applicability to assure that MODE 6 operation with water level <23 ft is not permitted unless two RHR loops are operable.

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR 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 RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. ~~Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS, because all of the unborated water sources are isolated~~. The suspension of any operation involving a reduction in Reactor Coolant Boron Concentration will reduce the likelihood of Boron Stratification in the RCS. DC-ALL-002

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop

(Continued)

BASES

requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate of more than 3000 gpm is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core prior to 57 hours subcritical. The second part of this Surveillance serves the same function but with 57 hours or more of core subcriticality and provides a reduced flow rate of 1300 gpm based upon a reduced decay heat load. Both of these flow rates are points of the same flow rate verses decay heat curve. The 1300 gpm limit also precludes exceeding the 1675 gpm upper flow limit to prevent vortexing and air entrainment of the RHR piping system. RHR pump vortexing (failure to meet pump suction requirements) during mid-loop operation may result in RHR pump failure and non-conservative RCS level indication. ~~In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met.~~ The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room ~~(Ref. 3)~~ 2. TR 3.9-001

SR 3.9.6.2 Redline Q3.9G-1

Verification that the required pump is OPERABLE ensures that an additional ~~RCS or~~ RHR pump 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. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. FSAR, Section ~~(~~ 5.5.7 ~~).~~ 1/22/88

TR 3.9-001

~~2. NOG Standard Technical Specification Change Traveler TSTF-21.~~

2 ~~3~~ LAR 88-01, dated ~~4/21/88~~, submitted by "RHR System Flow Rate Reduction," DCL ~~88-067.~~ 88-014 DC-ALL-002

(Continued)

~~B 3.9 REFUELING OPERATIONS~~

~~B 3.9.2 Unborated Water Source Isolation Valves~~

~~BASES~~

~~BACKGROUND~~ ~~During MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position.~~

~~The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by reducing the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.~~

~~APPLICABLE SAFETY ANALYSES~~ ~~The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required for MODE 6.~~

~~The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.~~

~~LCO~~ ~~This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in SDM.~~

~~(continued)~~

BASES (continued)

~~APPLICABILITY~~ ~~In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of all sources of unborated water to the RCS.~~

~~For all other MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated.~~

~~ACTIONS~~ ~~The ACTIONS table has been modified by a Note that allows separate Condition entry for each unborated water source isolation valve.~~

~~A.1~~

~~Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.~~

~~Condition A has been modified by a Note to require that Required Action A.3 be completed whenever Condition A is entered.~~

~~A.2~~

~~Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position, ensures that the valves cannot be inadvertently opened. The Completion Time of "immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.~~

~~(continued)~~

BASES

~~ACTIONS (continued)~~ ~~A.3~~

~~Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed whenever Condition A is entered to demonstrate that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.~~

~~SURVEILLANCE REQUIREMENTS~~ ~~SR 3.9.2.1~~

~~These valves are to be secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water sources are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. This Surveillance demonstrates that the valves are closed through a system walkdown. The 31 day Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.~~

~~REFERENCES~~

~~1. FSAR, Section [15.2.4].~~

~~2. NUREG-0800, Section 15.4.6.~~

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-i **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

General

A great majority of the DOCs state that the reasons for the relocation and for the proposed changes including deletions, additions, and revisions, are made to be consistent with NUREG-1431. While this is a valid statement, additional justifications are still required in order to support the proposed relocations and the CTS changes. The DOCs should be expanded to include additional justifications for the relocation and/or changes.

Comment: Revise those DOCS which do not provide technical justifications for the proposed relocations and changes, and indicate which DOCs are being revised under this comment.

FLOG RESPONSE: The statement that the changes are made to be consistent with NUREG-1431 in a number of the DOCs was not intended to be a justification for the proposed change but an indication that changes were being made to make the Technical Specifications similar to NUREG-1431. The DOCs were developed (specifically the Less Restrictive DOCs) with the intent that the No Significant Hazards Consideration would contain more detail justifying the change. The conversion license amendment application was developed using as a guide the Vogtle application for determining the level of detail needed for the DOCs. During the development of the conversion license amendment application in late 1996, several issues were identified with impact on the conversion process including "literal compliance," Generic Letter 96-01, and feedback from NRC concerning the acceptability of the submittals made by some other licensees. On January 24, 1997, senior managers from the FLOG met with the NRC (the Technical Specification Branch and Project Management) to discuss these and other issues. The utilities took additional time to review the conversion license amendment application to make sure that these issues were being properly addressed.

On June 25, 1998, a discussion was held with the NRC staff, in which it was believed that comments provided in Section 3.9 addressed those DOCs and JFDs that required additional justification. The FLOG has agreed to revise those DOCs, as identified by the NRC, by either bringing forward information from the No Significant Hazards Consideration (Enclosure 4) or providing additional justification as requested. No additional DOCs or JFDs have been revised unless indicated in the response to a specific Comment Number.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-1 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

CTS 3.9.1
DOC 1-01-A
ITS 3.9.1, LCO 3.9.1
JFD 3.9-15

a. (Comanche Peak, Callaway, and Wolf Creek)

The CTS and ITS are proposed to be revised by adding "when connected" preceding "Reactor Coolant System." The DOC provides a generic explanation, but it does not provide any specific technical justification for this addition. This revision is considered an administrative enhancement and a generic change to the ITS. Therefore, it must be reviewed and approved via the TSTF process before it may be adopted as the standard ITS language. Furthermore, Diablo Canyon does not include the proposed addition, "when connected,' in its CTS markup.

Comment: Either remove this item from the submittal and adopt the ITS language, or submit a TSTF for this generic change. Also, provide explanation why Diablo Canyon is not adopting the proposed language, "when connected."

FLOG Response: The proposed changes to CTS 3.9.1 and ITS 3.9.1 were based on traveler WOG-103, Revision 1. WOG-103, Revision 1, has recently been designated TSTF-272 and transmitted to the NRC in May 1998. The proposed wording in TSTF-272 was modified from WOG-103, Revision 1, and these modifications have been incorporated into the ITS and ITS Bases.

During preparation of the conversion license amendment request, WOG-103, Revision 1, was inadvertently omitted by Diablo Canyon. Diablo Canyon will incorporate TSTF-272 into the ITS and ITS Bases.

ATTACHED PAGES:

Encl. 2	3/4 9-1
Encl. 3A	1
Encl. 3B	1
Encl. 5A	Traveler Status page, 3.9-1
Encl. 5B	B 3.9-1, B 3.9-2, B 3.9-3
Encl. 6A	2
Encl. 6B	2

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

(Q 3.9-1a)

3.9.1 The boron concentration ~~of all filled portions~~ (Connected portions of) of the Reactor Coolant System ~~and~~ the refueling canal, and the refueling cavity shall be maintained ~~uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met either:~~ within the limits specified in the COLR. — ED, 01-01-A, 01-02-LG

~~a. A K_{eff} of 0.95 or less, which includes a 1% Δk/k conservative allowance for uncertainties, or~~ — 01-04-LG

~~b. A boron concentration of greater than or equal to 2000 ppm, which includes a 50 ppm conservative allowance for uncertainties.~~ — 01-02-LG

APPLICABILITY: MODE 6*. — 01-05-A

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate ~~and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2,000 ppm, whichever is the more restrictive.~~ action to restore boron concentration to within limits. — 01-06-LS1

SURVEILLANCE REQUIREMENTS

~~4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:~~ — 01-09-LS2

~~a. Removing or unbolting the reactor vessel head, and~~

~~b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.~~

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal and the refueling cavity shall be determined ~~by chemical analysis~~ to be within the limit specified in the COLR at least once ~~each~~ per 72 hours. — 01-10-LG, 01-02-LG

~~*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.~~ — 01-05-A

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[ ]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

These changes are consistent with current practice and current TS Bases, therefore, they have no technical impact and are administrative.

CHANGE NUMBER	NSHC	DESCRIPTION
01-01	A	Changes were made to add ~~Adds~~ the filled portion of the refueling cavity to the locations in which the boron concentration must be maintained and specifies that the concentration must be maintained in locations connected to the RCS. (Q3.9-1a)
01-02	LG	Specifies that the required limits for the boron concentration will be moved to the Core Operating Limits Report, in accordance with NUREG-1431. This change removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.
01-03	LG	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-04	LG	The specified limit on $k_{eff} \leq 0.95$ is moved to the Bases; however, the limit is effectively maintained by the requirement to keep boron concentration within limits which remains in the LCO. As noted in 1-02-LG above, the boron concentration limit will be maintained in the COLR. This change removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.
01-05	A	The footnote defining the "REFUELING" condition is not necessary because it duplicates the definition of MODE 6 in ITS Table 1.1-1. This change does not result in a change to technical requirements and is consistent with NUREG-1431.
01-06	LS1	The requirements to initiate boration at a specified flow rate having a specified boron concentration is replaced by the more general requirement to initiate boration to restore the required boron concentration. The reactor operators are expected to select the best method of increasing the boron concentration to the required value specified in the COLR. The proposed change clarifies that action is applicable only to restoring boron concentration to within limit. This change is acceptable because it is an example of removing procedural details while maintaining the actual limiting condition as a TS requirement.
01-07	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-08	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 A	Adds the filled portion of the refueling cavity to the locations in which the boron concentration must be maintained. and specifies that the concentration must be maintained in locations connected to the RCS	Yes	Yes	Yes	Yes Q3.9-1a
01-02 LG	Specifies that the required limits for the boron concentration will be in the Core Operating Limits Report (COLR). In addition, the provision to maintain a uniform concentration is discussed in the ITS Bases.	Yes	Yes	Yes	Yes
01-03 LG	Instead of providing the tag numbers of the valves used to isolate unborated water sources, the function of the valves is used. The valve tag numbers are moved to the Bases.	No, Current Technical Specifications (CTS) based upon licensed dilution accident.	Yes	Yes No, maintaining valve numbers in CTS.	No, Maintaining valve numbers in CTS WC 3.9-001
01-04 LG	The specified limit on $k_{eff} \leq 0.95$ is moved to the Bases.	Yes	Yes	Yes	Yes
01-05 A	The footnote defining the "REFUELING" condition is not necessary because it duplicates the definition of MODE 6.	Yes	No, not in CTS	Yes	Yes
01-06 LS1	The requirements to initiate boration at a specified flow rate having a specified boron concentration is replaced by the more general requirement to initiate boration to restore the required boron concentration. Additionally, the ACTION statement is revised to clarify that action is applicable only to boron concentration.	Yes	Yes	Yes	Yes
01-07 M	A new ACTION statement is incorporated that specifies the appropriate activities if the isolation valves for unborated water sources are not secured in the closed position.	No, CTS based upon Licensed Dilution Accident.	No	Yes	Yes
01-08 M	Separate entry into the ACTION is allowed for each unborated water source isolation valve.	No, CTS based upon Licensed Dilution Accident.	Yes	Yes	Yes

Industry Travelers Applicable to Section 3.9

Approved by NRC.

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-20	Incorporated	3.9-2	Approved by NRC
TSTF-21 Rev. 1	Incorporated Not	None	Change made to Bases for 3.9.6 TR 3.9-001
TSTF-22	Not incorporated	N/A	Changes not applicable for the specific plant application.
TSTF-23, Rev. 2 Rev. 3	Incorporated	3.9-3 Approved by NRC.	Traveler bracketed ITS 3.9.2 and revised the Bases for ITS 3.9.3 (DCPP maintaining CTS). TR 3.9-003
TSTF-51	Not incorporated	N/A	Minimal impact on plant specific applications.
TSTF-68, Rev. 1	Not incorporated	N/A	Similar changes were incorporated into the ITS based on current licensing basis. See change description 3.9-1. (Not Appl to DCPP)
TSTF-92, Rev. 1	Not incorporated	N/A	The proposed changes did not significantly affect current surveillance practices to warrant inclusion.
TSTF-96, Rev. 1	Incorporated	3.9-4	Approved by NRC. TR 3.9-003
WOG-63	Not incorporated	N/A	
WOG-76	Incorporated	3.9-11	Containment penetrations allowed to be open under administrative control.
TSTF-136	Incorporated	N/A	Approved by NRC
TSTF-139	Incorporated	N/A	
TSTF-153	Incorporated	N/A	
TSTF-272	Incorporated	3.9-15	Q 3.9-1a

TR 3.9-002

Editorial change to 3.9.5 unnecessary where as 3.4 and 3.5 were revised to match 3.9.5 wording.

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

Q3.9-1a

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, and, when connected the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

NOTE
While this LCO is not met, entry into MODE 6 from MODE 5 is not permitted.

3.9-14

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in COLR.	72 hours

strikeout

~~PLANT PROCEDURES ENSURE THE SPECIFIED BORON CONCENTRATION IN ORDER TO MAINTAIN THE OVERALL CORE REACTIVITY OF KEFF ≤ 0.95 DURING FUEL HANDLING, WITH CONTROL RODS AND FUEL ASSEMBLIES ASSUMED TO BE IN THE MOST ADVERSE CONFIGURATION (LEAST NEGATIVE REACTIVITY) ALLOWED BY PLANT PROCEDURES~~

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and when connected, the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling. Q3.9-1

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Q3.9.G-1 The refueling boron concentration is sufficient to maintain shutdown margin (SDM) with the most adverse conditions of fuel assembly and control rod position allowed by plant procedures. The boron concentration that is maintained in Mode 6 is sufficient to maintain $k_{eff} \leq 0.95$ with the most reactive rod control assembly completely removed from its fuel assembly.

RED LINE

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the principle system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with refueling grade borated water from the liquid hold up tanks or the refueling water storage tank ~~through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps~~.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level") to provide forced circulation cooling in the RCS and assist in maintaining the boron concentrations uniformity in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

(Continued)

BASES

APPLICABLE SAFETY ANALYSIS

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

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

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

The limiting boron dilution accident analyzed occurs in MODE 5 ~~3~~ (Ref. 2). ~~A detailed discussion of this event is provided in Bases B 3.1.2, "SHUTDOWN MARGIN (SDM) — T_{avg} ≤ 200°F." Boron dilution accidents are precluded in MODE 6 by isolating potential dilution flow paths. See LCO 3.9.2, "Unborated Water Source Isolation Valves." It is based upon a maximum dilution flow of 300 g.p.m. and prompt identification and operation preclude the event from proceeding to a boron dilution accident. Prompt identification is assured through audible count rate instrumentation, a ~~high~~ count rate ~~alarm~~ and a high source range flux level alarm.~~

visual instrumentation (DC 3.9-ED)

The RCS boron concentration satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii).

(THE SAME MINIMUM BORON CONCENTRATION, IS REQUIRED TO BE MAINTAINED) (Q3.9-1a)

LCO

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

(WHILE IN MODE 6. ADDITIONALLY, WHEN THE RCS IS FLOODED AND OPEN TO) (Q3.9-1a)

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a k_{eff} ≤ 0.95. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~— T_{avg} > 200°F," and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) — T_{avg} ≤ 200°F,"~~ LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical. A Note is added to the applicability to assure that MODE 6 cannot be entered unless boron concentration limits are met.

(Continued)

BASES

Q 3.9-1

ACTIONS

A.1 and A.2

AND WHEN CONNECTED,

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

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

A.3

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

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

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

WHEN CONNECTED (I.E., HAVING DIRECT ACCESS TO THE REACTOR VESSEL) ALL THE FILLED PORTIONS OF

THE FILLED PORTIONS OF

Q 3.9-1c

This SR ensures that the coolant boron concentration in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration of the coolant in each REQUIRED volume is determined periodically by chemical analysis.

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. ~~USAR Chapter [15]~~ FSAR, Chapter 15, Section 15.2.4

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.9

CHANGE NUMBER	JUSTIFICATION
3.9-10	Q 3.9-23 ~~Consistent with the CTS, ITS 3.9.7 Applicability is being revised to delete "During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts." The Applicability is being revised because this requirement is duplicative of a relocated technical specification requirement for reactor vessel water level during movement of control rods (relocated technical specification 3.9.10.2). The relocated specification places a lower limit on the amount of water above the top of the fuel assemblies in the reactor vessel during movement of control rods. The Bases (CTS 3/4.9.10) states that this ensures the water removes 99% of the assured 10% iodine gap activity released from the rupture of an irradiated fuel assembly in the event of a fuel handling accident (FHA). However, the movement of control rods is not associated with the initial conditions of a FHA, and the Bases do not address any concern regarding inadvertent criticality which could lead to a breach of the fuel rod cladding.~~
Insert →	
3.9-11	In accordance with a proposed traveler WOG-76, LCO is modified to permit a penetration flow path that provides direct access from the containment to the outside atmosphere to be unisolated under administrative controls. The allowance to have containment flow paths with direct access from the containment atmosphere to the outside atmosphere unisolated under administrative controls is based on confirmatory dose calculations of a fuel handling accident which indicate acceptable radiological consequences and to implement administrative controls that ensure that the flow penetrations will be promptly closed following a fuel handling accident, to provide a defense-in-depth approach to meet acceptable dose consequences. The administrative control requirements are defined in the Bases.
3.9-12	A note is added to the Applicability of LCO 3.9.6 indicating that entry into a MODE or other specified condition in the Applicability is not permitted while the LCO is not met. The addition of this note is based on the performance of a plant-specific LCO 3.0.4 matrix which identified where the requirements of 3.0.4 are still applicable in MODES 5 and 6 and in MODES 1, 2, 3, and 4 when the MODE is descending (i.e., from MODE 1 to MODE 2, etc.). This matrix was specified in the NUREG-1431 reviewer's note in LCO 3.0.4.
3.9-13	In accordance with DCPP CTS, LCO 3.9.2 would not be used. This new requirement is not applicable to DCPP which has a licensed dilution accident. The current licensing bases in accordance with NUREG 0800, Section 15.4.6 provides adequate assurance that a dilution event will recognized and arrested in a timely fashion
3.9-14	A Note is added to the Applicability of LCO 3.9.1 indicating that entry into MODE 6 from MODE 5 is not permitted while the LCO is not met. The addition of this Note is based on the performance of a plant-specific LCO 3.0.4 matrix which identified where the requirements of 3.0.4 are still applicable in MODES 5 and 6 and in MODES 1, 2, 3, and 4 when the MODE is descending (i.e., from MODE 1 to MODE 2, etc.). This matrix was specified in the NUREG-1431 reviewer's note in LCO 3.0.4. Insert Q3.9-24
3.9-15 Insert	Q 3.9-1a

JFD 3.9-15

LCO 3.9.1 has been revised in accordance with traveler TSTF-272 to clarify that boron concentration limits do not apply to the refueling [cavity and refueling canal] or other flooded areas when these areas are not connected to the RCS. This change is acceptable because the boron concentration limit is intended to ensure that the reactor remains subcritical in MODE 6. However, when areas containing boron solution are isolated from the RCS, no potential for boron dilution exists. Therefore, there is no need to place a limit on boron concentration in these areas when they are not connected to the RCS. This change is consistent with the intent of the Specification, as described in the Bases, and eliminates restrictions that have no effect on safety.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.9-8	The Note of ITS 3.9.5 is expanded to incorporate CTS allowing the required RHR pump to be removed from service for less than or equal to 2 hours per 8 hours for leak testing of the RHR suction isolation valves.	Yes	No, not in CTS	No, not in CTS	No, not in CTS
3.9-9	The Surveillances of ITS 3.9.5 and 3.9.6 are modified to incorporate CTS of two RHR flow rates dependent upon the number of hours the reactor has been subcritical.	Yes, See LAR 88-01 dated 4/21/88 and DCL 88-067	No, not in CTS	No, not in CTS	No, not in CTS
3.9-10	ITS 3.9.7 Applicability is being revised to delete "During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts." ~~The Applicability is being revised because this requirement is duplicative of a relocated technical specification requirement for reactor vessel water level during movement of control rods.~~	Yes	Yes	Yes, ~~relocated per Amendment 89~~	Yes, ~~relocated per Amendment 103~~ Q3.9-23
3.9-11	In accordance with traveler WOG-76, LCO is modified to permit a penetration flow path that provides direct access from the containment to the outside atmosphere to be unisolated under administrative controls.	Yes	Yes	Yes	Yes
3.9-12	A Note is added to the Applicability of LCO 3.9.6 indicating that entry into a MODE or other specified condition in the Applicability is not permitted while the LCO is not met.	Yes	Yes	Yes	Yes
3.9-13	In accordance with DCPP CTS, LCO 3.9.2 would not be used	Yes	No	No	No
3.9-14	A Note is added to the Applicability of LCO 3.9.1 indicating that entry into MODE 6 from MODE 5 is not permitted while the LCO is not met.	Yes	Yes	Yes	Yes
3.9-15	Insert				

Q 3.9-1a

JFD 3.9-15

LCO 3.9.1 has been revised to clarify that boron concentration limits do not apply to flooded areas that are not connected to the RCS.

Applicability:

DC	Yes
CP	Yes
WC	Yes
CA	Yes

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-4 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

CTS 4.9.2 b and c
CTS 4.9.2 b, c and Footnote * (Diablo Canyon)
DOC 2-03-LS3

Surveillance requirements 4.9.2 b and c for Analog Channel Operational Test are proposed to be deleted in CTS to be consistent with NUREG-1431, Rev. 1. ITS does not include these requirements. DOC 2-03-LS3 discusses the reasons for deletion, but it does not address the associated impact in regard to plant operation and design basis, and whether these surveillances would be moved to plant procedures or relocated to the UFSAR.

Comment: Revise DOC to justify as to why this is acceptable based on licensing and design basis. If these SRs should be relocated, identify the plant document that includes the CTS requirements.

FLOG RESPONSE: Additional information supporting this DOC is in NSHC LS-3 in Enclosure 4. As discussed with the NRC technical specification branch reviewers on June 25, 1998, DOC 2-03-LS-3 is revised to include the following information:

"During REFUELING, the source range monitors provide visual [and audible] indication of neutron count rate to plant operators. Core reactivity is maintained primarily by the requirements of ITS 3.9.1 [ ] which assure that the boron concentration in refueling water is within limit and that dilution of the boron will [be identified promptly]. Thus, the neutron monitoring channels provide further assurance that criticality will not occur. The proposed deletion of ACOTs for these channels would be offset by the CHANNEL CHECK and CHANNEL CALIBRATION requirements. The addition of a CHANNEL CALIBRATION to be performed every 18 months provides assurance that the instruments can provide the visual indication. There are no alarms, interlocks, or trip setpoints associated with these channels that are required to be OPERABLE during MODE 6. In addition, in MODE 6 the source range instruments provide no automatic actuation function used for mitigation of accidents, and they would have no effect on the outcome of an accident. Furthermore, the modification of SRs for these indicators does not imply that they will be unavailable when required. The CHANNEL CHECKS and CHANNEL CALIBRATION SR that remain in effect provide the necessary assurance of OPERABILITY."

ATTACHED PAGES:

Encl 3A	2
Encl 4	18, 19

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
01-09	LS2	The SR to verify reactivity conditions is deleted, as it is generally descriptive of the MODE 6 conditions, as defined in NUREG-1431, and is addressed by SR 4.9.1.2. This change is acceptable because the boron concentration is required to be within limit prior to entry into MODE 6 in accordance with the Applicability Note for ITS 3.9.1. Thus, the deleted SR is redundant to other requirements that remain in TS.
01-10	LG	Moves the description in the SR to determine the boron concentration by chemical analysis to the Bases. This change is consistent with NUREG-1431, and removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.
01-11	LS19	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-12	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-13	A	Q3.9-16
02-01	LS21	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-02	M	The ACTION statement is revised to require that restoration of one monitor is immediately initiated. This change adds a more stringent TS requirement which is appropriate and consistent with NUREG-1431.
02-03	LS3	The ANALOG CHANNEL OPERATIONAL TEST (ACOT) requirements are deleted and a Channel Calibration is added, in accordance with NUREG-1431. In Mode 6, the source range monitors are required for indication only and there are no precise set points associated with these instruments. In this capacity, the source range instrumentation is typically used to read a relative change in count rate. The source range instrumentation is monitored for significant changes in count rate which are important to evaluate the change in core status. Even the accepted convention defining criticality only requires a slowly increasing count rate be verified. Consistent with NUREG-1431, indicating instruments only require channel checks and channel calibrations. The more frequent ACOTs are applied only to those channels with operational interlocks or other set point actuations. Therefore, the MODE 6 channel checks and channel calibration requirements for the source range monitors are adequate to assure their operability, considering the more frequent ACOTs performed on this instrumentation in other Modes, the effectiveness of these surveillance requirements in maintaining other indicating instruments operable, and the accuracy required of these instruments in MODE 6.
	Insert →	Q3.9-4
02-04	LG	Consistent with NUREG-1431, the requirements related to indication provided by the source range detectors would be moved to the Bases.

DOC 02-03-LS3

During REFUELING, the source range monitors provide visual [and audible] indication of neutron count rate to plant operators. Core reactivity is maintained primarily by the requirements of ITS 3.9.1 [ ] which assure that the boron concentration in refueling water is within limit and that dilution of the boron will [be identified promptly]. Thus the neutron monitoring channels provide further assurance that criticality will not occur. The proposed deletion of ACOTs for these channels would be offset by the CHANNEL CHECK and CHANNEL CALIBRATION requirements. The addition of a CHANNEL CALIBRATION to be performed every 18 months provides assurance that the instruments can provide the visual indication. There are no alarms, interlocks, or trip setpoints associated with these channels that are required to be OPERABLE during MODE 6. In addition, in MODE 6 the source range instruments provide no automatic actuation function used for mitigation of accidents, and they would have no effect on the outcome of an accident. Furthermore, the modification of SRs for these indicators does not imply that they will be unavailable when required. The CHANNEL CHECKS and CHANNEL CALIBRATION SR that remain in effect provide the necessary assurance of OPERABILITY.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, the requirements of SR 4.9.2 b. and c. to perform ANALOG CHANNEL OPERATIONAL TESTS (ACOTs) would be deleted and a SR to perform a Channel Calibration is added, in accordance with NUREG-1431. In MODE 6, the source range monitors are required for indication only and there are no precise setpoints associated with these instruments. In this capacity, the source range instrumentation is typically used to read a relative change in count rate. The source range instrumentation is monitored for significant changes in count rate which are important to evaluate the change in core status. The accepted convention defining criticality does not require precise or specific set points or indication, but only requires verification of a slowly increasing count rate. Consistent with NUREG-1431, indicating instruments only require channel checks and channel calibrations. The more frequent ACOTs are applied only to those channels with operational interlocks or other setpoint actuations. Therefore, the MODE 6 channel checks and channel calibration requirements for the source range monitors are adequate to assure their operability considering the accuracy required of these instruments in MODE 6.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> *1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
>
> *2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
>
> *3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Q3.9-4 During REFUELING, the source range monitors provide visual and audible indication of neutron count rate to plant operators. The proposed deletion of ACOTs for these channels would be offset by the CHANNEL CHECK and CHANNEL CALIBRATION requirements. The addition of a CHANNEL CALIBRATION to be performed every 18 months provides assurance that the instruments can provide Q3.9-4 the visual and audible indication. There are no alarms, interlocks, or trip setpoints associated with these channels that are required to be OPERABLE during MODE 6. Thus, the proposed change would have no effect on the probability of an accident occurring. In addition, in Mode 6 the source range instruments provide no automatic actuation function used for mitigation of accidents, and they would have no effect on the outcome of an accident. Furthermore, the modification of SRs for these indicators does not imply that they will be unavailable when required. The CHANNEL CHECKS and CHANNEL CALIBRATION SR that remain in effect provide the necessary assurance of OPERABILITY. Therefore, there would be no increase in the probability or consequences of a previously evaluated accident as a result of making the proposed change.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3
(Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change would not create the possibility of a new or different accident, because the plant or its systems would not be operated any differently. The proposed change has to do with the type of SR applied to source range instrument channels. Therefore, there would be no operational changes to contribute to the possibility of a new accident resulting from the proposed change. Therefore, this change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety? and that dilution of Boron will [be identified promptly] Q3.9-4

   The margins of safety in question are those involved with preventing criticality during REFUELING operations. The monitors provide visual and audible indication of neutron count rate, and, therefore, provide assurance that the core reactivity is being maintained. [ ] However, reactivity is maintained primarily by the requirements of ITS 3.9.1 which assure that the boron concentration in refueling water is within limit. Thus the neutron monitoring channels provide further assurance that criticality will not occur. In addition, deletion of certain channel tests (ACOTs) would not prevent the channels from performing when required. The CHANNEL CHECK and CHANNEL CALIBRATION SRs for this equipment are appropriate for the function of the source range channels during REFUELING. Therefore, the deletion of ACOTs for the source range neutron monitoring channels during MODE 6 would have an insignificant effect on margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS3" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-5 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

CTS 3/4.9.3
DOC 3-01-R

The CTS requirements in 3/4.9.3 have been entirely relocated to an unspecified licensee controlled document. Though Conversion Comparison Table provides the new location of this item, it is still necessary to address where the CTS requirements have been relocated to in the DOC. In addition, the specific technical justification for the relocation is not addressed in the DOC.

Comment: Revise DOC by providing justification as to why the relocation is acceptable and identify the licensee controlled document to which the CTS requirements would be relocated.

FLOG RESPONSE: DOC 3-01-R has been revised to provide justification as to why the relocation is acceptable. The justification shows that this LCO ensures the decay of short lived fission products prior handling irradiated fuel. The associated decay time is not an installed instrument nor is it used to detect an abnormal degraded condition. This decay time is an initial condition of a DBA which assumes the failure of fission product barrier. The industry/NRC did, however, agree during development of NUREG-1431 that this LCO could be relocated and only included it in the Bases of NUREG-1431 (B 3.9.7). The decay time is not a structure, system, or component of the primary success path to mitigate a DBA.

The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specifications branch reviewers during telephone calls on June 25 and June 30, 1998.

ATTACHED PAGES:

Encl 3A	3
Encl 3B	2

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
3-01	R	Consistent with NUREG-1431, the subcriticality requirement prior to irradiated fuel movement is relocated to a licensee controlled document. This change is acceptable based on the schedule requirements following shutdown to attain plant conditions for movement of irradiated fuel. These schedule requirements provide assurance that the requirements of the decay time LCO would not be exceeded.
04-01	Insert → LG	Q 3.9-5 This change removes the word "automatic" from the requirement that each penetration be capable of being closed by an OPERABLE automatic containment purge isolation valve. The requirement for an automatic valve would be stated in the Bases. This change is consistent with NUREG-1431, and removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.
04-02	LS4	Removes Surveillance requirement to perform verification within 100 hours prior to the start of core alteration or movement of irradiated fuel. This is consistent with NUREG-1431, and is acceptable because the deleted requirement is redundant with the requirement to meet the LCO at the time that CORE ALTERATIONS or fuel movement begins.
04-03	LS5	Q 3.9-8 The frequency of verifying that the [Containment Purge and Exhaust isolation] occurs is changed from 7 days to 18 months. This is consistent with NUREG-1431. This change is acceptable because the revised frequency requirement will continue to assure the OPERABILITY of the valves. Insert The new frequency is consistent with those SRs applicable to ESFAS-type functions and in service valve testing which are appropriate for the containment isolation function.
04-04	TR1	Revised Surveillance requirement to allow for increased flexibility in using an actual or simulated actuation signal. Identification of specific signals is moved to the Bases.
04-05	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-06	LS23	Not applicable to DCPP: See Conversion Comparison Table (Enclosure 3B).
04-07	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-08	LG	Relocate the references to Heavy Loads in the Applicability and ACTION section of LCO 3.9.4 to the FSAR.
04-09	LS14	LCO 3.9.4 would be modified to permit an approved functional equivalent of a valve or blind flange to isolate containment penetrations. This is consistent with NUREG-1431.

DOC 03-01-R

Consistent with NUREG-1431, Rev. 1, the subcriticality requirement prior to irradiated fuel movement is relocated to a licensee controlled document. This CTS requirement ensured that sufficient time had elapsed to allow the radioactive decay of short-lived fission products prior to movement of irradiated fuel. This change is acceptable based on the schedule requirements following shutdown to attain plant conditions for movement of irradiated fuel. These schedule requirements provide assurance that the requirements of the decay time LCO would not be exceeded.

This proposed TS revision relocates requirements, which do not meet the TS criteria in 10CFR50.36(c)(2)(ii), to documents with established control programs. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS. An evaluation of the applicability of these criteria to this specification is provided in Attachment 21.

To ensure an appropriate level of control, these requirements will be relocated to 1) documents that are subject to the provisions of 10 CFR 50.59, 2) other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a), or 3) to programs that are controlled via the Administrative Controls section of the improved TS. The identification of the specific licensee controlled document containing this requirement is provided in Enclosure 3B of the conversion submittal.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-09 LS2	The SR to verify reactivity conditions is deleted.	Yes	Yes	Yes	Yes
01-10 LG	Moves the description in the SR to determine the boron concentration by chemical analysis to the Bases.	Yes	Yes	Yes	Yes
01-11 LS19	The time required to verify that the boron concentration is within its limits has been relaxed from 1 hour to 4 hours.	No, CTS based upon Licensed Dilution Accident.	Yes	No, not in CTS	No, not in CTS
01-12 LG	Generalizes the requirement to verify the dilution isolation valves are closed by mechanical stops or removal of motive power.	No, CTS based upon Licensed Dilution Accident.	Yes	No, not in CTS	No, not in CTS
02-01 LS21	The requirements related to indication provided by the source range detectors would be deleted from the LCO.	No, See 02-04-LG	Yes	Yes	Yes
02-02 M	The ACTION statement is revised to require that restoration of one monitor is immediately initiated.	Yes	Yes	Yes	Yes
02-03 LS3	The ANALOG CHANNEL OPERATIONAL TEST requirements are deleted and a channel calibration is added.	Yes	Yes	Yes	Yes
02-04 LG	The OPERABILITY requirements for the source range detectors in MODE 6 are moved to the Bases.	Yes	No, see 02-01-LS21	No, see 02-01-LS21	No, see 02-01-LS21
03-01 R	The subcriticality requirement prior to irradiated fuel movement is relocated to a licensee controlled document.	Yes, See attachment 21 page 17 – (relocated to an ECG)	Yes, relocated to TRM	Yes, relocated to USAR CH 16	Yes, relocated to FSAR CH 16 (Q3.9-5)
04-01 LG	This change removes the word "automatic" from the requirement that each penetration be capable of being closed by an OPERABLE automatic containment purge isolation valve. The requirement for an automatic valve would be stated in the Bases.	Yes	Yes	Yes	Yes

01-13 A Insert in WCNOC submittal (Q3.9-1b)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-7 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

CTS 3.9.4 c 1) Footnote ** (Comanche Peak)
CTS 3.9.4 c 1) Footnote ** and 4.9.4.1 Footnote ** (Callaway)
CTS 3.9.4.c and 4.9.4 Footnote * (Diablo Canyon)
CTS 3.9.4.c Footnote ** and 4.9.4 Footnote ** (Wolf Creek)
DOC 4-10-LS-20
ITS 3.9.4 NOTE and SR 3.9.4.1
JFD 3.9-11

In DOC 4-10-LS-20 and JFD 3.9-11, it was stated that this change is consistent with traveler WOG-76.

Comment: Revise DOC by providing the TSTF number associated with WOG-76 and when the associated TSTF was approved. If WOG-76 has not made it to the TSTF process or the TSTF has not yet been approved, remove this item from the submittal since the inclusion of this footnote will be pending on the approval of the TSTF change.

FLOG RESPONSE: WOG-76 was initiated by the WOG Mini-Group in October 1996. While we recognize that this is a generic change to the STS, the change was approved by the Westinghouse Owners Group over 18 months ago and was expected to have been approved by this time. We expect the TSTF committee to complete their review of WOG-76 in the very near future. We believe the technical merits of the change are consistent with traveler TSTF-68 which should justify rapid approval by the NRC. This traveler is of sufficient value in precluding confusion, LERs, and inspection findings that should we be required to remove it from our submittal, an LAR would be submitted upon NRC approval of the TSTF. We believe that it would be cost effective for all concerned to retain this change within the submittal pending NRC review of the proposed traveler.

Additional information supporting this DOC is in NSHC LS-8 in Enclosure 4. DOC 9-03-LS-8 is revised to include the following information:

"A note is added to LCO 3.9.4.c and the 7-day SR to state that containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere may be open under administrative controls. The note would allow these penetrations to be unisolated during CORE ALTERATIONS and movement of irradiated fuel assemblies within containment provided that specified administrative controls were employed. The proposed Note is acceptable based on administrative controls that consist of written procedures that require designated personnel having knowledge of the open status of the valves in question and specified persons designated and readily available to isolate the open penetration in the event of a fuel handling accident. These administrative controls provide protection equivalent to that afforded by the administrative controls used to establish containment closure for a containment personnel airlock. The NRC staff has allowed changes to the requirements for airlocks that allow both doors of an airlock to be open during CORE ALTERATIONS and during movement of irradiated fuel inside containment provided that

administrative controls are in place to quickly close one door and establish containment closure.

The isolation valve, or temporary closure device, serves to limit the consequences of accidents. The proposed change would ensure the isolation valves, or functional equivalent, will perform their required containment closure function and will serve to limit the consequences of a fuel handling accident as described in the Safety Analysis Report such that the results of the analyses in the Safety Analysis Report remain bounding. In considering the consequences of a design basis fuel handling accident inside containment, the assumptions in the analyses take no credit for the containment as a barrier to prevent the postulated release of radioactivity. For events that would occur during CORE ALTERATIONS or movement of irradiated fuel assemblies, containment closure is considered a defense-in-depth boundary to prevent uncontrolled release of radioactivity."

For DCPP, a preliminary dose calculation has been completed in accordance with the Reviewer's Note added by traveler WOG-76. This calculation shows sufficient time for closure with acceptable dose consequences.

ATTACHED PAGES:

Encl 3A 4

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
04-10	LS20	Adds a footnote stating that penetration flow paths that provide direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. This change is consistent with traveler WOG-76 and with previously approved administrative controls for personnel air locks.
05-01	Insert → R	Q 3.9-7 This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
06-01	Insert → R	Q 3.9-9 This change relocates the CTS section for the Manipulator Crane to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
07-01	Insert → R	Q 3.9-10 This change relocates the current section dealing with crane travel to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
08-01	Insert → A	Q 3.9-11 This change, consistent with NUREG-1431, provides technical guidance and clarification that loading irradiated fuel assemblies in the core is the specific activity of concern that could increase the reactor decay heat load. This is not a technical change because, under these conditions, the only activity that could increase reactor decay heat load is loading irradiated fuel into the reactor vessel.
08-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS6	This change allows the removal of the RHR loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs. This allows increased flexibility for core mapping and isolation valve testing. No operations are permitted that would cause a reduction of the RCS boron concentration. This change is consistent with NUREG-1431.
08-04	Insert → A	Q 3.9-12 This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred. Insert Q3.9-13
08-05	-	Not Used
08-06	M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days. This change adds a more stringent TS requirement which is appropriate and consistent with NUREG-1431.
09-01	A	~~Most of the requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. This change does not result in a change to technical requirements and is consistent with NUREG-1431.~~ Insert Q 3.9-14

DOC4-10-LS20

A note is added to LCO 3.9.4.c and the 7-day SR to state that containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere may be open under administrative controls. The note would allow these penetrations to be unisolated during CORE ALTERATIONS and movement of irradiated fuel assemblies within containment provided that specified administrative controls were employed. The proposed Note is acceptable based on administrative controls that consist of written procedures that require designated personnel having knowledge of the open status of the valves in question and specified persons designated and readily available to isolate the open penetration in the event of a fuel handling accident. These administrative controls provide protection equivalent to that afforded by the administrative controls used to establish containment closure for a containment personnel airlock. The NRC staff has allowed changes to the requirements for airlocks that allow both doors of an airlock to be open during CORE ALTERATIONS and during movement of irradiated fuel inside containment provided that administrative controls are in place to quickly close one door and establish containment closure.

The isolation valve, or temporary closure device, serves to limit the consequences of accidents. The proposed change would ensure the isolation valves, or functional equivalent, will perform their required containment closure function and will serve to limit the consequences of a fuel handling accident as described in the Safety Analysis Report such that the results of the analyses in the Safety Analysis Report remain bounding. In considering the consequences of a design basis fuel handling accidents inside containment, the assumptions in the analysis take no credit for the containment as a barrier to prevent the postulated release of radioactivity. For events that would occur during CORE ALTERATIONS or movement of irradiated fuel assemblies, containment closure is considered a defense-in-depth boundary to prevent uncontrolled release of radioactivity."

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-8 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

CTS 4.9.4 a (Diablo Canyon and Wolf Creek)
CTS 4.9.4 a.1 (Comanche Peak)
CTS 4.9.4.1 (Callaway)
DOC 4-03-LS-5
ITS SR 3.9.4.2

The frequency to verify the occurrence of containment ventilation isolation is proposed to be changed from 7 days to 18 months. Other than the statement that this change is consistent with NUREG-1431, Rev. 1, the DOC does not address any specific technical justifications associated with this change.

Comment: Revise the DOC to include specific technical justifications for this change.

FLOG RESPONSE: DOC 4-03-LS-5 has been revised to include specific technical justifications for this change.

ATTACHED PAGES:

Encl 3A 3

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
3-01	R	Consistent with NUREG-1431, the subcriticality requirement prior to irradiated fuel movement is relocated to a licensee controlled document. This change is acceptable based on the schedule requirements following shutdown to attain plant conditions for movement of irradiated fuel. These schedule requirements provide assurance that the requirements of the decay time LCO would not be exceeded.
04-01	Insert → LG	Q 3.9-5 This change removes the word "automatic" from the requirement that each penetration be capable of being closed by an OPERABLE automatic containment purge isolation valve. The requirement for an automatic valve would be stated in the Bases. This change is consistent with NUREG-1431, and removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.
04-02	LS4	Removes Surveillance requirement to perform verification within 100 hours prior to the start of core alteration or movement of irradiated fuel. This is consistent with NUREG-1431, and is acceptable because the deleted requirement is redundant with the requirement to meet the LCO at the time that CORE ALTERATIONS or fuel movement begins.
04-03	LS5	Q 3.9-8 The frequency of verifying that the [Containment Purge and Exhaust isolation] occurs is changed from 7 days to 18 months. This is consistent with NUREG-1431. This change is acceptable because the revised frequency requirement will continue to assure the OPERABILITY of the valves. Insert → The new frequency is consistent with those SRs applicable to ESFAS-type functions and in service valve testing which are appropriate for the containment isolation function.
04-04	TR1	Revised Surveillance requirement to allow for increased flexibility in using an actual or simulated actuation signal. Identification of specific signals is moved to the Bases.
04-05	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-06	LS23	Not applicable to DCPP: See Conversion Comparison Table (Enclosure 3B).
04-07	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-08	LG	Relocate the references to Heavy Loads in the Applicability and ACTION section of LCO 3.9.4 to the FSAR.
04-09	LS14	LCO 3.9.4 would be modified to permit an approved functional equivalent of a valve or blind flange to isolate containment penetrations. This is consistent with NUREG-1431.

DOC 04-03-LS5 The proposed change to 18 months would apply the same frequency of testing to containment purge isolation valves as is applied to other containment isolation valves that must be OPERABLE during reactor operations. The 18-month frequency has been found adequate for the type of testing applied to instrumentation and valves that must mitigate events much more sever and much more challenging to the containment boundary (e.g., LOCA, MSLB) than the FHA."

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-9 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

CTS 3/4.9.5
DOC 5-01-R

CTS 3/4.9.5 is proposed to be relocated to an unspecified licensee controlled document. The DOC does not provide any technical justification supporting this relocation.

Comment: Revise the DOC by providing additional justification for the relocation and identify the licensee controlled document containing this requirement. This requirement shall be relocated to a licensee controlled document controlled by 10 CFR 50.59.

FLOG RESPONSE: DOC 5-01-R has been revised and a Technical Specification Screening Form for CTS 3.9.5 has been prepared to provide justification as to why the relocation is acceptable. This justification shows that this LCO provides for the ability to communicate any abnormal changes to the facility or core reactivity to the refueling bridge during CORE ALTERATIONS. This communication is not applicable to any instrumentation used to detect a significant abnormal condition. It does not apply to any initial condition of a DBA or transient analysis. It is not a part of the primary success path in mitigating a DBA or a transient.

The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specifications branch reviewers during telephone calls on June 25 and June 30, 1998. The document indicated in Enclosure 3B is controlled by 10 CFR 50.59.

ATTACHED PAGES:

Encl 3A	4
Encl 3B	4

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
04-10	LS20	Adds a footnote stating that penetration flow paths that provide direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. This change is consistent with traveler WOG-76 and with previously approved administrative controls for personnel air locks.
05-01	Insert → R	Q 3.9-7 This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
06-01	Insert → R	Q 3.9-9 This change relocates the CTS section for the Manipulator Crane to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
07-01	Insert → R	Q 3.9-10 This change relocates the current section dealing with crane travel to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
08-01	Insert → A	Q 3.9-11 This change, consistent with NUREG-1431, provides technical guidance and clarification that loading irradiated fuel assemblies in the core is the specific activity of concern that could increase the reactor decay heat load. This is not a technical change because, under these conditions, the only activity that could increase reactor decay heat load is loading irradiated fuel into the reactor vessel.
08-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS6	This change allows the removal of the RHR loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs. This allows increased flexibility for core mapping and isolation valve testing. No operations are permitted that would cause a reduction of the RCS boron concentration. This change is consistent with NUREG-1431.
08-04	Insert → A	Q 3.9-12 This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred. Insert Q 3.9-13
08-05	-	Not Used
08-06	M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days. This change adds a more stringent TS requirement which is appropriate and consistent with NUREG-1431.
09-01	A	~~Most of the requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. This change does not result in a change to technical requirements and is consistent with NUREG-1431.~~ Insert Q 3.9-14

DOC 05-01-R

This change relocates the current TS section dealing with maintaining direct communication between the control room and the refueling station to a licensee controlled document as part of the conversion of the current Technical Specification to the format and expanded Bases of the improved Standard Technical Specifications. The specification requires communication between the control room and the refueling station to ensure that any abnormal change in the facility status or core reactivity observed on the control room instrumentation can be communicated to the refueling station personnel during core alterations.

This proposed TS revision relocates requirements, which do not meet the TS criteria in 10CFR50.36(c)(2)(ii), to documents with established control programs. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS. An evaluation of the applicability of these criteria to this specification is provided in Attachment 21.

To ensure an appropriate level of control, these requirements will be relocated to 1) documents that are subject to the provisions of 10 CFR 50.59, 2) other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a), or 3) to programs that are controlled via the Administrative Controls section of the improved TS. The identification of the specific licensee controlled document containing this requirement is provided in Enclosure 3B of the conversion submittal.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-01 R	This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station.	Yes, see Attachment 21 page 19 —	Yes, relocated to TRM relocated to	No, relocated per Amendment 89 an ECG	No, relocated per Amendment 103 Q3.9-9
06-01 R	This change relocates the CTS section for the Manipulator Crane.	Yes, see Attachment 21 page 21 —	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103 Q3.9-10
07-01 R	This change relocates the CTS section dealing with crane travel.	Yes, see Attachment 21 page 23 —	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103 Q 3.9-11
08-01 A	This change provides clarification that loading irradiated fuel assemblies is the activity that could increase the reactor decay heat load.	Yes	Yes	Yes	Yes
08-02 A	This change removes a limit on Reactor Coolant System temperature.	No, not in CTS.	No, not in CTS	Yes	Yes
08-03 LS6	This change allows the removal of the Residual Heat Removal (RHR) loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs.	Yes	Yes	Yes	Yes
08-04 A	This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred.	Yes	Yes	Yes	No
08-05	Not Used	N/A	N/A	N/A	N/A
08-06 M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days.	Yes	Yes	Yes	Yes
09-01 A	The requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. Insert	Yes	No - CPSES does not have containment ventilation specs in CTS 3/4.9.	Yes	Yes Q 3.9-14

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-10 **APPLICABILITY:** DC, CP

REQUEST:

CTS 3/4.9.6
DOC 6-01-R

CTS 3/4.9.6 would be entirely relocated to an unspecified licensee controlled document. The DOC does not have sufficient justification to support the relocation.

Comment: Provide additional justification as to why this relocation is acceptable and identify the name of the licensee controlled document containing this requirement. This requirement shall be relocated to a licensee controlled document controlled by 10 CFR 50.59.

FLOG RESPONSE: DOC 6-01-R has been revised and a Technical Specification Screening Form for CTS 3.9.6 has been prepared to provide justification as to why the relocation is acceptable. This justification shows that this LCO provides the ability to lift and manipulate fuel assemblies. It is not associated with installed instrumentation used to detect a significant abnormal condition in the reactor coolant pressure boundary. It is not associated with any variable or condition which is the initial condition of a DBA or transient which may challenge a fission product boundary. The manipulator crane is not a part of the primary success path for mitigating a DBA or transient.

The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specifications branch reviewers during telephone calls on June 25 and June 30, 1998. The document indicated in Enclosure 3B is controlled by 10CFR50.59.

ATTACHED PAGES:

Encl 3A	4
Encl 3B	4

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
04-10	LS20	Adds a footnote stating that penetration flow paths that provide direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. This change is consistent with traveler WOG-76 and with previously approved administrative controls for personnel air locks.
05-01	Insert → R	This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS. Q 3.9-7
06-01	Insert → R	This change relocates the CTS section for the Manipulator Crane to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS. Q 3.9-9
07-01	Insert → R	This change relocates the current section dealing with crane travel to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS. Q 3.9-10
08-01	Insert → A	This change, consistent with NUREG-1431, provides technical guidance and clarification that loading irradiated fuel assemblies in the core is the specific activity of concern that could increase the reactor decay heat load. This is not a technical change because, under these conditions, the only activity that could increase reactor decay heat load is loading irradiated fuel into the reactor vessel. Q 3.9-11
08-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS6	This change allows the removal of the RHR loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs. This allows increased flexibility for core mapping and isolation valve testing. No operations are permitted that would cause a reduction of the RCS boron concentration. This change is consistent with NUREG-1431.
08-04	Insert → A	This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred. Q 3.9-12 Insert
08-05	-	Not Used Q 3.9-13
08-06	M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days. This change adds a more stringent TS requirement which is appropriate and consistent with NUREG-1431.
09-01	A	~~Most of the requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. This change does not result in a change to technical requirements and is consistent with NUREG-1431.~~ Insert Q 3.9-14

This change relocates the current TS section for the Refueling Machine to a licensee controlled document as part of the conversion of the current Technical Specification to the format and expanded Bases of the improved Standard Technical Specifications. The OPERABILITY requirements for the refueling machine main hoist and auxiliary monorail hoist ensure that: (1) the main hoist will be used for movement of fuel assemblies, (2) the auxiliary monorail hoist will be used for latching, unlatching and movement of control rod drive shafts, (3) the main hoist has sufficient load capacity to lift a fuel assembly (with control rods), (4) the auxiliary monorail hoist has sufficient load capacity to latch, unlatch and move the control rod drive shafts, and (5) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertantly engaged during lifting operations.

This proposed TS revision relocates requirements, which do not meet the TS criteria in 10CFR50.36(c)(2)(ii), to documents with established control programs. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS. An evaluation of the applicability of these criteria to this specification is provided in Attachment 21.

To ensure an appropriate level of control, these requirements will be relocated to 1) documents that are subject to the provisions of 10 CFR 50.59, 2) other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a), or 3) to programs that are controlled via the Administrative Controls section of the improved TS. The identification of the specific licensee controlled document containing this requirement is provided in Enclosure 3B of the conversion submittal.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-01 R	This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station.	Yes, see Attachment 21 page 19 —	Yes, relocated to TRM relocated to an ECG	No, relocated per Amendment 89	No, relocated per Amendment 103 Q3.9-9
06-01 R	This change relocates the CTS section for the Manipulator Crane.	Yes, see Attachment 21 page 21 —	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103 Q3.9-10
07-01 R	This change relocates the CTS section dealing with crane travel.	Yes, see Attachment 21 page 23 —	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103 Q 3.9-11
08-01 A	This change provides clarification that loading irradiated fuel assemblies is the activity that could increase the reactor decay heat load.	Yes	Yes	Yes	Yes
08-02 A	This change removes a limit on Reactor Coolant System temperature.	No, not in CTS.	No, not in CTS	Yes	Yes
08-03 LS6	This change allows the removal of the Residual Heat Removal (RHR) loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs.	Yes	Yes	Yes	Yes
08-04 A	This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred.	Yes	Yes	Yes	No
08-05	Not Used	N/A	N/A	N/A	N/A
08-06 M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days.	Yes	Yes	Yes	Yes
09-01 A	The requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. Insert	Yes	No - CPSES does not have containment ventilation specs in CTS 3/4.9.	Yes	Yes Q 3.9-14

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-11 **APPLICABILITY:** DC, CP

REQUEST:

CTS 3/4.9.7
DOC 7-01-R

CTS 3/4.9.7 is proposed to be relocated to an unspecified licensee controlled document. The DOC does not provide any technical justification supporting this relocation.

Comment: Provide additional justification as to why this relocation is acceptable and identify the name of the licensee controlled document containing this requirement. This requirement shall be relocated to a licensee controlled document controlled by 10 CFR 50.59.

FLOG RESPONSE: DOC 7-01-R has been revised and a Technical Specification Screening Form for CTS 3.9.7 has been prepared to provide justification as to why the relocation is acceptable. This justification shown that the fuel handling building crane travel is used to restrict loads in excess of a fuel assembly from traveling over the spent fuel pool. It is not associated with any instrumentation used to detect a significant degradation of the reactor coolant pressure boundary. While it does provide a restriction used to prevent a heavy load drop event, it does not apply to initial condition of a DBA or transient analysis. Crane travel over the spent fuel pool does not provide a primary part of the primary success path for mitigating a DBA or transient.

The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specifications branch reviewers during telephone calls on June 25 and June 30, 1998. The document indicated in Enclosure 3B is controlled by 10 CFR 50.59.

ATTACHED PAGES:

Encl 3A	4
Encl 3B	4

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
04-10	LS20	Adds a footnote stating that penetration flow paths that provide direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. This change is consistent with traveler WOG-76 and with previously approved administrative controls for personnel air locks.
05-01	Insert → R	Q 3.9-7 This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
06-01	Insert → R	Q 3.9-9 This change relocates the CTS section for the Manipulator Crane to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
07-01	Insert → R	Q 3.9-10 This change relocates the current section dealing with crane travel to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
08-01	Insert → A	Q 3.9-11 This change, consistent with NUREG-1431, provides technical guidance and clarification that loading irradiated fuel assemblies in the core is the specific activity of concern that could increase the reactor decay heat load. This is not a technical change because, under these conditions, the only activity that could increase reactor decay heat load is loading irradiated fuel into the reactor vessel.
08-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS6	This change allows the removal of the RHR loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs. This allows increased flexibility for core mapping and isolation valve testing. No operations are permitted that would cause a reduction of the RCS boron concentration. This change is consistent with NUREG-1431.
08-04	Insert → A	Q 3.9-12 This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred. Insert Q 3.9-13
08-05	-	Not Used
08-06	M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days. This change adds a more stringent TS requirement which is appropriate and consistent with NUREG-1431.
09-01	A	~~Most of the requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. This change does not result in a change to technical requirements and is consistent with NUREG-1431.~~ Insert Q 3.9-14

This change relocates the TS section dealing with crane travel to a licensee controlled document as part of the conversion of the current Technical Specification to the format and expanded Bases of the improved Standard Technical Specifications. The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in a storage pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

This proposed TS revision relocates requirements, which do not meet the TS criteria in 10CFR50.36(c)(2)(ii), to documents with established control programs. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS. An evaluation of the applicability of these criteria to this specification is provided in Attachment 21.

To ensure an appropriate level of control, these requirements will be relocated to 1) documents that are subject to the provisions of 10 CFR 50.59, 2) other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a), or 3) to programs that are controlled via the Administrative Controls section of the improved TS. The identification of the specific licensee controlled document containing this requirement is provided in Enclosure 3B of the conversion submittal.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-01 R	This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station.	Yes, see Attachment 21 page 19	Yes, relocated to TRM relocated to an ECG	No, relocated per Amendment 89	No, relocated per Amendment 103 Q3.9-9
06-01 R	This change relocates the CTS section for the Manipulator Crane.	Yes, see Attachment 21 page 21	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103 Q3.9-10
07-01 R	This change relocates the CTS section dealing with crane travel.	Yes, see Attachment 21 page 23	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103 Q 3.9-11
08-01 A	This change provides clarification that loading irradiated fuel assemblies is the activity that could increase the reactor decay heat load.	Yes	Yes	Yes	Yes
08-02 A	This change removes a limit on Reactor Coolant System temperature.	No, not in CTS.	No, not in CTS	Yes	Yes
08-03 LS6	This change allows the removal of the Residual Heat Removal (RHR) loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs.	Yes	Yes	Yes	Yes
08-04 A	This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred.	Yes	Yes	Yes	No
08-05	Not Used	N/A	N/A	N/A	N/A
08-06 M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days.	Yes	Yes	Yes	Yes
09-01 A	The requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. Insert	Yes	No - CPSES does not have containment ventilation specs in CTS 3/4.9.	Yes	Yes Q 3.9-14

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-12 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

CTS 3.9.8.1 Footnote * (Comanche Peak and Callaway)
CTS 3.9.8.1 Footnote * and ** (Diablo Canyon)
CTS 4.9.8.1 Footnote * (Wolf Creek)
DOC 8-03-LS-6

The CTS requirement, which allows RHR loop to be removed from operation during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot leg, is proposed to be deleted. The DOC does not address any technical justification but states that this change would allow increased flexibility for core mapping and isolation valve testing, and that this change is consistent with NUREG-1431, Rev. 1. In addition, there is not any discussions on the possible increase in risk associated with decay heat removal.

Comment: Revise the DOC by providing the justification as to why this deletion is acceptable and how it relates to the current licensing and design bases. Was there any risk assessment performed in regards to this issue? If so, what were the conclusions that would support the proposed change?

FLOG RESPONSE: DOC 8-03-LS-6 is revised to provide the following additional justification as to why the deletion of the restriction on the 1 hour per 8 hour removal of operation of the RHR loop is acceptable: "The proposed change would permit the securing of RHR flow through the reactor vessel for up to 1 hour in every eight hours [and 2 hours in every 8 hours for testing of the RHR pump to RCS suction valves] provided that no operations involving a reduction in boron concentration were performed. The current TS already permit these interruptions in RHR operation but under more limited conditions. During this interruption, decay heat removal is assured by natural convection within the large amount of water in the refueling cavity (≥ 23 feet above the reactor vessel flange). Boron concentration concerns are avoided by prohibiting evolutions that would reduce the boron concentration. During MODE 6 operation with water levels ≥ 23 feet above the reactor vessel flange, the reactor coolant temperature is maintained significantly below boiling with the typical time-to-boil significantly in excess of [2 hours]. Decay heat removal is provided and inadvertent criticality is avoided."

ATTACHED PAGES:

Encl 3A 4

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
04-10	LS20	Adds a footnote stating that penetration flow paths that provide direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. This change is consistent with traveler WOG-76 and with previously approved administrative controls for personnel air locks.
05-01	Insert → R	Q 3.9-7 This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
06-01	Insert → R	Q 3.9-9 This change relocates the CTS section for the Manipulator Crane to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
07-01	Insert → R	Q 3.9-10 This change relocates the current section dealing with crane travel to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
08-01	Insert → A	Q 3.9-11 This change, consistent with NUREG-1431, provides technical guidance and clarification that loading irradiated fuel assemblies in the core is the specific activity of concern that could increase the reactor decay heat load. This is not a technical change because, under these conditions, the only activity that could increase reactor decay heat load is loading irradiated fuel into the reactor vessel.
08-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS6	This change allows the removal of the RHR loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs. This allows increased flexibility for core mapping and isolation valve testing. No operations are permitted that would cause a reduction of the RCS boron concentration. This change is consistent with NUREG-1431.
08-04	Insert → A	Q 3.9-12 This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred. Insert Q 3.9-13
08-05	-	Not Used
08-06	M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days. This change adds a more stringent TS requirement which is appropriate and consistent with NUREG-1431.
09-01	A	~~Most of the requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. This change does not result in a change to technical requirements and is consistent with NUREG-1431.~~ Insert Q 3.9-14

DOC 08-03-LS6

The proposed change would permit the securing of RHR flow through the reactor vessel for up [one hour in every eight hours and two hours in every eight hours for testing of the RHR pump to RCS suction valves] provided that no operations involving a reduction in boron concentration were performed. The current TS already permit these interruptions in RHR operation but under more limited conditions. During this interruption, decay heat removal is assured by natural convection within the large amount of water in the refueling cavity (≥ 23 feet above the reactor vessel flange). Boron concentration concerns are avoided by prohibiting evolutions that would reduce the boron concentration. During Mode 6 operation with water levels ≥ 23 feet above the reactor vessel flange, the reactor coolant temperature is maintained significantly below boiling with the typical time-to-boil significantly in excess of [2] hour. Decay heat removal is provided and inadvertent criticality is avoided.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-13 **APPLICABILITY:** DC, CP, WC

REQUEST:

CTS 3.9.8.2 Footnote *
DOC 8-04-A

The CTS footnote regarding the option of securing RHR prior to initial criticality is proposed to be deleted entirely. This change is acceptable because it is a relaxation provided in the guidance of NUREG-1431, Rev. 1. However, the categorization is in error. This change is not an administrative change, but it is a more restrictive change.

Comment: Provide a revised "L" DOC.

FLOG RESPONSE: This note is applicable to "initial criticality" only. All start ups including those occurring after refueling outages may not apply to this note since initial criticality has already occurred. This note is no longer applicable for plant operations and is obsolete cycle dependent information. This type of information may be removed under an administrative (A) change since no technical changes (either actual or interpretational) are being made.

DOC 08-04-A is revised to add the following statement: " This note is no longer applicable for plant operations and is obsolete cycle dependent information. This type of information may be removed under an administrative (A) change since no technical changes (either actual or interpretational) are being made."

ATTACHED PAGES:

Encl 3A 4

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
04-10	LS20	Adds a footnote stating that penetration flow paths that provide direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. This change is consistent with traveler WOG-76 and with previously approved administrative controls for personnel air locks.
05-01	Insert → R	Q 3.9-7 This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
06-01	Insert → R	Q 3.9-9 This change relocates the CTS section for the Manipulator Crane to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
07-01	Insert → R	Q 3.9-10 This change relocates the current section dealing with crane travel to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
08-01	Insert → A	Q 3.9-11 This change, consistent with NUREG-1431, provides technical guidance and clarification that loading irradiated fuel assemblies in the core is the specific activity of concern that could increase the reactor decay heat load. This is not a technical change because, under these conditions, the only activity that could increase reactor decay heat load is loading irradiated fuel into the reactor vessel.
08-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS6	This change allows the removal of the RHR loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs. This allows increased flexibility for core mapping and isolation valve testing. No operations are permitted that would cause a reduction of the RCS boron concentration. This change is consistent with NUREG-1431.
08-04	Insert → A	Q 3.9-12 This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred. Insert Q 3.9-13
08-05	-	Not Used
08-06	M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days. This change adds a more stringent TS requirement which is appropriate and consistent with NUREG-1431.
09-01	A	~~Most of the requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. This change does not result in a change to technical requirements and is consistent with NUREG-1431.~~ Insert Q 3.9-14

DOC 08-04-A

This note is no longer applicable for plant operations and is obsolete cycle dependent information. This type of information may be removed under an administrative (A) change since no technical changes (either actual or interpretational) are being made.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-14 **APPLICABILITY:** DC, WC, CA

REQUEST:

CTS 3.9.9
DOC 9-01-A

It is stated in the DOC that the requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. It is not clear exactly which one of these designations would address the requirements.

Comment: Revise the DOC to identify where, specifically, the requirements would be addressed in the ITS and provide the justification for the incorporation

FLOG RESPONSE: DOC 09-01-A is revised as follows: "CTS LCO 3.9.9 requires the containment ventilation system to be OPERABLE during CORE ALTERATIONS or during movement of irradiated fuel assemblies. NUREG-1431 reformats these requirements into ITS LCO 3.9.4.c for each penetration providing direct access from the containment atmosphere to the outside atmosphere. The ITS requires the penetration to be closed by the isolation valve or be capable of being closed by an OPERABLE containment purge isolation valve. CTS Action 3.9.9.b specifically denotes that the provisions of LCO 3.0.3 [is] not applicable. NUREG-1431 does not contain reference to this provision, but rather reformats the presentation of this requirement within the requirements denoted in ITS 3.0.3 [ ]. Specifically, ITS LCO 3.0.3 [is] only applicable in MODES 1, 2, 3, and 4. During this reformatting, no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified. This change is consistent with NUREG-1431."

ATTACHED PAGES:

Encl 3A	4
Encl 3B	5

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
04-10	LS20	Adds a footnote stating that penetration flow paths that provide direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. This change is consistent with traveler WOG-76 and with previously approved administrative controls for personnel air locks.
05-01	Insert → R	Q 3.9-7 This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
06-01	Insert → R	Q 3.9-9 This change relocates the CTS section for the Manipulator Crane to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
07-01	Insert → R	Q 3.9-10 This change relocates the current section dealing with crane travel to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
08-01	Insert → A	Q 3.9-11 This change, consistent with NUREG-1431, provides technical guidance and clarification that loading irradiated fuel assemblies in the core is the specific activity of concern that could increase the reactor decay heat load. This is not a technical change because, under these conditions, the only activity that could increase reactor decay heat load is loading irradiated fuel into the reactor vessel.
08-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS6	This change allows the removal of the RHR loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs. This allows increased flexibility for core mapping and isolation valve testing. No operations are permitted that would cause a reduction of the RCS boron concentration. This change is consistent with NUREG-1431.
08-04	Insert → A	Q 3.9-12 This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred. Insert
08-05	-	Not Used Q 3.9-13
08-06	M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days. This change adds a more stringent TS requirement which is appropriate and consistent with NUREG-1431.
09-01	A	~~Most of the requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. This change does not result in a change to technical requirements and is consistent with NUREG-1431.~~ Insert Q 3.9-14

DOC 09-01-A

CTS LCO 3.9.9 requires the containment ventilation system to be OPERABLE during CORE ALTERATIONS or during movement of irradiated fuel assemblies. NUREG-1431 reformats these requirements into ITS LCO 3.9.4.c for each penetration providing direct access from the containment atmosphere to the outside atmosphere. The ITS requires the penetration to be closed by the isolation valve or be capable of being closed by an OPERABLE containment purge isolation valve. CTS Action 3.9.9.b specifically denotes that the provisions of LCO 3.0.3 [ ] are not applicable. NUREG-1431 does not contain reference to this provision, but rather reformats the presentation of this requirement within the requirements denoted in ITS 3.0.3 [ ]. Specifically, ITS LCO 3.0.3 [ ] is only applicable in Mode 1, 2, 3, and 4. During this reformatting no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified. This change is consistent with NUREG-1431.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-01 R	This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station.	Yes, see Attachment 21 page 19 —	Yes, relocated to TRM relocated to an ECG	No, relocated per Amendment 89	No, relocated per Amendment 103 Q3.9-9
06-01 R	This change relocates the CTS section for the Manipulator Crane.	Yes, see Attachment 21 page 21 —	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103 Q3.9-10
07-01 R	This change relocates the CTS section dealing with crane travel.	Yes, see Attachment 21 page 23 —	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103 Q 3.9-11
08-01 A	This change provides clarification that loading irradiated fuel assemblies is the activity that could increase the reactor decay heat load.	Yes	Yes	Yes	Yes
08-02 A	This change removes a limit on Reactor Coolant System temperature.	No, not in CTS.	No, not in CTS	Yes	Yes
08-03 LS6	This change allows the removal of the Residual Heat Removal (RHR) loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs.	Yes	Yes	Yes	Yes
08-04 A	This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred.	Yes	Yes	Yes	No
08-05	Not Used	N/A	N/A	N/A	N/A
08-06 M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days.	Yes	Yes	Yes	Yes
09-01 A	The requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. Insert	Yes	No - CPSES does not have containment ventilation specs in CTS 3/4.9.	Yes	Yes Q 3.9-14

Enclosure 3B	5
DOC 09-01-A	CTS Action 3.9.9.b is moved to ITS 3.0.3 [ ]. The remaining requirements of this LCO are incorporated into ITS LCO 3.9.4, "Containment Penetrations."

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-15 **APPLICABILITY:** DC, WC, CA

REQUEST:

CTS 3.9.9 Action a
DOC 9-02-LS-7

The CTS requirement to close each purge valve when the containment ventilation system is inoperable is proposed to be deleted. The only statement made in the DOC related to this deletion is that "The ITS only requires that core alterations and irradiated fuel movement be suspended." There was not any technical discussion with respect to this proposed deletion, or how the deletion affects plant operation, licensing and design bases, and why it is acceptable.

Comment: Revise the DOC by providing technical justification to support why it is acceptable.

FLOG RESPONSE: Additional information supporting this DOC is in NSHC LS-7 in Enclosure 4. As discussed with the NRC technical specification branch reviews on June 25, 1998, DOC 9-02-LS7 is revised to include the following information:

"The containment ventilation TS requirements would be integrated into ITS 3.9.4. This has the effect of changing the actions required when the ventilation system is inoperable from closing the purge valves to suspending core alterations and irradiated fuel movement. The applicability of the LCO and required actions for both the current TS and ITS 3.9.4 are identical, i.e., during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. Therefore, neither of these LCOs would be in effect if CORE ALTERATIONS or movement of irradiated fuel were suspended. The function of the purge valves is to close following a FHA. In addition, the change from requiring the valves to be closed to prevent radioactivity release to suspending activities which could lead to a FHA (and radioactivity release) would reduce the effect with regard to consequences of the accident since the accident would be prevented."

ATTACHED PAGES:

Encl 3A 5

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
09-02	LS7	Q3.9-15 Deletes the requirement to close each purge valve when the Containment Ventilation System is inoperable. The ITS only requires that core alterations and irradiated fuel movement be suspended. Insert
09-03	LS8	The containment ventilation TS requirements would be integrated into ITS 3.9.4. This has the effect of changing the SR frequency from once per 7 days and within 100 hours prior to CORE ALTERATIONS to once per 18 months. This change is acceptable because it would apply the same 18-month frequency to the containment ventilation valves as applied to other containment isolation valves that must be OPERABLE for accidents more severe than an FHA.
09-04	Insert → LS15	Q3.9-16 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-01	R	This change relocates the CTS requirements concerning reactor vessel water level for movement of control rods.
10-02	Insert → LS22	Q3.9-17 This change deletes the surveillance requirement to verify water level within 2 hours prior to the start of movement of [irradiated] fuel assemblies. This is acceptable because the LCO must be met at the time that movement of [irradiated] fuel assemblies is performed.
10-03	LS18	Revises Applicability such that it applies only when irradiated fuel is moved. The CTS also applies to movement of un-irradiated fuel when there is irradiated fuel in the core. Insert Q3.9-18
10-04 11-01	Q3.9-21 LG	~~This change modifies the Applicability to "during" movement of irradiated fuel assemblies in the fuel storage pool to be consistent with the Fuel Handling Accident (FHA). The portions of this requirement applicable to whenever irradiated fuel is in the fuel storage pool will be moved to a Licensee controlled document. This change is consistent with NUREG-1431, and removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.~~
11-02	Insert → LS10	This change modifies the ACTION statement by eliminating the 4 hour time requirement to restore water level. This change, which is consistent with NUREG-1431, is acceptable because the ITS Required Action would suspend movement of irradiated fuel immediately which would establish conditions outside the Applicability of the LCO.
11-03	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-04	A	Q3.9-20

DOC 09-02-LS7 "The containment ventilation TS requirements would be integrated into ITS 3.9.4. This has the effect of changing the actions required when the ventilation system in inoperable from closing the purge valves to suspending core alterations and irradiated fuel movement. The applicability of the LCO and required actions for both the current TS and ITS 3.9.4 are identical, i.e., During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. Therefore, neither of these LCOs would be in effect if CORE ALTERATIONS or movement of irradiated fuel were suspended. The function of the purge valves is to close following a FHA. In addition, the change from requiring the valves to be closed to prevent radioactivity release to suspending activities which could lead to a FHA (and radioactivity release) would reduce the effect with regard to consequences of the accident since the accident would be prevented."

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-16 **APPLICABILITY:** DC, WC, CA

REQUEST:

CTS 4.9.9
DOC 9-03-LS-8
DOC 9-03-LS-8 and 9-03-LS-5 (Callaway)
ITS 3.9.4.1 and 3.9.4.2
See also question **3.9-8**

The CTS requirement is proposed to be incorporated into ITS 3.9.4. The DOC states that this has the effect of changing the SR frequency from once per 7 days and within 100 hours prior to CORE ALTERATIONS to once per 18 months. However, there is no provision added in ITS 3.9.4, including 3.9.4.1 and 3.9.4.2, to change the SR frequency within 100 hours as it is stated in CTS 4.9.9. The same question in **3.9-8** also applies to this question.

Comment: Revise the DOC by providing justification and explanation as to why the within 100 hours provision is not included. Also, incorporate the response to question **3.9-8** as well.

FLOG RESPONSE: Additional information supporting this DOC is in NSHC LS-8 in Enclosure 4. As discussed with the NRC technical specification branch reviewers on June 25, 1998, DOC 9-03-LS-8 is revised to include the following information:

"The purpose of the SR is to assure the OPERABILITY of the containment penetrations that must be closed or capable of closing to prevent the release of radioactivity in the event of a Fuel Handling Accident. So, the SR is intended to assure that mitigation features are available and has no impact on the probability of an accident occurring. In addition, the change involves a revision in the frequency of testing accident (Fuel Handling Accident) mitigating equipment.

The Applicability statement for this LCO is "During CORE ALTERATIONS or movement of irradiated fuel within the containment." Thus, the surveillance requirement to verify the LCO is met within 100 hours of starting the evolutions for which the LCO is applicable is redundant, because the Applicability states the LCO must be met at the time the evolutions occur. Also, the proposed change to 18 months would apply the same frequency of testing to containment purge isolation valves as applied to other containment isolation valves that must be OPERABLE during reactor operations. The 18-month frequency has been found adequate for the type of testing applied to instrumentation and valves that must mitigate events much more severe and much more challenging to the containment boundary (e.g., LOCA, MSLB) than the FHA."

The Comment includes a reference to Callaway for DOC 9-03-LS-5 in CTS 4.9.9 and ITS 3.9.4.2. It appears the reference for the DOC is incorrect and was intended to be DOC 4-03-LS-5. The response to Comment Number 3.9-8 and revision to DOC 4-03-LS-5 addresses the portion of Comment Number 3.9-16 for Callaway's use of DOC 4-03-LS-5 in CTS 4.9.9 and ITS 3.9.4.2.

ATTACHED PAGES:

Encl. 3A 5

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
09-02	LS7	Q3.9-15 Deletes the requirement to close each purge valve when the Containment Ventilation System is inoperable. The ITS only requires that core alterations and irradiated fuel movement be suspended. Insert
09-03	LS8	The containment ventilation TS requirements would be integrated into ITS 3.9.4. This has the effect of changing the SR frequency from once per 7 days and within 100 hours prior to CORE ALTERATIONS to once per 18 months. This change is acceptable because it would apply the same 18-month frequency to the containment ventilation valves as applied to other containment isolation valves that must be OPERABLE for accidents more severe than an FHA.
09-04	Insert → LS15	Q3.9-16 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-01	R	This change relocates the CTS requirements concerning reactor vessel water level for movement of control rods.
10-02	Insert → LS22	Q3.9-17 This change deletes the surveillance requirement to verify water level within 2 hours prior to the start of movement of [irradiated] fuel assemblies. This is acceptable because the LCO must be met at the time that movement of [irradiated] fuel assemblies is performed.
10-03	LS18	Revises Applicability such that it applies only when irradiated fuel is moved. The CTS also applies to movement of un-irradiated fuel when there is irradiated fuel in the core. Insert Q3.9-18
10-04 11-01	Q3.9-21 LG	~~This change modifies the Applicability to "during" movement of irradiated fuel assemblies in the fuel storage pool to be consistent with the Fuel Handling Accident (FHA). The portions of this requirement applicable to whenever irradiated fuel is in the fuel storage pool will be moved to a Licensee controlled document. This change is consistent with NUREG-1431, and removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.~~
11-02	Insert → LS10	This change modifies the ACTION statement by eliminating the 4 hour time requirement to restore water level. This change, which is consistent with NUREG-1431, is acceptable because the ITS Required Action would suspend movement of irradiated fuel immediately which would establish conditions outside the Applicability of the LCO.
11-03	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-04	A	Q3.9-20

DOC 9-03-LS8

The purpose of the SR is to assure the OPERABILITY of the containment penetrations that must be closed or capable of closing to prevent the release of radioactivity in the event of a Fuel Handling Accident. So, the SR is intended to assure that mitigation features are available and has no impact on the probability of an accident occurring. In addition, the change involves a revision in the frequency of testing accident (Fuel Handling Accident) mitigating equipment.

The Applicability statement for this LCO is "During CORE ALTERATIONS or movement of irradiated fuel within the containment." Thus the requirement to verify the LCO is met within 100 hours of starting the evolutions for which the LCO is applicable is redundant; because the LCO must be met at the time that the evolutions occur. Also, the proposed change to 18 months would apply the same frequency of testing to containment purge isolation valves as applied to other containment isolation valves that must be OPERABLE during reactor operations. The 18-month frequency has been found adequate for the type of testing applied to instrumentation and valves that must mitigate events much more severe and much more challenging to the containment boundary (e.g., LOCA, MSLB) than the FHA."

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-17 **APPLICABILITY:** DC, CP

REQUEST:

CTS 3/4.9.9, Control Rods
DOC 10-01-R

According to the DOC, the CTS requirements in 3/4.9.7 would be entirely relocated to an unspecified licensee controlled document. In addition, the DOC does not address adequate justification as to why the relocation is acceptable.

Comment: Provide additional justification as to why this relocation is acceptable and identify the name of the licensee controlled document containing this requirement. This requirement shall be relocated to a licensee controlled document controlled by 10 CFR 50.59.

FLOG RESPONSE: DOC 10-01-R has been revised and a Technical Specification Screening Form has been prepared to provide justification as to why the relocation is acceptable. This justification shows the LCO provides assurance that adequate water will be present for iodine removal in the event of a FHA. Control rod movement is not associated with FHA and this LCO does not address inadvertent criticality. This LCO is not associated with any instrumentation used to detect significant degradation of the reactor coolant pressure boundary. It is not associated with any variable design feature or restriction which is an initial condition of DBA or transient analysis. It is not a part of the primary success path for mitigation of a DBA or a transient.

The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specifications branch reviewers during telephone calls on June 25 and June 30, 1998. The document indicated in Enclosure 3B is controlled by 10CFR50.59.

ATTACHED PAGES:

Encl 3A	5
Encl 3B	5

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
09-02	LS7	Q3.9-15 Deletes the requirement to close each purge valve when the Containment Ventilation System is inoperable. The ITS only requires that core alterations and irradiated fuel movement be suspended. Insert
09-03	LS8	The containment ventilation TS requirements would be integrated into ITS 3.9.4. This has the effect of changing the SR frequency from once per 7 days and within 100 hours prior to CORE ALTERATIONS to once per 18 months. This change is acceptable because it would apply the same 18-month frequency to the containment ventilation valves as applied to other containment isolation valves that must be OPERABLE for accidents more severe than an FHA.
09-04	Insert → LS15	Q3.9-16 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-01	R	This change relocates the CTS requirements concerning reactor vessel water level for movement of control rods.
10-02	Insert → LS22	Q3.9-17 This change deletes the surveillance requirement to verify water level within 2 hours prior to the start of movement of [irradiated] fuel assemblies. This is acceptable because the LCO must be met at the time that movement of [irradiated] fuel assemblies is performed.
10-03	LS18	Revises Applicability such that it applies only when irradiated fuel is moved. The CTS also applies to movement of un-irradiated fuel when there is irradiated fuel in the core. Insert
10-04		Q3.9-18
11-01	Q3.9-21 LG	~~This change modifies the Applicability to "during" movement of irradiated fuel assemblies in the fuel storage pool to be consistent with the Fuel Handling Accident (FHA). The portions of this requirement applicable to whenever irradiated fuel is in the fuel storage pool will be moved to a Licensee controlled document. This change is consistent with NUREG-1431, and removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.~~
11-02	Insert → LS10	This change modifies the ACTION statement by eliminating the 4 hour time requirement to restore water level. This change, which is consistent with NUREG-1431, is acceptable because the ITS Required Action would suspend movement of irradiated fuel immediately which would establish conditions outside the Applicability of the LCO.
11-03	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-04	A	Q3.9-20

This specification places a lower limit on the amount of water above the top of the fuel assemblies in the reactor vessel during movement of control rods. The Bases state that this ensures the water removes 99 percent of the assumed 10 percent iodine gap activity released from the rupture of an irradiated fuel assembly in the event of a fuel handling accident (FHA) during core alterations. However, the movement of control rods is not associated with the initial conditions of an FHA, and the Bases do not address any concerns regarding inadvertent criticality which could lead to a breach of the fuel rod cladding. Inadvertent criticality during Mode 6 is prevented by maintaining proper boron concentration in the coolant in accordance with LCO 3.9.1.

This proposed TS revision relocates requirements, which do not meet the TS criteria in 10CFR50.36(c)(2)(ii), to documents with established control programs. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS. An evaluation of the applicability of these criteria to this specification is provided in Attachment 21.

To ensure an appropriate level of control, these requirements will be relocated to 1) documents that are subject to the provisions of 10 CFR 50.59, 2) other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a), or 3) to programs that are controlled via the Administrative Controls section of the improved TS. The identification of the specific licensee controlled document containing this requirement is provided in Enclosure 3B of the conversion submittal.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
09-02 LS7	This change deletes the requirement to close each purge valve when the containment ventilation system is inoperable.	Yes	No, CPSES does not have containment ventilation specs in CTS 3/4.9.	Yes	Yes
09-03 LS8	The containment ventilation TS requirements would be integrated into ITS 3.9.4. This has the effect of changing the SR frequency from once per 7 days and within 100 hours prior to CORE ALTERATIONS to once per 18 months.	Yes	No, CPSES does not have containment ventilation specs in CTS 3/4.9.	Yes	Yes
09-04 LS15	Removes the requirement for immediate action when one containment purge monitor is inoperable. ITS LCO 3.3.6 will allow one purge monitor to be inoperable for up to 4 hours during CORE ALTERATIONS or movement of irradiated fuel in containment.	No, not in CTS.	No, CPSES does not have Containment Ventilation Spec in CTS 3/4.9	No, Plant design different	Yes
10-01 R	This change relocates the CTS requirements concerning reactor vessel water level for movement of control rods.	Yes, see Attachment 21 page 25 — relocated to an ECG	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103 Q 3.9-17
10-02 LS22	This change deletes the surveillance requirement to verify water level within 2 hours prior to the start of movement of irradiated fuel assemblies.	Yes	Yes	Yes	Yes
10-03 LS18	Revises Applicability such that it applies only when irradiated fuel is moved. The CTS also applies to movement of un-irradiated fuel when there is irradiated fuel in the core.	Yes	Yes	No, already in CTS	No, already in CTS
11-01 LG	This change modifies the Applicability to during movement of irradiated fuel assemblies in the fuel storage pool to be consistent with the Fuel Handling Accident (FHA).	Yes, moved to an ECG	Yes	Yes	Yes
11-02 LS10	This change modifies the ACTION statement and eliminates the time requirement to restore water level.	Yes	Yes	Yes	Yes

10-04 A Insert in WCNOC submittal. Q3.9-20

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-18 **APPLICABILITY:** DC, CP

REQUEST:

CTS 3.9.9.1 Applicability
DOC 10-03-LS18

CTS requires movement of unirradiated fuel when there is irradiated fuel in the core. The licensee proposes to revise the applicability such that it applies only when irradiated fuel is moved. There is not any technical discussion provided in the DOC to justify this change.

Comment: Please provide technical justification in the DOC as to why this is technically acceptable and how it applies to current licensing basis.

FLOG RESPONSE: The text of DOC 10-03-LS-18 is revised to add the following statement: "The purpose of [CTS 3.9.10.1] is to assure that sufficient water is present to remove 99% of the release of 10% iodine gap activity during a FHA. A FHA could result during movement of an irradiated assembly due to the drop of the assembly on the floor of the refueling cavity or the drop of any load on the core. A fuel assembly and its handling tool are the analyzed limiting load for the FHA. However, dropping a new (unirradiated) assembly on the cavity floor would not result in an FHA and dropping a new assembly on the core is no different then dropping any other load on the core. This condition is already addressed by the programs associated with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and is redundant to this LCO requirement. The FHA resulting from the drop of a new assembly would occur at the active irradiated fuel already loaded in the core and maintenance of 23 feet of water above the flange (this LCO) would be unnecessarily conservative.

ATTACHED PAGES:

Encl 3A 5

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
09-02	LS7	Q3.9-15 Deletes the requirement to close each purge valve when the Containment Ventilation System is inoperable. The ITS only requires that core alterations and irradiated fuel movement be suspended. Insert
09-03	LS8	The containment ventilation TS requirements would be integrated into ITS 3.9.4. This has the effect of changing the SR frequency from once per 7 days and within 100 hours prior to CORE ALTERATIONS to once per 18 months. This change is acceptable because it would apply the same 18-month frequency to the containment ventilation valves as applied to other containment isolation valves that must be OPERABLE for accidents more severe than an FHA.
09-04	Insert → LS15	Q3.9-16 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-01	R	This change relocates the CTS requirements concerning reactor vessel water level for movement of control rods.
10-02	Insert → LS22	Q3.9-17 This change deletes the surveillance requirement to verify water level within 2 hours prior to the start of movement of [irradiated] fuel assemblies. This is acceptable because the LCO must be met at the time that movement of [irradiated] fuel assemblies is performed.
10-03	LS18	Revises Applicability such that it applies only when irradiated fuel is moved. The CTS also applies to movement of un-irradiated fuel when there is irradiated fuel in the core. Insert
10-04		Q3.9-18
11-01	Q3.9-21 LG	~~This change modifies the Applicability to "during" movement of irradiated fuel assemblies in the fuel storage pool to be consistent with the Fuel Handling Accident (FHA). The portions of this requirement applicable to whenever irradiated fuel is in the fuel storage pool will be moved to a Licensee controlled document. This change is consistent with NUREG-1431, and removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.~~
11-02	Insert → LS10	This change modifies the ACTION statement by eliminating the 4 hour time requirement to restore water level. This change, which is consistent with NUREG-1431, is acceptable because the ITS Required Action would suspend movement of irradiated fuel immediately which would establish conditions outside the Applicability of the LCO.
11-03	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-04	A	Q3.9-20

DOC 10-03-LS18 The purpose of [CTS 3.9.10.1] is to assure that sufficient water is present to remove 99% of the release of 10% iodine gap activity during a FHA. A FHA could result during movement of an irradiated assembly due to the drop of the assembly on the floor of the refueling cavity or the drop of any load on the core. A fuel assembly and its handling tool are the analyzed limiting load for the FHA. However, dropping a new (unirradiated) assembly on the cavity floor would not result in an FHA and dropping a new assembly on the core is no different then dropping any other load on the core. The FHA resulting from the drop of a new assembly would occur at the active irradiated fuel already loaded in the core and maintenance of 23 feet of water above the flange (this LCO) would be unnecessarily conservative.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-21 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

CTS 3.9.10 Applicability and 4.9.10 (Comanche Peak)
CTS 3.9.11 and 4.9.11 (Callaway)
CTS 3.9.11 Applicability and 4.9.11 (Diablo Canyon)
CTS 3.9.11 and 4.9.11 (Wolf Creek)
DOC 11-01-LG

The CTS requirement, applicable to whenever irradiated fuels are in the fuel storage racks, is proposed to be relocated to an unspecified licensee controlled document. The DOC does not provide any technical justification related to this relocation.

Comment: Revise the DOC by including the justification for the relocation and identify the licensee controlled document containing this requirement. This requirement shall be relocated to a licensee controlled document controlled by 10 CFR 50.59.

FLOG RESPONSE: Identification of the licensee controlled document containing this requirement is identified in Enclosure 3B of the conversion application. This was discussed with the NRC technical specification branch reviewers on June 25, 1998, and determined that the information provided in Enclosure 3B was acceptable.

DOC 11-01-LG has been modified to include the following information: "The Applicability is revised to require the water level be maintained only when moving irradiated fuel. The bounding design basis fuel handling accident in the fuel storage [pool] assumes an irradiated fuel assembly is dropped onto an array of irradiated fuel assemblies seated in the fuel storage racks. The revised Applicability is consistent with this design basis accident (DBA). The water level requirement is necessary to mitigate the consequences of this DBA. Moving the requirement to maintain fuel storage [pool] water level whenever irradiated fuel assemblies are in the fuel storage [pool] provides for conservative plant operations. Moving this information maintains consistency with NUREG-1431. The information is moved to a licensee controlled document which is controlled by a 10 CFR 50.59 change process."

ATTACHED PAGES:

Encl 3A 5

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
09-02	LS7	Q3.9-15 Deletes the requirement to close each purge valve when the Containment Ventilation System is inoperable. The ITS only requires that core alterations and irradiated fuel movement be suspended. Insert
09-03	LS8	The containment ventilation TS requirements would be integrated into ITS 3.9.4. This has the effect of changing the SR frequency from once per 7 days and within 100 hours prior to CORE ALTERATIONS to once per 18 months. This change is acceptable because it would apply the same 18-month frequency to the containment ventilation valves as applied to other containment isolation valves that must be OPERABLE for accidents more severe than an FHA.
09-04	Insert → LS15	Q3.9-16 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-01	R	This change relocates the CTS requirements concerning reactor vessel water level for movement of control rods.
10-02	Insert → LS22	Q3.9-17 This change deletes the surveillance requirement to verify water level within 2 hours prior to the start of movement of [irradiated] fuel assemblies. This is acceptable because the LCO must be met at the time that movement of [irradiated] fuel assemblies is performed.
10-03	LS18	Revises Applicability such that it applies only when irradiated fuel is moved. The CTS also applies to movement of un-irradiated fuel when there is irradiated fuel in the core. Insert Q3.9-18
10-04		
11-01	Q3.9-21 LG	~~This change modifies the Applicability to "during" movement of irradiated fuel assemblies in the fuel storage pool to be consistent with the Fuel Handling Accident (FHA). The portions of this requirement applicable to whenever irradiated fuel is in the fuel storage pool will be moved to a Licensee controlled document. This change is consistent with NUREG-1431, and removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.~~
11-02	Insert → LS10	This change modifies the ACTION statement by eliminating the 4 hour time requirement to restore water level. This change, which is consistent with NUREG-1431, is acceptable because the ITS Required Action would suspend movement of irradiated fuel immediately which would establish conditions outside the Applicability of the LCO.
11-03	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-04	A	Q3.9-20

DOC 11-01-LG

The Applicability is revised to require the water level be maintained only when moving irradiated fuel. The bounding design basis fuel handling accident in the fuel storage [pool] assumes an irradiated fuel assembly is dropped onto an array of irradiated fuel assemblies seated in the fuel storage racks. The revised Applicability is consistent with this design basis accident (DBA). The water level requirement is necessary to mitigate the consequences of this DBA. Moving the requirement to maintain fuel storage [pool] water level whenever irradiated fuel assemblies are in the fuel storage [pool] provides for conservative plant operations. Moving this information maintains consistency with NUREG-1431. The information is moved to a licensee controlled document which is controlled by a 10 CFR 50.59 change process.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-22 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

CTS 3.9.11 Action a
CTS 3.9.10 Action a (Comanche Peak)
DOC 11-04-LG

According to the DOC, the CTS requirement regarding restriction on crane operation is proposed to be moved to an unspecified licensee controlled document. In addition, the DOC does not provide specific justification as to why this relocation is acceptable.

Comment: Revise the DOC to include specific justification to this relocation and identify the licensee controlled document containing this requirement. This requirement shall be relocated to a licensee controlled document controlled by 10 CFR 50.59.

FLOG RESPONSE: Identification of the licensee-controlled document containing this requirement is identified in Enclosure 3B of the conversion application. This was discussed with the NRC technical specification branch reviewers on June 25, 1998, and determined that the information provided in Enclosure 3B was acceptable.

DOC 11-04-LG has been modified to include the following information: "The requirement to suspend crane operations over the spent fuel pool in the event pool water level is <23 feet, has been removed from the ACTION of CTS 3/4.9.11 (CTS 3/4.9.10 for Comanche Peak) in corresponding ITS 3.7.5, for the fuel pool water level. The bounding design basis fuel handling accident in the spent fuel pool assumes an irradiated fuel assembly is dropped onto [the floor of] the spent fuel pool. Crane operations that could adversely affect fuel stored in the spent fuel pool are controlled in accordance with plant procedures as analyzed in the review of heavy loads movements. Administrative controls are employed to prevent the handling of loads that have a greater potential energy than those which have been analyzed. Also see licensees' responses to NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment." Moving this information maintains consistency with NUREG-1431. The information is moved to a licensee controlled document which is controlled by a 10 CFR 50.59 change process.

ATTACHED PAGES:

Encl 3A 6

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
11-04	LG	This change moves the restriction on crane operation to a licensee controlled document. The restriction on crane operations may be removed because it is not in the assumptions used for the FHA. Crane operations that could adversely affect fuel stored in the spent fuel pool are controlled as analyzed in the review of heavy load movements. This change is consistent with NUREG-1431, and moves requirements that do not meet the criteria for inclusion in the TS.
12-01	LS24	Insert → (Q 3.9-22) The Applicability would be changed to "During movement of irradiated fuel in the fuel building" instead of "Whenever irradiated fuel is in the spent fuel pool" consistent with NUREG-1431. The proposed Applicability is consistent with the assumptions used in the FHA in the Fuel Handling Building which postulates the inadvertent drop of an irradiated fuel assembly. Potential damage to fuel assemblies due to dropping of heavy loads is addressed by CN 12-02-LG.
12-02	LG	Moves the restriction on crane operations over the spent fuel storage areas when the fuel building air cleanup system was inoperable. The restriction on crane operations may be removed because it is not consistent with the assumptions used for the FHA. Crane operations that could adversely affect fuel stored in the spent fuel pool is prohibited in accordance with plant procedures as analyzed in the review of heavy load movements.
12-03	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-04	A	The SR regarding filter testing would be moved to a "Ventilation Filter Testing Program" that is called out in the Administrative Controls Section 5.5.11 of the ITS. This change does not result in a change to technical requirements.
12-05	TR1	Revised SR to allow for increased flexibility in using an actual or simulated actuation signal. Identification of the specific signal is moved to the Bases.
12-06	A	This requirement would have the operability of each train of the [Fuel Handling Building Ventilation System (FHBVS)] (including maintaining negative pressure in the building) to be demonstrated. This is consistent with current practice. This change does not result in a change to technical requirements and is consistent with NUREG-1431.
12-07	LS25	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-08	LS16	The proposed change would allow the ~~18~~ [24]-month testing of the [FHBVS] ability to maintain the required pressure differential between the building and the outside atmosphere to be performed on a STAGGERED TEST BASIS. (DC-ALL-001)

DOC 11-04-LG

The requirement to suspend crane operations over the spent fuel pool in the event pool water level is <23 feet, has been removed from the ACTION of CTS 3/4.9.11 (CTS 3/4.9.10 for Comanche Peak) in corresponding ITS 3.7.5, for the fuel pool water level. The bounding design basis fuel handling accident in the spent fuel pool assumes an irradiated fuel assembly is dropped onto [the floor of] the spent fuel pool. Crane operations that could adversely affect fuel stored in the spent fuel pool are controlled in accordance with plant procedures as analyzed in the review of heavy loads movements. Administrative controls are employed to prevent the handling of loads that have a greater potential energy that those which have been analyzed. Also see licensees' responses to NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment." Moving this information maintains consistency with NUREG-1431. The information is moved to a licensee controlled document which is controlled by a 10 CFR 50.59 change process.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-23 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

ITS 3.9.7 Applicability and Action A
JFD 3.9-10

The licensee is proposing to delete ITS 3.9.7, Applicability, which states, "During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts." JFD 3.9-10 states, the reason for this revision is that this requirement is a duplication of a relocated technical specification requirement for reactor vessel water level during movement of control rods (relocated CTS 3.9.9.2). ITS 3.9.7 Applicability is not an exact duplication of CTS 3.9.9.2; further technical and licensing justifications are required for this deletion.

Comment: Provide technical and licensing bases justification for this proposed deletion from ITS. Why does the inclusion of the ITS requirement pose a hardship?

FLOG RESPONSE: Diablo Canyon, Comanche Peak, Wolf Creek, and Callaway continue to pursue this change. ITS 3.9.7 Applicability statement, "During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts," as applied to fuel movement is limited to movement of fuel within the reactor vessel. The STS wording does not address the movement of fuel in transit to or from the reactor vessel in containment. "CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts" as applied to CORE ALTERATIONS other than fuel movement are not associated with the CTS LCO 3.9.10.1 and CTS LCO 3.9.9.1 for CPSES. All control rod movement in MODE 6 (CTS LCO 3.9.9.2 for CPSES and CTS LCO 3.9.10.2 for DCPP) is proposed to be relocated to licensee-controlled documents based upon 10 CFR 50.36 (see DOC 10-01-R), for Wolf Creek and Callaway, CTS 3.9.10.2 was already relocated by Amendment 89 and 103, respectively.

The purpose of ITS LCO 3.9.7 (CTS LCO 3.9.10.1 (for CPSES CTS LCO 3.9.9.1)) is to provide sufficient water to assure removal of 99% of a 10% iodine gap release in the event of a FHA, whereas, ITS LCO 3.9.1, "Boron Concentration," is provided to address reactivity changes and to prevent inadvertent criticality during movement of fuel or control rods.

JFD 3.9-10 is revised to provide the following statement:

"Consistent with the CTS LCO [3.9.10.1], ITS 3.9.7 Applicability is being revised to delete "During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts." This STS Applicability does not address the movement of fuel in containment except within the reactor vessel. It does, however, apply to CORE ALTERATIONS other than fuel movement. This requirement does not meet the criteria of 10 CFR 50.36 in terms of its association with an FHA. The revised Applicability provides assurance of sufficient water to assure removal of 99% of a 10% iodine gap release in the event of a FHA in the containment while moving irradiated fuel."

For DCPP and CPSES, the revised applicability is not strictly consistent with CTS, but it is within the current licensing bases (See Q 3.9-18).

ATTACHED PAGES:

Encl 6A	2
Encl 6B	2

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.9

CHANGE NUMBER	JUSTIFICATION
3.9-10	Q 3.9-23 ~~Consistent with the CTS, ITS 3.9.7 Applicability is being revised to delete "During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts." The Applicability is being revised because this requirement is duplicative of a relocated technical specification requirement for reactor vessel water level during movement of control rods (relocated technical specification 3.9.10.2). The relocated specification places a lower limit on the amount of water above the top of the fuel assemblies in the reactor vessel during movement of control rods. The Bases (CTS 3/4.9.10) states that this ensures the water removes 99% of the assured 10% iodine gap activity released from the rupture of an irradiated fuel assembly in the event of a fuel handling accident (FHA). However, the movement of control rods is not associated with the initial conditions of a FHA, and the Bases do not address any concern regarding inadvertent criticality which could lead to a breach of the fuel rod cladding.~~
Insert → 3.9-11	In accordance with a proposed traveler WOG-76, LCO is modified to permit a penetration flow path that provides direct access from the containment to the outside atmosphere to be unisolated under administrative controls. The allowance to have containment flow paths with direct access from the containment atmosphere to the outside atmosphere unisolated under administrative controls is based on confirmatory dose calculations of a fuel handling accident which indicate acceptable radiological consequences and to implement administrative controls that ensure that the flow penetrations will be promptly closed following a fuel handling accident, to provide a defense-in-depth approach to meet acceptable dose consequences. The administrative control requirements are defined in the Bases.
3.9-12	A note is added to the Applicability of LCO 3.9.6 indicating that entry into a MODE or other specified condition in the Applicability is not permitted while the LCO is not met. The addition of this note is based on the performance of a plant-specific LCO 3.0.4 matrix which identified where the requirements of 3.0.4 are still applicable in MODES 5 and 6 and in MODES 1, 2, 3, and 4 when the MODE is descending (i.e., from MODE 1 to MODE 2, etc.). This matrix was specified in the NUREG-1431 reviewer's note in LCO 3.0.4.
3.9-13	In accordance with DCPP CTS, LCO 3.9.2 would not be used. This new requirement is not applicable to DCPP which has a licensed dilution accident. The current licensing bases in accordance with NUREG 0800, Section 15.4.6 provides adequate assurance that a dilution event will recognized and arrested in a timely fashion
3.9-14	A Note is added to the Applicability of LCO 3.9.1 indicating that entry into MODE 6 from MODE 5 is not permitted while the LCO is not met. The addition of this Note is based on the performance of a plant-specific LCO 3.0.4 matrix which identified where the requirements of 3.0.4 are still applicable in MODES 5 and 6 and in MODES 1, 2, 3, and 4 when the MODE is descending (i.e., from MODE 1 to MODE 2, etc.). This matrix was specified in the NUREG-1431 reviewer's note in LCO 3.0.4. Insert Q3.9-24
3.9-15 Insert	Q 3.9-1a

JFD 3.9-10

Consistent with the CTS LCO [3.9.10.1], ITS 3.9.7 Applicability is being revised to delete "During CORE ALTERATIONS, except during latching of control rod drive shafts." This Applicability does not address the movement of irradiated fuel in containment except in the core region of the reactor vessel. It does, however, apply to control rod movement other than latching and unlatching of control rods. This requirement does not meet the criteria of 10CFR50.36 in terms of its association with an FHA. The revised Applicability provides assurance of sufficient water to assure removal of 99% of a 10% iodine gap release in the event of a FHA in the containment while moving irradiated fuel.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.9-8	The Note of ITS 3.9.5 is expanded to incorporate CTS allowing the required RHR pump to be removed from service for less than or equal to 2 hours per 8 hours for leak testing of the RHR suction isolation valves.	Yes	No, not in CTS	No, not in CTS	No, not in CTS
3.9-9	The Surveillances of ITS 3.9.5 and 3.9.6 are modified to incorporate CTS of two RHR flow rates dependent upon the number of hours the reactor has been subcritical.	Yes, See LAR 88-01 dated 4/21/88 and DCL 88-067	No, not in CTS	No, not in CTS	No, not in CTS
3.9-10	ITS 3.9.7 Applicability is being revised to delete "During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts." ~~The Applicability is being revised because this requirement is duplicative of a relocated technical specification requirement for reactor vessel water level during movement of control rods.~~	Yes	Yes	Yes~~, relocated per Amendment 89~~	Yes~~, relocated per Amendment 103~~ Q3.9-23
3.9-11	In accordance with traveler WOG-76, LCO is modified to permit a penetration flow path that provides direct access from the containment to the outside atmosphere to be unisolated under administrative controls.	Yes	Yes	Yes	Yes
3.9-12	A Note is added to the Applicability of LCO 3.9.6 indicating that entry into a MODE or other specified condition in the Applicability is not permitted while the LCO is not met.	Yes	Yes	Yes	Yes
3.9-13	In accordance with DCPP CTS, LCO 3.9.2 would not be used	Yes	No	No	No
3.9-14	A Note is added to the Applicability of LCO 3.9.1 indicating that entry into MODE 6 from MODE 5 is not permitted while the LCO is not met.	Yes	Yes	Yes	Yes
3.9-15	Insert				

Q 3.9-1a

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-24 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

ITS 3.9.1 LCO 3.9.1 Note
JFD 3.9-14

A note, "While this LCO is not met, entry into MODE 6 from MODE 5 is not permitted" is added to LCO 3.9.1. JFD states that this restriction would prevent a transition from MODE 5 to MODE 6 if boron concentration limit for MODE 6 is not met. While the intent of this note is understandable, why this note is in ITS 3.9, Refueling Operations, is not clear since the plant would already be in MODE 6 for refueling operations.

Comment: Provide detailed technical discussion in JFD addressing the significance of this note, and why this note should be included.

FLOG RESPONSE: A Reviewer's Note in STS LCO 3.0.4 states: "LCO 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, LCO 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified conditions in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in LCO 3.0.4 were previously applicable in all MODES. Before this version of LCO 3.0.4 can be implemented on a plant-specific basis, the licensee must review the existing TS to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change. Such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS." Based on this Reviewer's Note, a matrix of this evaluation was placed in the NSHC LS-1 in Enclosure 4 of the Section 3.0 package (Attachment No. 6).

JFD 3.9-14 has been revised to incorporate additional justification from NSHC LS-1 from Enclosure 4 of the Section 3.0 package (Attachment No. 6). JFD 3.9-14 has been revised to include: "LCO 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, LCO 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified conditions in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in LCO 3.0.4 were previously applicable in all MODES. ITS LCO 3.9.1 was modified by a Note stating: "While this LCO is not met, entry into MODE 6 from MODE 5 is not permitted." The Required Actions would prevent a transition from defueled to MODE 6 by suspending CORE ALTERATIONS and positive reactivity additions. The transition from MODE 5 to MODE 6 could occur without adequate boration for MODE 6 requirements."

ATTACHED PAGES:

Encl 6A 2

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.9

CHANGE NUMBER	JUSTIFICATION
3.9-10	Q 3.9-23 ~~Consistent with the CTS, ITS 3.9.7 Applicability is being revised to delete "During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts." The Applicability is being revised because this requirement is duplicative of a relocated technical specification requirement for reactor vessel water level during movement of control rods (relocated technical specification 3.9.10.2). The relocated specification places a lower limit on the amount of water above the top of the fuel assemblies in the reactor vessel during movement of control rods. The Bases (CTS 3/4.9.10) states that this ensures the water removes 99% of the assured 10% iodine gap activity released from the rupture of an irradiated fuel assembly in the event of a fuel handling accident (FHA). However, the movement of control rods is not associated with the initial conditions of a FHA, and the Bases do not address any concern regarding inadvertent criticality which could lead to a breach of the fuel rod cladding.~~
Insert → 3.9-11	In accordance with a proposed traveler WOG-76, LCO is modified to permit a penetration flow path that provides direct access from the containment to the outside atmosphere to be unisolated under administrative controls. The allowance to have containment flow paths with direct access from the containment atmosphere to the outside atmosphere unisolated under administrative controls is based on confirmatory dose calculations of a fuel handling accident which indicate acceptable radiological consequences and to implement administrative controls that ensure that the flow penetrations will be promptly closed following a fuel handling accident, to provide a defense-in-depth approach to meet acceptable dose consequences. The administrative control requirements are defined in the Bases.
3.9-12	A note is added to the Applicability of LCO 3.9.6 indicating that entry into a MODE or other specified condition in the Applicability is not permitted while the LCO is not met. The addition of this note is based on the performance of a plant-specific LCO 3.0.4 matrix which identified where the requirements of 3.0.4 are still applicable in MODES 5 and 6 and in MODES 1, 2, 3, and 4 when the MODE is descending (i.e., from MODE 1 to MODE 2, etc.). This matrix was specified in the NUREG-1431 reviewer's note in LCO 3.0.4.
3.9-13	In accordance with DCPP CTS, LCO 3.9.2 would not be used. This new requirement is not applicable to DCPP which has a licensed dilution accident. The current licensing bases in accordance with NUREG 0800, Section 15.4.6 provides adequate assurance that a dilution event will recognized and arrested in a timely fashion
3.9-14	A Note is added to the Applicability of LCO 3.9.1 indicating that entry into MODE 6 from MODE 5 is not permitted while the LCO is not met. The addition of this Note is based on the performance of a plant-specific LCO 3.0.4 matrix which identified where the requirements of 3.0.4 are still applicable in MODES 5 and 6 and in MODES 1, 2, 3, and 4 when the MODE is descending (i.e., from MODE 1 to MODE 2, etc.). This matrix was specified in the NUREG-1431 reviewer's note in LCO 3.0.4. Insert Q3.9-24
3.9-15 Insert	Q 3.9-1a

JFD 3.9-14

LCO 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, LCO 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified conditions in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in LCO 3.0.4 were previously applicable in all MODES. ITS LCO 3.9.1 was modified by a Note stating: "While this LCO is not met, entry into MODE 6 from MODE 5 is not permitted." The Required Actions would prevent a transition from defueled to MODE 6 by suspending CORE ALTERATIONS and positive reactivity additions. The transition from Mode 5 to Mode 6 could occur without adequate boration for MODE 6 requirements.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.9-25 **APPLICABILITY:** DC

REQUEST:

CTS 3/4.9.13 and Figure 3.9-1
DOC 15-01-R

The CTS requirements in 3/4.9.13 are proposed to be entirely relocated to an unspecified licensee controlled document. Though it is addressed in the Conversion Comparison Table where this item is being relocated to, it is necessary to address this in the DOC. In addition, the DOC does not contain any justification as to why the relocation is acceptable.

Comment: Revise the DOC by including the justification for the relocation and identify the licensee controlled document in which the CTS requirements would be relocated.

FLOG RESPONSE: The format for specifying the location of relocated requirements (in Enclosure 3B of the conversion submittal) was found to be acceptable by the NRC technical specification branch reviews during telephone calls June 25 and June 30, 1998.

DOC 15-01-R has been revised and a Technical Specification Screening Form was provided in Attachment 21 to provide justification as to why the relocation is acceptable. This justification shows that this LCO provide assurance that the spent fuel shipping cask will not inadvertently be dropped on irradiated fuel in the spent fuel pool. The dose consequences of the postulated accident addressed by the restriction of this specification are less then the 10 CFR 100 guidelines. This specification does not contain any requirements for installed instrumentation used to indicate an abnormality in the reactor coolant boundary. It does not contain requirements for process variables, design features, or operating restriction that are monitored or controlled during power operation nor is it an initial condition of DBA or transient analysis that assumes the failure of or presents a challenge to the integrity of a fission products barrier. There are no requirements for a structure, system, or component included in this LCO, which are a part of the primary success path, which actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to a fission product barrier. Spent fuel shipping cask movement is not modeled in the DCPP individual Plant Examination for power operation and there is no indication this would be identified as a significant risk if it were included in the PRA models.

ATTACHED PAGES:

Encl 3A	7
Encl 3B	8

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
12-09	LG	The requirement for an OPERABLE emergency power source for an OPERABLE FHBVS train is moved to the Bases. This is consistent with NUREG-1431.
12-10	LS9	The "within 31 days after removal" requirement for completion of laboratory analyses is deleted. This requirement is not contained in the ITS nor is it contained in the regulatory guide or ANSI standards.
12-11	A	The SR to measure [FHBVS] flow rate is not identified as a separate SR in the ITS because it is verified during the other in-place filter tests (see ITS 5.5.11 a. and b.). This change does not result in a change to technical requirements.
12-12	LS26	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
14-01	LS11	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
14-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
14-03	LS12	This change would delete the ACTION requirements to suspend all other movement of fuel assemblies and crane operations. This change is consistent with NUREG-1431.
14-04	LS13	Deletes the action statement requirement to verify spent fuel pool boron concentration every 8 hours while ACTION is being taken to relocate noncomplying spent fuel assemblies from Region 2 to Region 1. This change is consistent with NUREG-1431.
14-05	LG	The requirement to keep records of the burnup analysis for all assemblies in Region 2 would be relocated to a licensee controlled document. This change is consistent with NUREG-1431, and moves requirements that do not meet the criteria for inclusion in the TS.
14-06		Not used.
14-07		Not used.
14-08		Not used.
14-09		Not used.
14-10	A	The Statement that 3.0.4 is not applicable is deleted. This is consistent with NUREG-1431. This change does not result in a change to technical requirements.
15-01	R	The requirement to empty the spent fuel exclusion zone area prior to any spent fuel shipping cask handling operations is relocated to a Licensee controlled document.

Insert →

Q 3.9-25

DOC 15-01-R

This justification shows that this LCO provide assurance that the spent fuel shipping cask will not inadvertently be dropped on irradiated fuel in the spent fuel pool. The dose consequences of the postulated accident addressed by the restriction of this specification are less then the 10 CFR 100 guidelines. This specification does not contain any requirements for installed instrumentation used to indicate an abnormality in the reactor coolant boundary. It does not contain requirements for process variables, design features, or operating restriction that are monitored or controlled during power operation nor is it an initial condition of DBA or transient analysis that assumes the failure of or presents a challenge to the integrity of a fission products barrier. There are no requirements for a structure, system, or component included in this LCO, which are a part of the primary success path, which actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to a fission product barrier. Spent fuel shipping cask movement is not modeled in the DCPP individual Plant Examination for power operation and there is no indication this would be identified as a significant risk if it were included in the PRA models.

This proposed TS revision relocates requirements, which do not meet the TS criteria in 10CFR50.36 (c) (2) (ii), to documents with established control programs. This regulation addresses the scope and purpose of the TS. In doing so, It sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. Relocation of these requirements allow the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS. An evaluation of the applicability of these criteria to this specification is provided in Attachment 21.

To ensure an appropriate level of control, these requirements will be relocated to 1) documents that are subject to the provisions of 10CFR50.59, 2) other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in FSAR which is controlled by 10CFR50.54a), or3) to programs that are controlled via the Administrative Controls section of the improved TS. The identification of the specific licensee controlled document containing this requirement is provided in Enclosure 3B of the conversion submittal.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure the limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
14-04 LS13	Deletes the ACTION statement requirement to verify spent fuel boron concentration every 8 hours while action is being taken to relocate noncomplying spent fuel assemblies from Region 1 to Region 2.	Yes	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes
14-05 LG	The requirement to keep records of the burnup analysis for all assemblies in Region 1 and 2 would be relocated to a licensee controlled document.	Yes, to the Bases	No, CPSES does not have this specification in CTS 3/4.9	Yes, to USAR	Yes, to FSAR
14-06	Not Used	N/A	N/A	N/A	N/A
14-07	Not Used	N/A	N/A	N/A	N/A
14-08	Not Used	N/A	N/A	N/A	N/A
14-09	Not Used	N/A	N/A	N/A	N/A
14-10 A	The statement that 3.0.4 is not applicable would be removed.	Yes	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes
15-01 R	The requirement to empty the spent fuel exclusion zone area prior to any spent fuel shipping cask handling operations is relocated to Licensee controlled document.	Yes, see Attachment 21page 27— relocated to the FSAR.	No, CPSES does not have this specification in CTS 3/4.9	No, not in CTS	No, not in CTS Q3.9-25

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC 3.9-ED **APPLICABILITY:** DC

REQUEST:

Various changes that do not impact the technical content of the submittal or other FLOG members.

Changes are noted with DC 3.9-ED in the margin and noted below:

1) Enclosure 2, page 3/4 9-11, text deleted per 10-03-LS18.
2) Enclosure 4, Table of Contents, page 46, and 47 - LS-21 is not applicable to DCPP.
3) Enclosure 5B, page B 3.9-2, replaces the phrase "high count rate alarm" in the Applicable Safety Analysis section with "visual count rate instrumentation."
4) Enclosure 5B, page B 3.9-4, replaces the phrase "high count rate alarm" in the Applicable Safety Analysis section with "visual flux indication."
5) Enclosure 5B, page B 3.9-4, replaces "....visual indication and at least one of the two monitors must provide an audible alarm and count rate function in the Control Room. Therefore, with no audible alarm and count rate functions from at least one monitor, both monitors are inoperable," in the LCO Section with "....visual indication and alarm and at least one of the two monitors must provide an audible count rate indication in the Control Room. Therefore, with no audible count rate indication from at least one monitor, both monitors are inoperable until the audible indication is restored to the operable monitor. ACTION A must also be entered with no audible count rate indication in the control room."
6) Enclosure 5B, page 3.9-16 includes a comment that DCPP meets intent of 1971 GDC 34.
7) Enclosure 5B, page 3.9-19 sentence deleted per JFD 3.9-10 should also be deleted in Bases.

ATTACHED PAGES:

Encl. 2	3/4 9-11
Encl. 4	Table of Contents, 46, 47
Encl. 5B	B 3.9-2, B 3.9-4, B 3.9-16, B 3.9-19

3/4.9.10 WATER LEVEL - REACTOR VESSEL

FUEL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

DC-39 ED

APPLICABILITY: During movement of irradiated fuel assemblies within containment ~~when the reactor pressure vessel pressure contains irradiated fuel assemblies.~~ 10-03-LS18

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of irradiated fuel assemblies within the ~~reactor pressure vessel~~ containment. ~~The provisions of Specification 3.0.3 are not applicable~~. 10-03-LS18

SURVEILLANCE REQUIREMENTS

4.9.10.1 The water level shall be determined to be at least its minimum required depth ~~within 2 hours prior to the start of and~~ at least once per 24 hours~~, thereafter during movement of irradiated fuel assemblies within containment.~~ 10-02-LS22

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

PAGE

DC 3.9-ED

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Not Applicable to DCPP

NSHC LS21
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431 and TSTF-23, Rev. 2, the requirements related to indication provided by the source range detectors would be deleted from the Limiting Condition for Operation (LCO). In accordance with TSTF-23, Rev. 2, the requirements for visual indication for plants that do not rely on a boron dilution analysis would be discussed in the Bases; while the requirements for audible indication would be eliminated as a Technical Specification requirement. In MODE 6, the source range monitors are required for indication only and there are no precise setpoints associated with these instruments. In this capacity, the source range instrumentation is typically used to read a relative change in count rate. The source range instrumentation is monitored for significant changes in count rate which are important to evaluate the change in core status. The accepted convention for defining criticality does not require precise or specific setpoints or indication, but only requires verification of a slowly increasing count rate.

Consistent with NUREG-1431, Rev. 1, the Technical Specification requirements consist of maintaining two source range neutron flux monitors OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> 1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
>
> 2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
>
> 3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   During REFUELING, the source range monitors are designed to provide visual and audible indication of neutron count rate to plant operators. The proposed move of audible indication for these channels to the Bases would not affect the availability of visual or audible indication. There are no alarms, interlocks, or trip setpoints associated with these channels that are required to be OPERABLE during MODE 6. Thus, the proposed change would have no significant effect on the probability of an accident occurring. In addition, in MODE 6 the source range instruments provide no automatic actuation function used for mitigation of accidents, and the change would have no effect on the outcome of an accident. Therefore, there would be no significant increase in the probability or consequences of a previously evaluated accident as a result of making the proposed change.

DC 3.9-ED

NSHC LS21
(Continued)

Not Applicable to DCPP

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change would not create the possibility of a new or different accident, because the plant or its systems would not be operated any differently. Visual indication in the trend of reactivity would remain available to the operating staff. Therefore, there would be no operational changes to contribute to the possibility of a new accident resulting from the proposed change. Therefore, this change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The margins of safety in question are those involved with preventing criticality during REFUELING operations. The monitors provide visual indication of neutron count rate, and, therefore, provide assurance that the core reactivity is being maintained. However, reactivity is maintained primarily by the requirements of ITS 3.9.1 which assure that the boron concentration in refueling water is within limit and that dilution of the boron will not occur. Thus the neutron monitoring channels provide further assurance that criticality will not occur. Therefore, moving of audible indication for the source range neutron monitoring channels to the Bases during MODE 6 would have an insignificant effect on margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS21" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

BASES

APPLICABLE SAFETY ANALYSIS

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

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

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

The limiting boron dilution accident analyzed occurs in MODE 5 ~~3~~ (Ref. 2). ~~A detailed discussion of this event is provided in Bases B 3.1.2, "SHUTDOWN MARGIN (SDM) — T_{avg} ≤ 200°F." Boron dilution accidents are precluded in MODE 6 by isolating potential dilution flow paths — See LCO 3.9.2, "Unborated Water Source Isolation Valves."~~ It is based upon a maximum dilution flow of 300 g.p.m. and prompt identification and operation preclude the event from proceeding to a boron dilution accident. Prompt identification is assured through audible count rate instrumentation, a ~~high~~ count rate ~~alarm~~ and a high source range flux level alarm.

visual instrumentation

DC 3.9-ED

The RCS boron concentration satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii).

THE SAME MINIMUM BORON CONCENTRATION, IS REQUIRED TO BE MAINTAINED

Q3.9-1a

LCO

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

WHILE IN MODE 6. ADDITIONALLY, WHEN THE RCS IS FLOODED AND OPEN TO

Q3.9-1a

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a k_{eff} ≤ 0.95. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~— T_{avg} > 200°F," and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) — T_{avg} ≤ 200°F,"~~ LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical. A Note is added to the applicability to assure that MODE 6 cannot be entered unless boron concentration limits are met.

(Continued)

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed source range neutron flux monitors are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range (source range drawer) covers six decades of neutron flux ~~(1E+6 cps)~~(1A to 1E+6 cps) with a ~~[5]~~±3% instrument (DC 3.9-ED) accuracy. The detectors also provide continuous visual indication in the control room and an audible alarm and count rate to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1.

The Gamma-Metrics neutron flux monitors (N-51 and N-52) are designed in accordance with Regulatory Guide 1.97. The wide range neutron flux monitors in this system provide indication of neutron flux from reactor shutdown to reactor full power level (source range through power range). The wide range monitors (1E-8 to 1E+2% power) provide continuous visual (DC39-ED) indication in the control room to allow operators to monitor core flux. The narrow range monitors (1E-1 to 1E+5 cps) provides indication of neutron flux to the hot shut down panel and control room by way of the plant process computer (PPC).

APPLICABLE SAFETY ANALYSIS

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly. ~~The need for a safety analysis for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2, "Unborated Water Source Isolation Valves."~~ Prompt identification is required to assure sufficient time for operator action to preclude the event from proceeding to a Boron Dilution Accident. Prompt identification is assured through audible count rate ~~instrumentation~~ indication, a ~~high count rate alarm~~ and a high source range flux level alarm in the control room, visual flux indication (DC-3.9-ED)

The source range neutron flux monitors satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii).

LCO

(DC-39-ED)

This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each monitor must provide visual indication and at least one of the two monitors must provide an audible alarm and count rate ~~functions~~ indication in the Control Room. Therefore, with no audible alarm and count rate ~~functions~~ indication from at least one monitor, both monitors are inoperable, until the audible indication is restored to the operable monitor — Action A must also be entered with no audible count rate indication in the control room.

(Continued)

B 3.9 REFUELING OPERATIONS

B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level DC.3.9-ED

BASES (DCPP meets intent of this 1971 GDC)

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown 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 RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSIS

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a 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. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, ~~in order~~ to prevent this challenge.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in ~~the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii) as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat;

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

c. Indication of reactor coolant temperature. An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow

(Continued)

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

DC 3.9-ED

BACKGROUND

The movement of irradiated fuel assemblies ~~or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, or friction testing of individual control rods~~ within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3 ~~and acceptance in Reference 6~~

APPLICABLE SAFETY ANALYSIS

During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained well within allowable limits (Refs. 4 and 5 ~~and 6~~).

Refueling cavity water level satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii).

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 3.

(Continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC ALL-001 **APPLICABILITY:** DC

REQUEST:

LAs 119/117 and 118/116 were issued 7/13/97 and addressed CTS surveillance interval increases due to 24-month fuel cycles. These changes on pages affected by NRC comment numbers are indicated with "DC-ALL-001." These changes were previously submitted to the NRC in an errata to LAR 97-09 via DCL-98-003 (dated January 8, 1998). Other changes in the errata submittal are noted with "DC-ALL-002."

ATTACHED PAGES:

See notations on applicable pages for each comment number.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: DC ALL-003 **APPLICABILITY:** DC

REQUEST:

Diablo Canyon submitted the ITS conversion LAR two weeks after the other FLOG members. Technical reviews were being finalized which resulted in a change to Enclosure 3B (Conversion Comparison Table). This change was identified with "{ }" to ensure the difference between the other FLOG members was noted. Thus, DOC 12-12-LS26 applicability column for Diablo Canyon should read "No, maintaining CTS" in all FLOG submittals.

ATTACHED PAGES:

Encl 3B page 7

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
12-08 LS16	The proposed change would allow the ~~18~~ [24]-month testing of the [FHBVs] ability to maintain the required pressure differential between the building and the outside atmosphere to be performed on a STAGGERED TEST BASIS.	Yes	No, CPSES does not have this specification	Yes	Yes DC-ALL-001
12-09 LG	The requirement for an OPERABLE emergency power source for an OPERABLE FHBV train is moved to the Bases.	Yes	No, CPSES does not have this specification.	No	No
12-10 LS9	The "within 31 days after removal" requirement for completion of laboratory analyses is deleted.	Yes	No, CPSES does not have this specification in CTS 3/4.9.	Yes	Yes
12-11 A	The SR to measure [FHBVs] flow rate is not identified as a separate SR in the ITS because it is verified during the other in-place filter tests (see ITS 5.5.11 a. and b.)	Yes	No, CPSES does not have this specification in CTS 3/4.9.	Yes	Yes
12-12 LS26	This change establishes appropriate ACTIONS and Completion Times for Fuel Building pressure envelope degradation.	No, maintaining CTS	No, CPSES does not have this specification in CTS 3/4.9	Yes DC-ALL-003	No, maintaining CTS
14-01 LS11	This change deletes the restrictions on placing spent fuel assemblies into Region 2 of the spent fuel pool and changing storage locations designations from Region 1 to Region 2.	No, Requirement not in CTS.	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes
14-02 M	This changes the Applicability from "Whenever irradiated fuel assemblies are in the spent fuel pool" to "Whenever any fuel assembly is in Region 2 of the spent fuel pool."	No, already in CTS.	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes
14-03 LS12	This change would delete the ACTION requirements to suspend all other movement of spent fuel and crane operations.	Yes	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: TR 3.9-001 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

It is purposed to "not incorporate" TSTF-21 but rather add a statement to the Bases to state our current plant practice. It would be summarized by the following statement: "An operable RHR loop must be capable of being realigned to provide an operable flow path."

ATTACHED PAGES:

Enc. 5A	Traveler Status Sheet
Encl 5B	B 3.9-17

Industry Travelers Applicable to Section 3.9

Approved by NRC.

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-20	Incorporated	3.9-2	Approved by NRC
TSTF-21 Rev. 1	Incorporated Not	None	Change made to Bases for 3.9.6 TR 3.9-001
TSTF-22	Not incorporated	N/A	Changes not applicable for the specific plant application.
TSTF-23, Rev. 2 Rev. 3	Incorporated	3.9-3 Approved by NRC.	Traveler bracketed ITS 3.9.2 and revised the Bases for ITS 3.9.3 (DCPP maintaining CTS). TR 3.9-003
TSTF-51	Not incorporated	N/A	Minimal impact on plant specific applications.
TSTF-68, Rev. 1	Not incorporated	N/A	Similar changes were incorporated into the ITS based on current licensing basis. See change description 3.9-1. (Not Appl to DCPP)
TSTF-92, Rev. 1	Not incorporated	N/A	The proposed changes did not significantly affect current surveillance practices to warrant inclusion.
TSTF-96, Rev. 1	Incorporated	3.9-4	Approved by NRC. TR 3.9-003
WOG-63	Not incorporated	N/A	
WOG-76	Incorporated	3.9-11	Containment penetrations allowed to be open under administrative control.
TSTF-136	Incorporated	N/A	Approved by NRC
TSTF-139	Incorporated	N/A	
TSTF-153	Incorporated	N/A	
TSTF-272	Incorporated	3.9-15	Q 3.9-1a

TR 3.9-002

Editorial change to 3.9.5 unnecessary where as 3.4 and 3.5 were revised to match 3.9.5 wording.

BASES

An operable RHR loop must be capable of being realigned to provide an operable flow path.

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. ~~One or both RHR pumps maybe aligned to the RWST to support filling the refueling cavity or for performance of required testing (Ref. 2)~~.

Q 3.9.G-1 Redline→

TR 3.9-001

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level." A Note is added to the applicability to assure that MODE 6 operation with water level <23 ft is not permitted unless two RHR loops are operable.

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR 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 RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. ~~Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS, because all of the unborated water sources are isolated~~. The suspension of any operation involving a reduction in Reactor Coolant Boron Concentration will reduce the likelihood of Boron Stratification in the RCS. DC-ALL-002

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop

(Continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: TR 3.9-002 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

TSTF-136 is changed to "Approved by NRC." Revise Comment for TSTF-153 to read "Editorial change to 3.9.5 is unnecessary whereas 3.4 and 3.5 were revised to match 3.9.5 wording."

ATTACHED PAGES:

Encl 5A Traveler Status Sheet

Industry Travelers Applicable to Section 3.9

Approved by NRC.

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-20	Incorporated	3.9-2	Approved by NRC
TSTF-21 Rev. 1	Incorporated Not	None	Change made to Bases for 3.9.6 TR 3.9-001
TSTF-22	Not incorporated	N/A	Changes not applicable for the specific plant application.
TSTF-23, Rev. 2 Rev. 3	Incorporated	3.9-3 Approved by NRC.	Traveler bracketed ITS 3.9.2 and revised the Bases for ITS 3.9.3 TR 3.9-003 (DCPP maintaining CTS).
TSTF-51	Not incorporated	N/A	Minimal impact on plant specific applications.
TSTF-68, Rev. 1	Not incorporated	N/A	Similar changes were incorporated into the ITS based on current licensing basis. See change description 3.9-1. (Not Appl to DCPP)
TSTF-92, Rev. 1	Not incorporated	N/A	The proposed changes did not significantly affect current surveillance practices to warrant inclusion.
TSTF-96, Rev. 1	Incorporated	3.9-4	Approved by NRC. TR 3.9-003
WOG-63	Not incorporated	N/A	
WOG-76	Incorporated	3.9-11	Containment penetrations allowed to be open under administrative control.
TSTF-136	Incorporated	N/A	Approved by NRC
TSTF-139	Incorporated	N/A	
TSTF-153	Incorporated	N/A	
TSTF-272	Incorporated	3.9-15	Q 3.9-1a

TR 3.9-002

Editorial change to 3.9.5 unnecessary where as 3.4 and 3.5 were revised to match 3.9.5 wording.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: TR 3.9-003 **APPLICABILITY:** DC

REQUEST:

The traveler status page is updated to reflect the following: TSTF-23, Revision 3, is "Approved by NRC" and TSTF-96, Revision 1, has been issued.

ATTACHED PAGES:

Encl 5A Traveler Status Sheet

Industry Travelers Applicable to Section 3.9

Approved by NRC.

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-20	Incorporated	3.9-2	Approved by NRC
TSTF-21 Rev. 1	Incorporated Not	None	Change made to Bases for 3.9.6 TR 3.9-001
TSTF-22	Not incorporated	N/A	Changes not applicable for the specific plant application.
TSTF-23, Rev. 2 Rev. 3	Incorporated	3.9-3 Approved by NRC.	Traveler bracketed ITS 3.9.2 and revised the Bases for ITS 3.9.3 (DCPP maintaining CTS). TR 3.9-003
TSTF-51	Not incorporated	N/A	Minimal impact on plant specific applications.
TSTF-68, Rev. 1	Not incorporated	N/A	Similar changes were incorporated into the ITS based on current licensing basis. See change description 3.9-1. (Not Appl to DCPP)
TSTF-92, Rev. 1	Not incorporated	N/A	The proposed changes did not significantly affect current surveillance practices to warrant inclusion.
TSTF-96, Rev. 1	Incorporated	3.9-4	Approved by NRC. TR 3.9-003
WOG-63	Not incorporated	N/A	
WOG-76	Incorporated	3.9-11	Containment penetrations allowed to be open under administrative control.
TSTF-136	Incorporated	N/A	Approved by NRC
TSTF-139	Incorporated	N/A	
TSTF-153	Incorporated	N/A	
TSTF-272	Incorporated	3.9-15	Q 3.9-1a

TR 3.9-002

Editorial change to 3.9.5 unnecessary where as 3.4 and 3.5 were revised to match 3.9.5 wording.

JLS CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS

CTS 5.0 - DESIGN FEATURS
ITS 4.0 - DESIGN FEATURES

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION AND LICENSEE INITIATED ADDITIONAL CHANGES

INDEX OF ADDITIONAL INFORMATION

ADDITIONAL INFORMATION NUMBER	APPLICABILITY	ENCLOSED
4.3.2	DC, CP, WC, CA	YES
CP 4.0-002	CP	NA

JOINT LICENSING SUBCOMMITTEE METHODOLOGY FOR PROVIDING ADDITIONAL INFORMATION

The following methodology is followed for submitting additional information:

1. Each licensee is submitting a separate response for each section.

2. If an RAI does not apply to a licensee (i.e., does not actually impact the information that defines the technical specification change for that licensee), "NA" has been entered in the index column labeled "ENCLOSED" and no information is provided in the response for that licensee.

3. If a licensee initiated change does not apply, "NA" has been entered in the index column labeled "ENCLOSED" and no information is provided in the response for that licensee.

4. The common portions of the "Additional Information Cover Sheets" are identical, except for brackets, where applicable (using the same methodology used in enclosures 3A, 3B, 4, 6A and 6B of the conversion submittals). The list of attached pages will vary to match the licensee specific conversion submittals. A licensee's FLOG response may not address all applicable plants if there is insufficient similarity in the plant specific responses to justify their inclusion in each submittal. In those cases, the response will be prefaced with a heading such as "PLANT SPECIFIC DISCUSSION."

5. Changes are indicated using the redline/strikeout tool of WordPerfect or by using a hand markup that indicates insertions and deletions. If the area being revised is not clear, the affected portion of the page is circled. The markup techniques vary as necessary, based on the specifics of the area being changed and the complexity of the changes, to provide the clearest possible indication of the changes.

6. A marginal note (the Additional Information Number from the index) is added in the right margin of each page being changed, adjacent to the area being changed, to identify the source of each change.

7. Some changes are not applicable to one licensee but still require changes to the Tables provided in Enclosures 3A, 3B, 4, 6A, and 6B of the original license amendment request to reflect the changes being made by one or more of the other licensees. These changes are not included in the additional information for the licensee to which the change does not apply, as the changes are only for consistency, do not technically affect the request for that licensee, and are being provided in the additional information being provided by the licensees for which the change is applicable. The complete set of changes for the license amendment request will be provided in a licensing amendment request supplement to be provided later.

JOINT LICENSING SUBCOMMITTEE METHODOLOGY FOR PROVIDING ADDITIONAL INFORMATION (cont)

8. The item numbers are formatted as follows:

   [Source] [ITS Section]-[nnn]

   Source =
   - Q - NRC Question
   - CA - AmerenUE
   - DC - PG&E
   - WC - WCNOC
   - CP - TU Electric
   - TR - Traveler

   ITS Section = The ITS section associated with the item (e.g., 3.3). If all sections are potentially impacted by a broad change or set of changes, "ALL" is used for the section number.

   nnn = a three digit sequential number

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 4.3.2 **APPLICABILITY:** DC, CP, WC, CA

REQUEST:

4.3.2 DRAINAGE

The ISTS for this section is as follows:

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation [23 ft].

Comment: This section for all four FLOG plants are the same as the ISTS above. Each has a plant specific elevation for the bracket instead of the 23 ft. They are: Callaway 2040 ft; Wolf Creek 2040 ft; Comanche Peak 845 ft; Diablo Canyon 133 ft. Provide explanations that these elevation levels are at 23 ft above the spent fuel in the pool.

FLOG RESPONSE: General Discussion - The elevation reported by each of the FLOG plants is based upon reference elevations used in the plant's design and construction. The design feature (plant specific elevation) of CTS 5.6.2 is provided in accordance with RG 1.13, "Fuel Storage Facility Design Basis," C.6. Per RG 1.13, this design feature is required to assure that an inadvertent drain down of the Spent Fuel Pool will not "cause the fuel to be uncovered." This RG also states that loss of inventory from the Spent Fuel Storage Pool "could cause overheating of spent fuel and resultant damage to cladding integrity." The margin of coverage above the spent fuel is in excess of the 10 feet stated in the Standard Review Plan, NUREG-0800, Section 9.1.3.III.1.e.

Plant Specific Discussion: For DCPP, the top of the active fuel in the spent fuel pool is at the 111'-5" elevation. Consequently, the 133 foot elevation is 21 feet 7 inches over the top of the active fuel. This elevation is based upon the location of the Spent Fuel Pool Cooling system suction pipe. The discharge pipe anti-siphon device is located at 136'-6. This is consistent with SER 0 (10/16/74) which states that the DCPP Spent Fuel Pool meets RG 1.13 and GDC 61 and "therefore the design is acceptable." This also is consistent with the FSAR, Section 9.1.3.2, which states that the bottom of the pump suction line is "4 feet below the normal Spent Fuel Pool level" (or 137 feet 8 inches minus 4 feet) and CTS 5.6.2, which rounded 133 feet 8 inches to elevation 133 feet.

ATTACHED PAGES:

None