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*. 13-06-A

ACTION:

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

1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and within 4 hours ~~immediately~~ 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-LS

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours, or

3. Be in HOT STANDBY within the next 6 hours. 13-04-M

b. With more than one digital rod position indicator per group ~~bank~~ inoperable either: 13-05-A

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 ~~immediately~~ after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, and 13-03-LS

~~1.b) Place the control rods under manual control and limit rod motion as specified above, and~~ ~~13-09-LS~~

Q 3.1-19

~~1.c) Monitor and record Reactor Coolant System average temperature (T_{avg}) at least once per hour, and~~ ~~13-09-LS~~

1.b) ~~1.d)~~ Restore the digital rod position indicators to OPERABLE status within 24 hours such that a maximum of one digital rod position indicator per group ~~bank~~ is inoperable, or 13-05-A

(Retain)

2. Be in HOT STANDBY within the next 6 hours.

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 Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or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

3. Be in HOT STANDBY within the next 6 hours. 13-04-M

* INSERT 3/4 1-17 13-06-A

CHANGE NUMBER	NSHC	DESCRIPTION
		plant shutdown to Mode 3 requirement would be required in one less hour.
13-05	A	This proposed change would involve retaining an action statement, currently in the plant TS, that permits continued POWER OPERATION with more than one digital rod position indicator per group inoperable. This is in accordance with the current licensing basis of the plant.
13-06	A	Consistent with NUREG-1431, Rev. 1, a separate condition entry allowance is permitted in current TS 3.1.3.2 for each inoperable rod position indicator and each demand position indicator. This is an administrative change since the Required Actions address each rod or bank with inoperable indication separately [and Action b addresses the condition of multiple inoperable DRPIs, up to a complete loss of digital position indication].
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 surveillance requirements 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. 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	LS-20	Not applicable to Callaway. See Conversion Comparison Table (Enclosure 3B).
13-09	~~LS-23~~ —	Current TS 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. TR-3.1-006

~~INSERT 3A-12~~ ~~Q 3.1-19~~

Not used. Q 3.1-19

INSERT 3A-12

~~The proposed change would delete the Actions to place control rods in manual and record RCS T_{avg} hourly if multiple DRPIs per group are inoperable. Multiple inoperable DRPIs, of themselves, have no impact on SDM in MODES 1 and 2 if the control rod positions are verified by alternate means (e.g., movable incore detectors). The requirement to place control rods in manual may not be appropriate in all situations and may be detrimental for load rejection transients unless operator action is assumed to simulate the rod control system in automatic. Accidents analyzed using the [Improved Thermal Design Procedure (ITDP)] assume that the control rods are in [automatic]. Automatic rod movement can accommodate a 10% load rejection. Placing rods in manual may impact the load rejection capability assumed when the P-9 setpoint was established at 50% RTP. The steam dump system can accommodate a 40% RTP load rejection and with the rod control system in automatic, a 50% RTP load rejection can be accommodated without a reactor trip. While manual operator action can be just as timely as automatic rod control, there is no need to have this limitation in the Technical Specifications. Corrective actions for excessive rod motion are covered under ITS 3.1.7 Condition C. The requirement to monitor and record T_{avg} hourly is unnecessary given the available indicators and alarms, e.g., $T_{avg} - T_{ref}$ deviation alarm, to alert operators to changing moderator conditions.~~ Q 3.1-19

Not used. Q 3.1-19

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
13-05 A	The proposed change would retain an action statement, currently in the plant TS, that permits continued POWER OPERATION with more than one digital rod position indicator per group inoperable.	No. See CN 13-08-LS-20.	No. See CN 13-08-LS-20.	Yes	Yes
13-06 A	The change would allow separate condition entry for each inoperable DRPI or each demand indicator.	No. See CN 13-08-LS-20.	No. See CN 13-08-LS-20.	Yes	Yes
13-07 M	The proposed modifications to the SR would verify agreement between digital and demand indicator systems prior to criticality after each removal of the reactor vessel head instead of every 12 hours. The Frequency change is based on traveler TSTF-89.	Yes	Yes	Yes	Yes
13-08 LS-20	Adds provision, from Callaway's current specifications as revised which, under certain conditions, would allow continued operation with more than one inoperable DRPI per group. ~~This is consistent with WOG traveler 73.~~ TSTF-234.	Yes Q3.1-20	Yes	No. Already in current TS.	No. Already in current TS. ~~IR-3.1-006~~
13-09 ~~LS-23~~ —	Current TS 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 71~~ TSTF-234.	~~No. Not in current TS.~~ N/A Not used.	~~No. Not in current TS.~~ N/A Q3.1-19	~~Yes~~ N/A	~~Yes~~ N/A ~~IR-3.1-006~~
14-01 R	Relocates current Specification 3.1.3.3 to licensee controlled document.	No - See Amendments 120/118. ~~Yes (see LAR 95-07 dated 10/4/95, DCL 95-222).~~	Yes. Relocated to TRM.	No. See OL Amendment No. 89.	No. See OL Amendment No. 103. DC-ALL-004
15-01 ~~R~~ —	~~The Rod Drop Time Specification 3.1.3.4 is relocated outside of the Technical Specifications. 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. This change is consistent with License Amendment Request 95-07 dated October 4, 1995 (DCL 95-222).~~ Not used.	~~Yes (see LAR 95-07 dated 10/4/95, DCL 95-222).~~ N/A	~~No. Not in current TS. See CN 15-02-A.~~ N/A	~~No. See OL Amendment No. 89.~~ N/A	~~No. See OL Amendment No. 103.~~ N/A DC-ALL-004

ENCLOSURE 4

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

CONTENTS

DELETE Q 3.1-19

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS-23
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change would delete the Actions to place control rods in manual and record RCS T_{avg} hourly if multiple DRPIs per group are inoperable, Actions b.1.b) and b.1.c) of LCO 3.1.3.2. Multiple inoperable DRPIs will have no impact on SDM in Modes 1 and 2 if the control rod positions are verified by alternate means (e.g., movable incore detectors). The requirement to place control rods in manual is not appropriate in all situations and may be detrimental for load rejection transients unless operator action is assumed to simulate the rod control system in automatic. Accidents analyzed using the [Improved Thermal Design Procedure (ITDP)] assume that control rods are in [automatic]. Automatic rod movement can accommodate a 10% load rejection. The requirement to monitor and record T_{avg} hourly is unnecessary given the available indicators and alarms, e.g., T_{avg} - T_{ref} deviation alarm, to alert operators to changing moderator conditions.

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. The reactivity transients analyzed in FSAR Section 15.4 will be unaffected

DELETE Q 3.1-19

NSHC LS-23
(continued)

since rod position will be ascertained to be consistent with those analyses. 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. This change 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 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-23" 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.

Industry Travelers Applicable to Section 3.1

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-9, Revision 1	Incorporated	3.1-1	NRC approved.
TSTF-12, Revision 1	Incorporated	3.1-15	NRC approved. ITS Special Test Exception 3.1.10 is retained and re-numbered as 3.1.8, consistent with this traveler and TSTF-136.
TSTF-13, Revision 1	Incorporated	3.1-4	NRC approved.
TSTF-14, Revision ~~3~~ 4	Incorporated	3.1-13	NRC approved. TR-3.1-005
TSTF-15, Revision 1	Incorporated	NA	NRC approved.
TSTF-89	Incorporated	3.1-8	NRC approved.
TSTF-107, Rev. 1	Incorporated	3.1-6	Q 3.1-15
TSTF-108, Revision 1	~~Not~~ Incorporated	~~NA~~ 3.1-21	~~Not~~ NRC approved. ~~as of traveler cut-off date.~~ TR-3.1-001
TSTF-110, Revision ~~1~~ 2	Incorporated	3.1-10	NRC approved. TR-3.1-004
TSTF-136	Incorporated	3.1-9, 3.1-15	NRC approved. TR-3.1-006
TSTF-141	Not incorporated	NA	Disagree with change; traveler issued after cut-off date.
TSTF-142	~~Not~~ Incorporated	~~NA~~ 3.1-22	NRC approved. ~~Traveler issued after cut-off date.~~ TR-3.1-003
~~TSTF-234~~ ~~WOG-75, Rev. 1~~	~~Incorporated~~	~~3.1-7~~	Q 3.1-19 ~~TR-3.1-006~~
WOG-105	Incorporated	3.1-16	

3.1 REACTIVITY CONTROL SYSTEMS

~~3.1.8~~ 3.1.7 Rod Position Indication

LCO ~~3.1.8~~ 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE. B-PS

APPLICABILITY: MODES 1 and 2.

ACTIONS

NOTE

Separate Condition entry is allowed for each inoperable rod position indicator ~~per group~~ and each demand position indicator. ~~per bank.~~ ~~3.1-7~~ ED Q3.1-19

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

B-PS 3.1-12

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
	INSERT 3.1-18 Q 3.1-19		
B. More than one DRPI per group inoperable for one or more groups.	~~B.1~~ B.3 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors.	Once per 8 hours	3.1-7 3.1-12
	AND		
	~~B.2~~ B.4 Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours	
~~B.~~C. One or more rods with inoperable DRPIs ~~position indicators~~ have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	~~B.~~C.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors.	4 hours	B 3.1-17 3.1-12
	OR		
	~~B.~~C.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours	

(continued)

REQUIRED ACTION	COMPLETION TIME
B.1 Place the control rods under manual control.	Immediately
AND	
B.2 Monitor and record RCS T_{avg}.	Once per 1 hour
AND	

BASES

ACTIONS

A.1 (continued)

simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3).

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1, ~~and~~ B.2, B.3, and B.4 Q 3.1-19

Placing the Rod Control System in manual assures unplanned rod motion will not occur. Together with the

When more than one DRPI per group fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. ~~The~~ indirect position determination available via movable incore detectors will minimize the potential for rod misalignment.

, this

The position of the rods may be determined indirectly by use of the movable incore detectors. The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the nonindicating rods have not been moved. Verification of RCCA position once per 8 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the DRPI system to operation while avoiding the plant challenges associated with a shutdown without full rod position indication (Ref. 4).

INSERT B 3.1-48

(continued)

The Immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition. Monitoring and recording reactor coolant system T_{avg} help to assure that significant changes in power distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

CHANGE
NUMBER JUSTIFICATION

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 traveler TSTF-107.

3.1-7 This change to the ISTS would incorporate, into ITS 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 Digital Rod Position Indicator per rod group inoperable. The Action Statement specifies additional required actions beyond those applicable to the condition of one DRPI 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.~~ Q 3.1-19 Q 3.1-19 TR-3.1-006

3.1-8 The Frequency for ITS SR 3.1.7.1 for comparing DRPI and group demand position 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 traveler 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 ISTS LCO 3.1.1 and ISTS LCO 3.1.2. Differences above and below 200°F will be addressed in the COLR. Subsequent sections have been re-numbered. TR-3.1-006

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 moved 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, moving this detail is acceptable and is consistent with traveler 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 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.

3.1-13 This change adds an LCO requirement and SR to MODE 2 Physics Tests Exceptions 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

TECH SPEC 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 TR-3.1-006
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.~~ Not used.	~~Yes~~ ~~No - Maintaining ISTS wording.~~ N/A	~~Yes~~ N/A	~~No - Maintaining ISTS wording.~~ N/A	~~No - Maintaining ISTS wording.~~ N/A ~~DC 3.1-002~~ Q 3.1-4
3.1-3	Not used. ~~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."~~	~~No~~ N/A	~~No~~ N/A	~~Yes~~ N/A	~~No~~ N/A Q 3.1-23
3.1-4	In accordance with industry traveler TSTF-13, ~~Rev. 1,~~ ISTS 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	Yes	Yes	Yes TR-3.1-006
3.1-5	Per current TS [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 traveler 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 improved TS 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 Q 3.1-19

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: CA-3.1-005 **APPLICABILITY:** CA

REQUEST: Revise ITS SR 3.1.3.2 Bases to correct a typographical error. The Note 1 SR limit is 300 ppm, not 200 ppm.

ATTACHED PAGES:

Attachment 7, CTS 3/4.1 – ITS 3.1
Enclosure 5B, page B 3.1-20

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.2 ~~3.1.4.2 and SR 3.1.4.3~~ (continued)

~~equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.~~ CA-3.1-005
3

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 effective full power days is sufficient to avoid exceeding the EOC limit. Q 3.1.G-1

b. 3. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is less negative ~~more positive~~ 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. Q 3.1.G-1

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. FSAR, Chapter 15. ← (redline) Q 3.1.G-1
3. WCAP-9272-P-A ~~9273-NP-A~~, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
4. ~~FSAR, Chapter 15.~~

ADDITIONAL INFORMATION COVER SHEET

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

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:

(original) 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. (supplement) DOC 2-06-A is also revised to be DOC 2-06-M because this change is more restrictive than the CTS.

(original) 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 JDF 3.2-17 are no longer used.

(supplement) As discussed in a telecon with the NRC staff on October 1, 1998, additional justification for the basis of the 24 hour surveillance frequency has been added to JFD 3.2-12.

Additionally, this item is related to Comment Number Q 3.2-7 for Callaway and Wolf Creek. No additional response is required for Comment Number Q 3.2-7.

ATTACHED PAGES:

Att. No. 8	CTS 3/4.2 - ITS 3.2
Encl. 6A	2, Insert 3.2-12

CHANGE NUMBER | JUSTIFICATION

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 exactly 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 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 $F^C_Q(Z)$ is increasing are required regardless of whether $F^C_Q(Z)$ is within its limits.

3.2-08 Consistent with 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 ISTS.

3.2-09 For consistency with current TS 3.2.4 and improved TS 3.3.1, condition D, the breakpoints for the applicability of the surveillances in the notes in improved TS 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. ~~This is an administrative change~~ that retains current TS requirements and is consistent with TSTF-241 Q 3.2-6

3.2-10 Consistent with 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.

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

3.2-12 ~~Not applicable to Callaway. See Conversion Comparison Table (Enclosure 6B).~~ Q 3.2-5 Q 3.2-7

See Insert 3.2-12

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 3.2-13 Q 3.2-4

INSERT 3.2-12

3.2-12 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 3.2.1.2. A flux map is taken after a power level increase greater than a specified amount to verify FQ is within limits and to provide assurance that FQ will remain within limits until the next required flux map is taken. Based on plant experience, the flux maps taken during power ascension provide a high degree of confidence that FQ will be within limits at the next power plateau. As such, the exact time period allowed for performance of the surveillance, after reaching equilibrium, is not a significant safety consideration. The proposed time (24 hours) is a reasonable time period for obtaining and evaluating a flux map and then completing the procedural steps associated with this surveillance. Further, the 24 hour time period provides a reasonable limit on the length of time that the plant can operate in an unconfirmed condition.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.13-6 **APPLICABILITY:** CA

REQUEST: ITS 3.4.13 Bases LCO a. (Callaway)

> **Comment**: The intent of the addition that leakage past instrumentation lines not being pressure boundary leakage is unclear. Is that leakage upstream of isolation valves? If it is, is there a line size limit and is this consistent with the description of pressure boundary in the FSAR and the definition in ITS Section 1.1?

FLOG RESPONSE (original): This Bases change refers to 3/8 inch tubing for instrument connections to ASME Class 1 fluid piping downstream of the root valves and to 1/8 inch core exit thermocouple sheaths. These instrument lines are not part of the reactor coolant pressure boundary (RCPB), as discussed in FSAR Table 3.2-1 Notes (9) and (10). As further stated in Sub-article NCA-1130(c), the scope of ASME Section III does not apply to instrument tubing and that tubing is not designed or specified to be part of the RCPB or provide a pressure retaining barrier. As discussed in FSAR Sections 9.3.4.2.3.5 and 15.6.5.2, normal charging can accommodate a 3/8 inch break and maintain normal pressurizer level such that the ECCS is not actuated. This Bases change does not refer to leakage upstream of instrument root valves. There is no conflict with ITS Section 1.1.

FLOG RESPONSE (supplement): As discussed with NRC on October 8, 1998, the Bases has been revised to provide additional discussion.

ATTACHED PAGES:

Attachment 10, CTS 3/4.4 - ITS 3.4
Enclosure 5B, page B 3.4-84

BASES (continued)

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB.

Q 3.4.Gen-1

LEAKAGE past seals[redacted] ~~and~~ gaskets[redacted] is not pressure boundary LEAKAGE.

INSERT B 3.4-84 Q 3.4.13-6

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of [redacted]identified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). [redacted] Violation of this LCO could result in continued degradation of a component or system.

Q 3.4.13-3

Instrumentation lines are 3/8 inch tubing for instrument connections to ASME Class 1 fluid piping downstream of the root valves and 1/8 inch core exit thermocouple sheaths. These instrument lines are not part of the reactor coolant pressure boundary (RCPB) nor do they provide a pressure retaining barrier. Normal charging can accommodate a 3/8 inch break and maintain normal pressurizer level such that the ECCS is not actuated.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: CA-3.5-002 **APPLICABILITY:** CA, CP, DC, WC

REQUEST (original): 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).

REQUEST (revised): Revise various additional ITS Bases regarding the correct application of Criterion 2 of 10CFR50.36(c)(2)(ii). These changes are consistent with the attachment to a May 9, 1988 letter from T.E. Murley (NRC) to R.A. Newton (WOG) entitled "NRC Staff Review of NSSS Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications."

1. Revise ITS 3.5.1 Bases to indicate that the Accumulators LCO, by virtue of its pressure, volume, and boron concentration limits, also satisfies Criterion 2 (initial conditions of accident analyses).

2. 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).

3. Revise ITS 3.6.7 Bases to indicate that the Recirculation Fluid pH Control (RFPC) System, by virtue of its TSP-C depth limit which ensures a minimum equilibrium sump pH of 7.1, also satisfies Criterion 2 (initial conditions of accident analyses). (Callaway only)

4. Revise ITS 3.7.6 Bases to indicate that the CST (and FWST for DCPP) LCO, by virtue of its water volume limit, also satisfies Criterion 2 (initial conditions of accident analyses).

ATTACHED PAGES:

Attachment 11, CTS 3/4.5 – ITS 3.5
Enclosure 5B, pages B 3.5-5 and B 3.5-30

Attachment 12, CTS 3/4.6 - ITS 3.6
Enclosure 5B, page B 3.6-53

Attachment 13, CTS 3/4.7 - ITS 3.7
Enclosure 5B, page B 3.7-44

BASES

Criterion 2 and

APPLICABLE SAFETY ANALYSES (continued)

The accumulators satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii). CA-3.5-002

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. 2) could be violated. CA-3.5-001

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above ~~2000~~ 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. 2) limit of 2200°F. CA-3.5-001

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed with power removed from the valve operators to isolate the accumulators from the RCS (Refs. 6 and 7). This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators. CA-3.5-001

Accumulator isolation is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature, as allowed by the P/T limit curves provided in the PTLR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

assumes that all control rods are out of the core. The limits on contained water volume and boron concentration of the RWST also ensure a minimum equilibrium sump pH of 7.1 for the solution recirculated within containment after a LOCA. This pH level minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The upper limit on boron concentration of ~~2200~~ 2500 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg ~~injection~~ recirculation is to avoid boron precipitation in the core following the accident.

In the containment backpressure portion of the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of ~~35~~ 37°F (Ref. 3). If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of 100°F is used in the small break LOCA analysis and containment OPERABILITY analysis. Exceeding this temperature will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA, and higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment. CA-3.5-003

Safety analyses assume a B-10 abundance of 19.9 a/o (Ref. 2). Administrative controls ensure that the reactivity insertion from the RWST reflects this assumption.

The RWST satisfies Criterion 2 and Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii). CA-3.5-002

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

(continued)

BASES

BACKGROUND (continued)

~~The Containment Spray System actuation signal opens the valves from the spray additive tank to the spray pump suctions or the containment spray pump start signal opens the valves from the spray additive tank after a 5 minute delay. The 28% to 31% NaOH solution is drawn into the spray pump suctions. The spray additive tank capacity provides for the addition of NaOH solution to all of the water sprayed from the RWST into containment. The percent solution and volume of solution sprayed into containment ensures a long term containment sump pH of ≥ 9.0 and ≤ 9.5. This ensures the continued iodine retention effectiveness of the sump water during the recirculation phase of spray operation and also minimizes the occurrence of chloride induced stress corrosion cracking of the stainless steel recirculation piping.~~

APPLICABLE SAFETY ANALYSES

The RFPC ~~Spray Additive~~ System is essential to the removal and retention of airborne iodine within containment following a DBA. CA-3.6-002

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value volume following the accident. The analysis assumes that ~~100%~~ more than 90% of containment is covered by the spray (Ref. 1).

The DBA response time assumed for the RFPC ~~Spray Additive~~ System is the same as for the Containment Spray System and is discussed in the Bases for LCO 3.6.6, "Containment Spray and Cooling Systems."

The DBA analyses assume that one train of the Containment Spray System~~/Spray Additive System~~ is inoperable. ~~and that the entire spray additive tank volume is added to the remaining Containment Spray System flow path.~~

The RFPC ~~Spray Additive~~ System satisfies ~~Criterion~~ Criteria 2 and 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36(c)(2)(ii). CA-3.5-002

LCO

The RFPC ~~Spray Additive~~ System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume ~~and concentration~~ of TSP-C ~~the spray additive solution~~ must be sufficient ~~to provide NaOH~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

other physical characteristics. Additional details regarding the design of the AFW system can be found in FSAR 10.4.9.

CA-3.5-002

The CST satisfies ~~Criterion~~ Criteria 2, 3, and 4 of ~~the NRC NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii).

LCO

To satisfy ~~accident~~ analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for four hours following a reactor trip from 102% RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. ~~In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line.~~

The required CST ~~level~~ contained water volume ~~required~~ is ~~equivalent to a usable volume of~~ ≥ 281,000 gallons, which is based on ~~holding the unit in MODE 3 for 4 hours, followed by~~ a cooldown to RHR entry conditions during a Station Blackout event. ~~at °F/hour.~~ This basis is established in Reference 2. ~~4 and exceeds the volume required by the accident analysis.~~

The OPERABILITY of the CST is determined by maintaining the tank ~~level~~ contained water volume at or above the minimum required volume. ~~level.~~

APPLICABILITY

In MODES 1, 2, and 3~~, and in MODE 4, when steam generator is being relied upon for heat removal,~~ the CST is required to be OPERABLE.

In MODES 4, 5, or 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST ~~level~~ contained water volume is not within limits, the OPERABILITY of the backup ESW supply should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup ~~feedwater~~ ESW supply must

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: CA-3.5-004 **APPLICABILITY:** CA

REQUEST: Clarify ITS SR 3.5.2.3 Bases to reflect OL Amendment No. 127 dated August 17, 1998 regarding ECCS pump venting.

ATTACHED PAGES:

Attachment 11, CTS 3/4.5 – ITS 3.5
Enclosure 5B, page B 3.5-20

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist 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 nonaccident 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

ECCS

RHR and SI

~~With the exception of the operating 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 by venting pump casings and accessible discharge piping high point vents ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. Accessible high point vents are those that can be reached without hazard or high radiation dose to personnel. This will also prevent water hammer, pump cavitation, and pumping of noncondensible 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.

Q3.5.6-1

CA-3.5-004

CA-3.5-004

INSERT B 3.5-20

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other

(continued)

INSERT FOR B 3.5-20 CA-3.5-004

The design of the centrifugal charging pump is such that significant noncondensible gases do not collect in the pump. Therefore, it is unnecessary to require periodic pump casing venting to ensure the centrifugal charging pumps will remain OPERABLE.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: NR 5.0-001 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: The NRC requested the following:

For the following plants (and CTS sections), the applications identify the CTS requirements are being relocated to the FSAR: CW (6.2.3, ISEG; 6.5, review and audit; 6.10.1, record retention); CP (none); DC (6.10.1, record retention); and WC (6.2.3, ISEG; 6.5, review and audit; 6.8.2.3, procedure changes; 6.10.1, record retention). We discussed relocations to the QA plan with Ray Smith (QA branch) several weeks ago. The staff needs to have the licensees identify that these requirements are going to the QA plan and thus controlled by 50.54(a). The DOCs for relocating the above CTS sections are 1-04-LG and 3-09-LG. These DOCs only state the relocation is to the FSAR. The relocation should be to the QA plan.

FLOG RESPONSE: Enclosures 3A and 3B have been updated to reflect the location of the subject relocated items.

ATTACHED PAGES:

Attachment 18, CTS 6.0 - ITS 5.0
Enclosure 3A, page 8
Enclosure 3B, pages 1 and 6

CHANGE NUMBER	NSHC	DESCRIPTION
03-06	A	CTS [6.9.1.7], "Annual Radioactive Effluent Release Report" and CTS [6.14.c] is revised consistent with NUREG-1431, Rev. 1, to delete the term "Annual" and modify the submittal date. This change provides a reference to 10 CFR 50.36a since 10 CFR specifies that the report must be submitted annually and include the results from the previous 12 months of operation.
03-07	A	CTS [6.9.1.6], "Annual Radiological Environmental Operating Report" is revised to include specific details concerning the contents of the report. This change is consistent with NUREG-1431, Rev. 1.
03-08	A	CTS Specification [6.9.1.8, 6.9.1.9 and 6.9.2] are revised to delete the reference to submittal location for the monthly report, core operating limits report and special reports. The requirements related to report submittal are contained in 10 CFR. Since conformance to 10 CFR is a condition of the license, specific identification of this requirement in the TS would be duplicative and is not necessary. Since the plant requirements remain the same, the change is considered an administrative change. This change is consistent with NUREG-1431, Rev. 1. NR 5.0-001
03-09 a licensee controlled document.	LG	The record retention requirements are moved to ~~the FSAR and implementing procedures.~~ The removal of this detail from the CTS is consistent with NUREG-1431. The requirement for retention of records related to activities affecting quality is contained in 10 CFR 50, Appendix B, Criteria XVII and other sections of 10 CFR 50 that are applicable to the plant (i.e., 50.71, etc.). Post-completion review of records does not directly assure operation of the facility in a safe manner, as the activities described in the documents have already been performed. By retaining these requirements in ~~plant~~ a ~~procedures and~~ licensee controlled documents, any changes in these record retention requirements will be adequately controlled under the provisions of 10 CFR ~~50.59~~ 50.54(a) and the applicable regulations.
03-10	LG	The Radiation Protection Program is moved to the FSAR consistent with NUREG-1431. This program requires procedures to be prepared for personnel radiation protection consistent with 10 CFR Part 20. These procedures are for the protection of nuclear plant personnel and have no impact on nuclear safety or the health and safety of the public. Requirements to have procedures to implement 10 CFR Part 20 are contained in 10 CFR 20.1101(b). Periodic review of these procedures is

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 A	The "Responsibility" section is revised to be consistent with current plant practice. The requirement to issue a management directive annually (i.e., control room command function) is deleted. The TS already adequately defines the function, and therefore, the management directive is redundant.	Yes	Yes	Yes	Yes
01-02 A	The "Plant/Unit Staff" section is revised consistent with current plant practice. Sections are revised to reflect the shift crew composition table removal (if applicable), non-licensed personnel, and changes made to the section to be on a unit basis vs. plant basis. Various editorial changes are made to accomplish the removal of the table and revisions to be consistent with current plant practice.	Yes	No - CTS already incorporates changes	Yes	Yes
01-03 A	The requirement for an SRO to be present during fuel handling and to supervise all core alterations is not retained in ITS. This requirement is deleted. This requirement essentially duplicates the regulation in 10 CFR 50.54(m)(2)(iv).	Yes	No - Deleted per CTS Amendment 50/36	Yes	Yes NR 5.0-001
01-04 LG	The details of the review and audit, the independent safety engineering group and training functions are moved from the CTS. Those items not specifically covered by a regulation are moved to licensee controlled documents; otherwise the requirements are deleted.	No - Deleted per CTS Amendment 117/115	No - Deleted per CTS Amendment 50/36	Yes, move to USAR (training functions) and QA Plan in Chapter 17 of the USAR.	Yes, moved to ~~FSAR and OQAM~~ QA Plan in Chapter 17 of the FSAR. Review and audit deleted per Amendment 107.
01-05 A	The requirement for the presence of an RO or an SRO in the control room is deleted from the TS since the requirement is adequately controlled by 10 CFR 50.54(m)(2)(iii).	Yes	Yes	Yes	Yes
01-06 LG	The details regarding the minimum shift crew requirements have been removed from the CTS because they are redundant to 10 CFR 50.54(k), (l), and (m) with the exception of the requirement for non-licensed operators. The minimum shift crew requirements will be moved to a licensee controlled document.	Yes - Move to FSAR	No - CTS already contains changes	Yes, move to USAR	Yes, move to FSAR

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-06 A	CTS [6.9.1.7], "Annual Radioactive Effluent Release Report" and CTS [6.14.c] is revised consistent with NUREG-1431, Rev. 1, to delete the term "Annual" and modify the submittal date.	Yes	Yes	Yes	Yes
03-07 A	CTS [6.9.1.6], "Annual Radiological Environmental Operating Report" is revised to include specific details concerning the contents of the report.	Yes	Yes	Yes	Yes
03-08 A	CTS Specification [6.9.1.8, 6.9.1.9 and 6.9.2] are revised to delete the reference to submittal location for the monthly report, core operating limits report and special reports.	Yes	Yes	Yes	Yes
03-09 LG	The record retention requirements are moved to ~~the FSAR and implementing procedures~~ a licensee controlled document. The requirement for retention of records related to activities affecting quality is contained in 10 CFR 50, Appendix B, Criteria XVII and other sections of 10 CFR 50 that are applicable to the plant (i.e., 50.71, etc.).	Yes - QA Plan in Chapter 17 of the FSAR	Yes - QA Plan in Chapter 17 of the FSAR	Yes - QA Plan in Chapter 17 of the USAR	Yes - QA Plan in Chapter 17 of the FSAR
03-10 LG	The Radiation Protection Program is moved to the FSAR. This program requires procedures to be prepared for personnel radiation protection consistent with 10 CFR Part 20. Periodic review of these procedures is required by 10 CFR 20.1101(c).	Yes	No - Deleted from CTS per Amendment 50/36	Yes	Yes NR 5.0-001
03-11 A	The High Radiation Area is revised to be consistent with the new Part 20 requirements. Changes are non-technical to add clarification.	Yes	Yes	Yes	Yes
03-12 LG	The Process Control Program (PCP) section is proposed to be moved outside the CTS. The PCP implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71.	Yes - move to FSAR	No - Deleted from CTS per Amendment 50/36	Yes - moved to USAR	Yes - moved to the FSAR
03-13 M	The following report[s] will be added to to the ITS Admin Controls Section: "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" [and "Post Accident Monitoring (PAM) Report."]	Yes	Yes	Yes	Yes