BACKGROUND (continued)

The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of three safety valves. Accidents that could result in overpressurization if not properly terminated include: Q-3.4.Gen-1

a. Uncontrolled rod withdrawal from full power;

b. ~~Loss of reactor coolant flow~~ Feedline break;

c. Loss of external electrical load;

d. Loss of normal feedwater;

e. Loss of all AC power to station auxiliaries; and

f. Locked rotor.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required ~~in events b, c, d, e and f (above)~~ to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10CFR50.36(c)(2)(ii).

LCO

The three pressurizer safety valves are set to open at the RCS design pressure (~~2500 psia~~ 2485 psig), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig.

BASES (continued)

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice. The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the surveillance to be performed in ~~any~~ MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. ~~below MODE 2.~~ In accordance with References 4, 5 and 6, administrative controls require this test be performed in MODES 3, 4 or 5 to adequately simulate operating temperature and pressure effects on PORV operation.

Q-3.4.Gen-1

Q-3.4.11-2

~~SR 3.4.~~

~~Operatir the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. The Frequency of [18] months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.~~

~~SR 3.4.11.4~~

~~This Surveillance is not required for plants with permanent 1E power supplies to the valves.~~
~~The surveillance demonstrates that emergency power can be provided and is performed by transferring power from normal to emergency supply and cycling the valves. The frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

REFERENCES

1. Regulatory Guide 1.32, February 1977.

2. FSAR, Chapter ~~Section~~ 15.

3. ASME, Boiler and Pressure Vessel Code, Section XI.

4. Generic Letter 90-06, "resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and generic issue 94, 'Additional Low-Temperature Overpressure for Light-Water Reactors,' Pursuant to 10CFR50.54(f)," June 25, 1990.

5. CPSES License Amendment 11, July 15, 1992

6. NUREG-0797, Supplement 25, September 1992

Q-3.4.11-2

BACKGROUND (continued)

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than ~~one High Pressure Injection (HPI) pump and one~~ two charging pumps for makeup in the event of loss of inventory, then pumps can be made available through manual actions. Q-3.4.Gen-1

The LTOP System for pressure relief consists of two PORVs with reduced lift settings, or two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve, or a depressurized RCS and an RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to keep from overpressurizi~~ngation~~ the RCS for the required coolant input capability.

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The LTOP System actuation logic monitors both RCS temperature and RCS pressure and determines when ~~a condition not acceptable in~~ the PTLR limits are ~~is~~ approached. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The PTLR presents the ~~PORV~~ setpoints for the LTOP System. The setpoints are normally staggered so only one valve typically opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Rendering all ~~but one HPI~~ safety injection pumps and one charging pump incapable of injection;

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. ~~Disallowing~~ Precluding start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," provide this protection.

The ~~Reference 4~~ analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only ~~one [HPI] pump [and one~~ two charging pump~~s~~ ~~are] is [~~are actuated. Thus, the LCO allows only ~~[one] [HPI] pump [and one~~ two charging pumps OPERABLE during the LTOP MODES. Since neither one RCS relief valve nor the RCS vent can handle the pressure transient ~~need~~ from accumulator injection, when RCS temperature is low, the LCO also requires ~~the~~ accumulator~~s~~ isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

Q-3.4.Gen-1

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions. ~~The analyses show the effect of accumulator discharge~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES

RCS Vent Performance (continued)

The LTOP System satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10CFR50.36(c)(2)(ii).

LCO

This LCO requires that the LTOP System is OPERABLE. ~~The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE.~~ Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires ~~one~~ ~~HPI~~ zero safety injection pumps and ~~one~~ a maximum of two charging pumps be capable of injecting into the RCS, and all accumulator discharge isolation valves be closed and immobilized~~. When~~, when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. Q-3.4.Gen-1 Q-3.4.12-2

The LCO is modified by a Note stating that the accumulator ~~isolation is only required~~ may be unisolated when the accumulator pressure is less than ~~more than or at~~ the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

~~a. Two RCS relief valves, as follows:~~ Q-3.4.12-1

a.~~1~~. Two OPERABLE PORVs; or

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

b~~2~~. Two OPERABLE RHR suction relief valves; or

An RHR suction relief valve is OPERABLE for LTOP when its RHR suction isolation valves ~~and its RHR suction valve~~ are open, its setpoint is at or between 436.5 psig and 463.5 psig, and testing has proven its ability to open at this setpoint.

(continued)

BASES (continued)

ACTIONS

G.1 (continued)

mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 Q-3.4.Gen-1

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of ~~one~~ zero ~~HPI~~ safety injection pumps and a maximum of ~~one~~ two charging pumps are verified ~~in~~capable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and locked out. Verification that each accumulator is isolated is only required when accumulator isolation is required as stated in Note 1 to the Applicability.

The ~~HPI~~ safety injection pumps and charging pump are rendered incapable of injecting into the RCS, for example, through removing the power from the pumps by racking the breakers out under administrative control or by isolating the discharge of the pump by closed isolation valves with power removed from the operators or by a manual isolation valve secured in the closed position. ~~An a~~Alternate methods of LTOP ~~control~~ prevention may be employed ~~ing at least two independent means~~ to prevent a pump start such that a single failure ~~or single action~~ will not result in an injection into the RCS. ~~This may be accomplished through the pump control switch being placed in pull to lock and at least one valve in the discharge flow path being closed.~~ Providing pumps are rendered incapable of injecting into the RCS, they may be energized for purposes such as testing or for filling accumulators.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

(continued)

BASES (continued)

e. Primary to Secondary LEAKAGE through Any One SG

Q-3.4.Gen-1

The 500 gallons per day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or if unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor

(continued)

BASES

ACTIONS (continued)

degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB or the high pressure portion of the system.

Q-3.4.Gen-1

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring ~~one leaking~~ the RCS PIV to within limits. The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This time frame considers the time required to complete the Action and the low probability of a second valve failing during this time period.

~~or~~

~~The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period. (Reviewer Note: Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition.)~~

B.1 and B.2

Q-3.4.Gen-1

If leakage cannot be reduced, the system isolated, or the other Required Actions accomplished, ~~the plant must~~ be brought to a MODE in which the requirement does not apply. To achieve this

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.2 ~~and SR 3.4.14.3~~ (continued)

beyond 125% of its design pressure of 600 psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < ~~425~~ 442 psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The 18 month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

Q-3.4.Gen-1

~~These SRs are modified by Notes allowing the RHR autoclosure function to be disabled~~ This SR is not applicable when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. Technical Requirements Manual
7. ASME, Boiler and Pressure Vessel Code, Section XI.
8. 10 CFR 50.55a(g).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE ~~is or~~ and air cooler condensate flow rate monitor are instrumented to alarm for increases of 0.5 to 1.0 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE. Q-3.4.Gen-1

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} μCi/cc radioactivity for particulate monitoring and of 10^{-6} μCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an

ACTIONS

C.1.1, C.1.2, C.2.1 and C.2.2 (continued)

are the ~~containment sump monitor~~ Containment Sump Level and Flow Monitoring System and the containment atmosphere particulate radioactive monitor. This Condition does not provide all the required diverse means of leakage detection. With both gaseous containment atmosphere radioactivity monitoring and containment air cooler condensate flow rate monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Q-3.4.15-4

The followup Required Action is to restore either of the inoperable required monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.

A note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process necessary data after stable plant conditions are established.

~~E~~D.1 and ~~E~~D.2

If a Required Action of Condition A, B or C ~~or D~~ cannot be met, 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 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. Q-3.4.Gen-1

~~F~~E.1

With all required monitors/systems inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required. Q-3.4.15-4

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.1-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: Difference 3.4-38

Comment: TSTF-105 has been rejected by the NRC.

FLOG Response: The July 27, 1998 industry traveler status reports indicate the status of TSTF-105 as rejected by the NRC with the TSTF considering. The FLOG has reviewed the traveler and is withdrawing the traveler from the conversion application.

For Diablo Canyon, the CTS will be used which does not require a specific method for measuring RCS flow. This difference from the STS is justified by revised JFD 3.4-38.

ATTACHED PAGES:

Attachment 8 CTS 3/4.2 ITS 3.2

Encl. 2	3/4.4-12
Encl. 3A	8 and 8a
Encl. 3B	7

Attachment 10 CTS 3/4.4 ITS 3.4

Encl. 5A	Traveler Status page , 3.4-3
Encl. 6A	7
Encl. 6B	5

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the stated limits:

a. ~~Indicated~~ Reactor Coolant System $T_{avg} \leq 592°F$ 05-01-LG

b. ~~Indicated~~ Pressurizer Pressure
$\geq$ 2219 psig* 05-01-LG

c. ~~Indicated~~ Reactor Coolant System (RCS) Flow
$\geq$ 403,400 gpm** for Unit 1 05-01-LG
$\geq$ 408,000 gpm** for Unit 2

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next ~~4~~ 6 hours. 05-06-LS

With RCS flow measurement per Specification 4.2.5.4 not meeting limit, do not exceed 85% RTP. 05-11-A

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the above parameters (RCS T_{avg}, Pressurizer Pressure, indicated RCS flow) shall be verified to be within its limits at least once per 12 hours. 05-11-A

~~4.2.5.2 The RCS total flow rate shall be verified to be within its limits at least once per 31 days by plant computer indication or measurement of the RCS elbow tap differential pressure transmitters' output voltage.~~ 05-02-LS

4.2.5.3 The RCS loop flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. ~~4.2.5.3 The RCS loop flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The channels shall be normalized based on the RCS flow rate determination of Surveillance Requirement 4.2.5.4.~~ 05-12-A CP-3.2-001 05-03-LG CP-3.2-001

4.2.5.4 The RCS total flow rate shall be ~~determined by precision heat balance measurement~~ ~~measured~~ determined by precision heat balance measurement after each fuel loading and prior to operation above 85% of RATED THERMAL POWER. ~~The feedwater pressure and temperature, the main steam pressure, and feedwater flow differential pressure instruments shall be calibrated within 90 days of performing the calorimetric flow measurement.~~ 05-04-LG Q-3.4.1-1 05-04-LG

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

~~** Includes a 1.8% flow measurement uncertainty.~~ 05-01-LG

CHANGE NUMBER	NSHC	DESCRIPTION
04-07	A	Not applicable. See conversion comparison table (enclosure 3B).
04-08		Not used.
04-09	A	Not applicable. See conversion comparison table (enclosure 3B).
04-10	LS-14	Not used. ~~applicable. See conversion comparison table (enclosure 3B).~~ Q-3.2-6
04-11	A	Not applicable. See conversion comparison table (enclosure 3B).
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	LS-7	The 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. This action was included in the TS based on the potential drift of the RCS flow indication. The essence of this activity is performed on a quarterly basis through the Channel Operability Test performed in accordance with the Reactor Trip System surveillances.
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 RCS loop flow rate indicators ~~channels are~~ is moved to the Bases for ~~the RCS flow - low reactor trip function in~~ ITS Section 3.34.1. 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 a licensee controlled document. The requirement to verify that the RCS flow is within limits remains within the Technical Specification. This is an~~ Q-3.4.1-1

CHANGE NUMBER	NSHC	DESCRIPTION
		~~example of removing unnecessary details from the TS and is acceptable based on the guidance provided in 10 CFR 50.36.~~ TS SR 4.2.5.4 provides descriptive detail of the method for the determination of RCS total flow rate during a Surveillance. This detail is moved to the ITS SR 3.4.1.4 Bases. These details are not necessary to ensure the RCS total flow rate is within required limits. The requirements of ITS SR 3.4.1.4 are adequate for ensuring the RCS total flow rate is within required limits. These details are not necessary to be in the TS to ensure the RCS total flow rate is within required limits. Moving this information maintains consistency with NUREG-1431. Any change to this descriptive information will be made in accordance with the Bases Control Program described in ITS Section 5.5.14. Q-3.4.1-1
05-05	LG	Not applicable. See conversion comparison table (enclosure 3B).

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-03 LG	The requirement ~~to perform a CHANNEL CALIBRATION at least once per 18 months and the requirement~~ to normalize the RCS loop flow rate indicators ~~channels are~~ is moved to the Bases for ~~the surveillance requirements for the RCS flow - low reactor trip function in~~ ITS 3.34.1.	~~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 details regarding the timing of calibration of the instrumentation used in the performance of the calorimetric flow measurement ~~is~~ are moved to the Bases.	No - Requirement not in CTS.	Yes	Yes	Yes Q-3.4.1-1
05-05 LG	The Wolf Creek required actions would be modified to move details regarding identification of the cause for low flowrate to the Bases.	No	No	Yes	No
05-06 LS-8	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 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	NA	NA	NA	NA
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.
05-10 A	The requirement to verify RCS flow rate within limits prior to operation above 75% RTP after each fuel loading and at least every 31 EPFDs would be eliminated from the SRs for DNB parameters.	Yes	No - Requirement not in CTS.	No - Requirement not in CTS.	Yes

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev.3 ~~2~~	Incorporated	3.4-33	Approved by NRC Q-3.4.2-1
TSTF-28	Incorporated	3.4-22	Approved by NRC.
TSTF-54, Rev. 1	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-60	Incorporated	3.4-15	Approved by NRC.
TSTF-61	Not incorporated	NA	Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. 2~~1~~	Incorporated	3.4-31	Approved by NRC. TR-3.4-004
TSTF-93, Rev. 3~~1~~	Incorporated	3.4-17	Approved by NRC. Q-3.4.9-3
TSTF-94, Rev.1	Not incorporated	NA	Retained current TS. TR-3.4-005
~~TSTF-105~~	~~Incorporated~~	~~3.4-38~~	Q-3.4.1-1
TSTF-108, Rev. 1	Not incorporated	NA	LCO 3.4.19 does not apply.
TSTF-113, Rev. 4 ~~3~~	Incorporated	3.4-39	Q-3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev.2 ~~1~~	Incorporated	3.4-36	Q-3.4.13-2
TSTF-136	Incorporated	NA	Approved by NRC TR-3.4-007
TSTF-137	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-138	Not incorporated	NA	Inconsistent with RCS loops requirements of ITS 3.4.5 and 3.4.6.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.1.1	Verify pressurizer pressure is ≥ ~~[2200]~~2219 psig.	12 hours	B-PS
SR 3.4.1.2	Verify RCS average temperature is ≤ ~~[581]~~592°F.	12 hours	B-PS
SR 3.4.1.3	Verify RCS total flow rate is ≥ ~~[284,000] gpm~~ 403,400 gpm for Unit 1 408,000 gpm for Unit 2.	12 hours	B-PS
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until ~~24 hours after ≥ [90]% RTP.~~ after exceeding 85% RTP after each refueling outage. --- Verify ~~by precision heat balance that~~ by precision heat balance that ~~measured~~ RCS total flow rate is ≥ ~~[284,000] gpm~~ 403,400 gpm for Unit 1 408,000 gpm for Unit 2.	18 months	3.4-34 ~~3.4-38~~ Q-3.4.1-1 B B-PS

CHANGE NUMBER	JUSTIFICATION
3.4-36	SR 3.4.13.1 and ACTIONS for LCO 3.4.15 are revised with the addition of a note per TSTF-116. The note addresses the concern that an RCS water inventory balance cannot be meaningfully performed unless the unit is operating at or near steady state conditions. The note added to the surveillance provides an exception for operation at less than steady state conditions. The RCS water inventory balance will only be allowed to be deferred for 12 hours after re-establishing steady state conditions.
3.4-37	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
3.4-38	~~Consistent with TSTF-105, the details on the method by which the RCS flow rate are verified is moved from the SR 3.4.1.4 to the Bases. Moving this information to the Bases, allows the use of precision heat balances, elbow taps, and other acceptable methods in order to perform this verification and is consistent with the NUREG-1431 philosophy of moving clarifying information and descriptive details out of the TS to the Bases.~~ Not applicable to CPSES. See conversion comparison table (enclosure 6B). Q-3.4.1-1
3.4-39	The shutdown requirements of ITS 3.4.11 would require the plant to reduce T_{avg} to <500°F within 12 hours, rather than MODE 4, to address the concern of entering [LTOP] LCO 3.4.12 Applicability with inoperable PORVs. New initial Required Actions are added to Conditions D, E and G to immediately initiate actions for restoration of the inoperable PORV(s) (and or PORV block valves) to OPERABLE status. These immediate actions will ensure expedient measures are taken to re-establish the operability of PORV(s) (and PORV block valves) while maintaining plant conditions above MODE 4 but less than 500°F. For consistency, the shutdown requirements of ITS 3.4.16 are also revised to allow 12 hours to reduce T_{avg} to <500°F. This change is consistent with TSTF-113. Q-3.4.11-6
3.4-40	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
3.4-41	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
3.4-42	Not applicable to CPSES. See Conversion Comparison Table (enclosure 6B).
3.4-43	Not applicable to CPSES. See Conversion Comparison Table

DIFFERENCE FROM NUREG-1431		APPLICABILITY				
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY	
3.4-38	~~Consistent with TSTF-105, the details on the method by which the RCS flow rate is verified are moved from the SR 3.4.1.4 to the Bases.~~ Consistent with DCPP CTS SR 4.2.3.3, the details on the method by which the RCS flow rate is verified is removed from STS SR 3.4.1.4 text.	Yes	~~Yes~~ No	~~Yes~~ No	~~Yes~~ No	Q-3.4.1-1
3.4-39	The shutdown requirements of ITS 3.4.11 would require the plant to reduce T_{avg} to <500°F within 12 hours, rather than MODE 4, to address the concern of entering [LTOP] LCO 3.4.12 Applicability with inoperable PORVs. For consistency, the shutdown requirements of ITS 3.4.16 are also revised to allow 12 hours to reduce T_{avg} to <500°F. This change is consistent with TSTF-113.	Yes	Yes	Yes	Yes	
3.4-40	The Note to SR 3.4.1.4 would be modified to specify a plant specific power and to provide additional time to perform an RCS precision flow rate measurement.	No - See CN 3.4-51	No - See CN 3.4-34	Yes	Yes	
3.4-41	LCO 3.4.1 is revised to reference Tables 3.4.1-1 and 3.4.1-2 for RCS total flow rate limits for DCPP Units 1 and 2 respectively.	Yes Allowance added per Amendment 60/59.	No	No	No	
3.4-42	An exception to SR 3.4.14.1 frequency to leak test PIVs 8802A, 8802B and 8703 has been added. This change is consistent with the DCPP current TS.	Yes	No	No	No	
3.4-43	A new Condition C is added to LCO 3.4.1 to reflect the current TS of Wolf Creek for RCS flow rate.	No .	No	Yes	No	
3.4-44	Steam generator levels for Modes 3, 4 and 5 are specified to ensure SG tubes are covered. The ~~Callaway~~ current TS did not ensure tube coverage.	No	No	~~No~~Yes	Yes	Q-3.4.5-2 Q-3.4.5-3

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.1-2 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: Difference 3.4-40

Comment: WOG-99 has not yet become a TSTF.

FLOG Response: WOG-99 has been designated TSTF-282 which is currently under NRC review. No changes to the ITS mark-ups were made in the process of assigning this traveler a TSTF number. As explained in Enclosure 6B to Attachment 10, JFD 3.4-40 does not apply to CPSES or DCPP. Those plants are retaining their CTS, as explained under JFDs 3.4-34 and 3.4-51, respectively. Callaway and Wolf Creek continue to pursue the changes proposed by this traveler.

ATTACHED PAGES:

Encl 5A Traveler Status Sheet

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-151 Rev. 1	Incorporated	NA	TR-3.4-009
TSTF-153	Incorporated	3.4-01	Approved by NRC TR-3.4-009
TSTF-162	Incorporated	NA	Approved by NRC. TR-3.4-006
~~WOG-51, Rev.1~~ TSTF-285	Incorporated	3.4-45 & 3.4-~~23~~52	See also CNs 3.4-18 and 3.4-20. Q-3.4-12-2
~~WOG-60~~ TSTF-288	Incorporated	3.4-35	Q-3.4.11-2
~~WOG-67, Rev. 1~~ TSTF-233	Incorporated	3.4-10	Approved by NRC. DCPP only. TR-3.4-009
WOG-87 Rev. 2	Incorporated	3.4-47	Q-3.4.11-4
~~WOG-99~~ TSTF-282	Incorporated	3.4-40	Applicable to Callaway and Wolf Creek only. Q-3.4.1-2
~~WOG-100~~TSTF-280	Incorporated	3.4-49	Q-3.4.12-1

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.2-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST Difference 3.4-33

Comment: TSTF-27 Rev. 3 is still pending NRC approval.

FLOG Response: The July 27, 1998 industry traveler status reports indicate the status of TSTF-27, Rev. 3 as approved by the NRC. The proposed wording in TSTF-27, Rev. 3 was modified from TSTF-27, Rev. 2, and these modifications have been incorporated into the ITS. The FLOG continues to pursue the changes approved in TSTF-27, Rev. 3.

ATTACHED PAGES:

Encl. 5A	Traveler Status page
Encl. 5B	B 3.4-9
Encl 6A	6

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev.3 2	Incorporated	3.4-33	Approved by NRC Q-3.4.2-1
TSTF-28	Incorporated	3.4-22	Approved by NRC.
TSTF-54, Rev. 1	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-60	Incorporated	3.4-15	Approved by NRC.
TSTF-61	Not incorporated	NA	Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. 2~~1~~	Incorporated	3.4-31	Approved by NRC. TR-3.4-004
TSTF-93, Rev. 3~~1~~	Incorporated	3.4-17	Approved by NRC. Q-3.4.9-3
TSTF-94, Rev.1	Not incorporated	NA	Retained current TS. TR-3.4-005
~~TSTF-105~~	~~Incorporated~~	~~3.4-38~~	Q-3.4.1-1
TSTF-108, Rev. 1	Not incorporated	NA	LCO 3.4.19 does not apply.
TSTF-113, Rev. 4 ~~3~~	Incorporated	3.4-39	Q-3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev.2 ~~1~~	Incorporated	3.4-36	Q-3.4.13-2
TSTF-136	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-137	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-138	Not incorporated	NA	Inconsistent with RCS loops requirements of ITS 3.4.5 and 3.4.6.

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, 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 2 with K_{eff} < 1.0 ~~3~~ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with K_{eff} < 1.0 ~~3~~ in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above ~~541~~551°F every 12 hours ~~30 minutes when T_{avg}-T_{ref} deviation, low low T_{avg} alarm not reset and any RCS loop T_{avg} < 547561°F.~~

~~The Note modifies the SR. When any RCS loop average temperature is < 547561°F and the T_{avg} - T_{ref} deviation, low low T_{avg} alarm is alarming, RCS loop average temperatures could fall below the LCO requirement without additional warning.~~ The SR to verify operating RCS loop average temperatures every ~~30 minutes~~ 12 hours ~~is frequent enough to prevent the inadvertent violation of the LCO.~~ takes into account indications and alarms that are continously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

Q-3.4.2-1

REFERENCES

1. FSAR, ~~Section [15.0.3]~~ Chapter 15.

CHANGE NUMBER	JUSTIFICATION
3.4-33	The Frequency of SR 3.4.2.1 to verify operating RCS loop average temperature at or above [551]°F is changed to "12 hours" from the current surveillance frequency of 30 minutes. The SR to verify operating loop average temperatures every 12 hours ~~is sufficiently frequent to prevent inadvertent violation of the LCO and~~ considers indications and alarms that are continuously available to the operator in the control room. This change is based on industry traveler TSTF-27. Q-3.4.2-1
3.4-34	The current CPSES licensing basis, which does not require that the precision RCS flow measurement be current until after exceeding 85% RTP following each refueling outage, is retained. As discussed in the Bases, the analyses supporting this requirement are predicated on the performance of a gross flow measurement prior to entry into Mode 1 and the maintenance of a reduced power range neutron flux - high reactor trip setpoint until the RCS flow has been verified. Through the use of the Transit Time Flow Meter and other precision instrumentation which has been installed by TU Electric, a sufficiently accurate RCS flow measurement may be made prior to 85% RTP. This capability allows for the use of a single power plateau below 85% RTP for performing required surveillances during the post-refueling power ascension. It also allows the plant to remain below 85% RTP following a refueling outage until the RCS flow is verified to meet the required flow. This change affects the Conditions (a separate condition is required for the precision flow measurement prior to exceeding 85% RTP), and the note and frequency associated with SR 3.4.1.4.
3.4-35	This change adds a note to SR 3.4.11.1 and SR 3.4.11.2 stating that the SRs are only required to be performed in Modes 1 and 2. The Actions Note "LCO 3.0.4 is not applicable" is intended to allow Mode changes for testing purposes (per Bases). This allowance is properly presented as an SR Note. A properly placed exception (i.e., an SR Note exception) would not allow the SR to be considered to be met until the appropriate conditions were available for it to be performed without entering the actions. The Note to these SRs would allow startup in Mode 3 if the SR had not been performed during the required frequency, but would limit the exception to prior to entering Mode 2. The change is consistent with traveler TSTF-288 ~~WOG-60~~. Q-3.4.11-2

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.3-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: ITS 3.4.3 Bases References

Comment: WCAP-14040-NP-A, Rev. 2 January 1996, has replaced WCAP-7924-A, April 1975. Please summarize the differences/applicability to the FLOG.

FLOG Response: WCAP-14040-NP-A, Rev. 1 was NRC approved as an acceptable reference for "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Curves" by SER dated 10/16/95 with minor comments which did not affect the SER. These comments were incorporated and the WCAP-14040-NP-A was issued as Revision 2 in January 1996. NRC acceptance of this WCAP as a reference was based upon the following key elements:

1) The WCAP incorporates state of the art fast neutron radiation transport.
2) The WCAP cold overpressure mitigating system satisfies SRP Section 5.2.2 and BTP RSB 5-2.
3) The WCAP fracture mechanics calculation conforms to 10CFR50, Appendix G and SRP Section 5.3.2.
4) The WCAP conforms to Reg. Guide 1.99, Rev. 2 in calculation of the adjusted reference temperature.
5) The WCAP conforms to 10CFR50, Appendix G for methodology for calculating minimum temperature in the P-T limit curves.
6) The WCAP satisfies the provisions of the draft generic letter published in the Federal Register for comment of June 2,1995.

These items are consistent with the STS reviewer's Note on STS 5.6.6.

<u>Plant Specific Discussion:</u>

The CPSES-specific RCS pressure and temperature analyses were last revised to support CTS Amendment 14 for Unit 1 and the original issue for Unit 2. As such, these analyses predate WCAP-14040; however, they were performed in a manner generally consistent with the bases of WCAP-14040. As described in ITS Section 5.6.6, the CPSES RCS Pressure and Temperature Limits (ITS 3.4.3) were developed in accordance with 10CFR50 Appendix G and Appendix H, Regulatory Guide 1.99, Revision 2, NUREG-0800 Section 5.3.2, and ASME Code Section III, Appendix G. Because the CPSES licensing basis does not currently include WCAP-14040, the reference in ITS Section 3.4.3 will be revised accordingly.

ATTACHED PAGES:

Encl 5B B 3.4-10 and B 3.4-16.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature ~~(Ref. 1)~~. Q-3.4.3-1

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based

BASES

reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. ~~WCAP-14040-NP-A, Rev. 2, January 1996.~~ Not used Q-3.4.3-1

2. 10 CFR 50, Appendix G.

3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

4. ASTM E 185-82, July 1982.

5. 10 CFR 50, Appendix H.

6. Regulatory Guide 1.99, Revision 2, May 1988.

7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.4-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST ITS 3.4.4 Bases

Comment: The Bases refer to the DNBR limit in the safety limits. Where is it? (this appears to be a problem with the STS, as well as these conversions).

FLOG Response: As described in the Applicable Safety Analyses Bases for ITS Section 2.1.1, the DNBR limit is: "There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB." The actual numerical value is specific to a given DNBR correlation and analytical methodology. The correlations and methodologies are NRC-approved. More than one correlation or methodology, as generally documented in the FSAR, may be used depending on core design and the particular transient being analyzed. For this reason, a more general term such as "DNBR limit" is used. This convention has been used throughout the Bases for ITS Sections 2.0, 3.1, 3.2, 3.3, and elsewhere in 3.4.

In the process of responding to this RAI, it was noted that all FLOG plants except DCPP and CPSES have a markup methodology error in the second to last paragraph of the Applicable Safety Analyses Bases for ITS Section 3.4.4. The acronym "SL" should have been struck-through; this is addressed under Comment Number 3.4.Gen-1.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.5-4 **APPLICABILITY:** CP

REQUEST: ITS SR 3.4.5.2 (Comanche Peak)

Comment: It should read "SR" rather than "Sr".

FLOG Response: The "Sr" in the "clean" version of ITS SR 3.4.5.2 has been corrected to be "SR".

ATTACHED PAGES:

Attachment 19 - ITS Specifications (clean) Page 3.4-9

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	12 hours
SR 3.4.5.2	Verify steam generator secondary side water levels are ≥ 10% for required RCS loops.	12 hours
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

Q-3.4.5-4

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.6-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: Difference 3.4-02

Comment: The difference states that the STS doesn't cover all possible configurations and the language of the STS is potentially confusing. Please explain the basis for these comments.

FLOG Response: The STS wording for Condition A, "One required RCS loop inoperable AND Two RHR loops inoperable", and for Condition B, "One required RHR loop inoperable AND Two required RCS loops inoperable", is confusing. This confusion arises from the fact the LCO allows any combination of two RCS or RHR loops, including one RCS loop and one RHR loop, to satisfy the OPERABILITY requirement yet Conditions A and B are worded as if either two RCS loops or two RHR loops, exclusively, were the required loops.

By way of illustration, the following scenarios are presented. Assume the LCO's OPERABILITY requirements are satisfied by one RCS loop and one RHR loop. These loops are serving as the "required" loops. If the RCS loop becomes inoperable, Condition A does not apply because it is "ANDED" with "Two RHR loops inoperable" yet one RHR loop remains OPERABLE in this scenario. Conversely, if the RHR loop becomes inoperable, Condition B does not apply because it is "ANDED" with "Two required RCS loops inoperable" yet one RCS loop remains OPERABLE. In fact, the wording of STS Condition B is at odds with the LCO since Condition B requires three loops to be OPERABLE (one RHR and two RCS loops).

The FLOG considered this wording to be a potential source of error for plant operators. Since the corresponding CTS specification is not confusing it was adopted in lieu of the STS wording. This confusion also led to the WOG creating a traveler, WOG-109, which was subsequently withdrawn and superseded by TSTF-263 which is currently under NRC review. TSTF-263 presents a very similar approach to that used by the FLOG to correct STS 3.4.6; however, TSTF-263 has not been incorporated by the FLOG. TSTF-263 was not issued until several months after the FLOG submittals. The changes incorporated in ITS 3.4.6 are based on the CTS which has less rigid logic connectors than the STS.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.8-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: Difference 3.4-48

Comment: It is unclear why TS 3.0.4 would not apply. If this change is to be considered it should be done on a generic basis.

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 3.4-48 has been revised to incorporate additional justification from NSHC LS-1 in Enclosure 4 of the Section 3.0 package (Attachment No. 6). JFD 3.4-48 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 3.4.8 is modified by a Note stating: "While this LCO is not met, entry into MODE 5 Loops Not Filled from MODE 5 Loops Filled is not permitted." The transition from MODE 5 (loops filled) to MODE 5 (loops not filled) removes the steam generators as a decay heat removal system while the RHR System is potentially degraded. Therefore, the Note ensures that the transition is precluded if LCO 3.4.7.b (two SGs) were chosen (in lieu of the second RHR loop) to ensure decay heat removal capability prior to draining the RCS."

It should be noted that the Applicability Bases for ITS 3.4.8 already provides a similar discussion.

ATTACHED PAGES:

Encl. 6A 8a

status within time t_0 + 72 hours allows the plant to exit Condition B. If power were not restored to the block valve at this time, the new Note on Condition C would have no standing and Condition C would be entered. Similar conclusions can be drawn for the relationship between Conditions E and F. If Condition E is the original Condition entered, there is nothing to be gained by Required Action F.1 and Required Action F.2 can't be satisfied with block valve power removed. With F.2 not satisfied, Required Action G.2 would require the plant to be in MODE 3, but Required Action E.4 would have already had the plant in MODE 3 two hours earlier. This change is consistent with traveler WOG-87. Q-3.4.11-4

3.4-48 A note is added to ITS 3.4.8 ACTIONS, indicating that entry into MODE 5 Loops Not Filled from MODE 5 Loops Filled is not permitted while LCO 3.4.8 is not met. The addition of this note is based on the performance of a plant specific LCO 3.0.4 matrix (see CN 1-02-LS-1 of the CTS 3/4.0 package). 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 3.4.8 is modified by a Note stating: "While this LCO is not met, entry into MODE 5 Loops Not Filled from MODE 5 Loops Filled is not permitted." The transition from MODE 5 (loops filled) to MODE 5 (loops not filled) removes the steam generators as a decay heat removal system while the RHR System is potentially degraded. Therefore, the Note ensures that the transition is precluded if LCO 3.4.7.b (two SGs) were chosen (in lieu of the second RHR loop) to ensure decay heat removal capability prior to draining the RCS. Q-3.4.8-1

3.4-49 LCO 3.4.12 "[LTOP] System", provides four different methods for pressure relief. Any of the four methods may be used. However, Surveillance Requirement 3.4.12.5 requires testing whether or not the equipment is being credited to meet the LCO. The proposed change adds the words "required" to the Surveillance to exempt its performance if the equipment to be tested is not being used to meet the LCO. In addition, two editorial changes were made. The LCO requirement presentation was clarified. Also, the Note to SR 3.4.12.8 was revised to replace "required" to be met" with "required to be performed" since the "performed" nomenclature is appropriate here, consistent with the CTS. This change is consistent with traveler TSTF-280 ~~WOG-100~~. Q-3.4.12-1

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.9-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: ITS 3.4.9

Comment: Does 92% (90% for Diablo Canyon) in the pressurizer ensure that upon an inadvertent SI that the pressurizer will not overfill before the operator is assumed to take action? Other plants have lowered this limit (Robinson) or qualified the PORVs for water (Millstone 3).

FLOG Response: ITS Surveillance Requirement 3.4.9.1 requires the pressurizer water level to be less than 92% (90% for Diablo Canyon). This requirement is not related to the assumptions used in the inadvertent Safety Injection analysis. The basis for this requirement is given in the ITS Bases for the SR (as clarified by NRC approved TSTF-162), which states that it is to ensure provision of a minimum space for a steam bubble which is an assumption in the safety analyses (i.e., the pressurizer must not be water solid). This maximum pressurizer level is not assumed in any safety analysis.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.9-3 **APPLICABILITY:** CP, DC, WC

REQUEST: Difference 3.4.17 (Wolf Creek, Diablo Canyon and Comanche Peak)

Comment: TSTF-93 Rev. 3 was approved with a reviewer's note which says that for non-dedicated safety-related heaters which normally operate the frequency is 18 months and for dedicated safety-related heaters which normally don't operate the frequency is 92 days. Each of the plants is asking for the 18 month frequency but it is unclear from the submittals if they meet the criterion. Please provide information demonstrating consistency with the TSTF.

FLOG Response: DCPP and WCGS have two-groups of non-safety related pressurizer backup heaters. The pressurizer heaters together with the pressurizer spray valves are used to control RCS pressure. This change is consistent with TSTF-93 and the added Reviewer's note.

For Diablo Canyon, the NRC recently approved (6/5/98) changing DCPP current Technical Specification 4.4.3.2 from 92 days to "Refueling Interval" in LA 126/124.

For Comanche Peak, the pressurizer heaters used to satisfy the pressure control function are comprised of one proportional control group and three backup groups. The design and operation is consistent with the basis for an 18 month surveillance described in Section 6.6 of NUREG-1366 (which was the basis for TSTF-93). The heater groups are normally connected to the emergency power supplies (two to each Class 1E train of emergency power) and normally operate. CPSES will revise the 3.4.9 BASES to reflect the NUREG-1366 basis for the 18 month frequency.

ATTACHED PAGES:

Encl 5A	Traveler Status page
Encl 5B	B 3.4-43 and B 3.4-44

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev.3 ~~2~~	Incorporated	3.4-33	Approved by NRC Q-3.4.2-1
TSTF-28	Incorporated	3.4-22	Approved by NRC.
TSTF-54, Rev. 1	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-60	Incorporated	3.4-15	Approved by NRC.
TSTF-61	Not incorporated	NA	Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. 2~~1~~	Incorporated	3.4-31	Approved by NRC. TR-3.4-004
TSTF-93, Rev 3~~1~~	Incorporated	3.4-17	Approved by NRC. Q-3.4.9-3
TSTF-94, Rev.1	Not incorporated	NA	Retained current TS. TR-3.4-005
~~TSTF-105~~	~~Incorporated~~	~~3.4-38~~	Q-3.4.1-1
TSTF-108, Rev. 1	Not incorporated	NA	LCO 3.4.19 does not apply.
TSTF-113, Rev. 4 ~~3~~	Incorporated	3.4-39	Q-3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev.2 ~~1~~	Incorporated	3.4-36	Q-3.4.13-2
TSTF-136	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-137	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-138	Not incorporated	NA	Inconsistent with RCS loops requirements of ITS 3.4.5 and 3.4.6.

C.1 and C.2

If one required group of pressurizer heaters is ~~are~~ inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, 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 to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is ~~within~~ consistent with the safety analyses assumptions of ensuring that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The pressurizer heaters used to satisfy the pressure control function are comprised of one proportional control group and three backup groups. The heater groups are normally connected to the emergency power supplies (two to each Class 1E train of emergency power) and normally operate. The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. This may also be done by energizing the heaters and measuring circuit current. The Frequency of 18 months ~~92 days~~ is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable. The heater design and operation is consistent with the basis for an 18 month surveillance described in Section 6.6 of Ref. 3.

Q-3.4.9-3

~~SR 3.4.9.3~~

~~This SR is not applicable if the heaters are permanently powered by Class 1E power supplies.~~

~~This Surveillance demonstrates that the heaters can be manually transferred from the normal to the emergency power supply and energized. The Frequency of 18 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.~~

REFERENCES

1. FSAR, Section 15.
2. NUREG-0737, November 1980.
3. NUREG-1366, Improvements to Technical Specification Surveillance Requirements. Q-3.4.9-3

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.10-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: ITS 3.4.10 Bases Applicable Safety Analyses

Comment: What justifies the differences between the ITS Bases and the STS Bases and between the plant Bases (especially Callaway and Wolf Creek) of the lists of possible over pressurization events?

FLOG Response: These Bases changes reflect each plant's licensing basis as expressed in their respective versions of FSAR Chapter 15.

Plant Specific Discussion

In the CPSES Applicable Safety Analysis Bases for ITS 3.4.10, changes were made to item "b" to replace the loss of reactor coolant flow with the feedwater line break because the FSAR Section 15.2.8 analysis of feedwater line break shows the primary and secondary side safeties will lift. The loss of reactor coolant flow does not cause the safeties to lift.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.11-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: Change 4-04 LG

Comment : The requirement is in the CTS and the STS. The justification for not putting it in the ITS is that automatic actuation to open is not required. However, proper calibration also ensures that the PORV does not prematurely open creating as stated in the Bases "in effect a small break LOCA."

FLOG Response: There are two CTS LCOs (3.4.9.3 (3.4.8.3 for CPSES) & 3.4.4) and corresponding ITS LCOs (3.4.12 & 3.4.11) controlling pressurizer PORV operability. One of these, CTS 3.4.9.3 (3.4.8.3 for CPSES) and corresponding ITS 3.4.12, governs their operability as part of the LTOP/COMS system. Both the CTS (SR 4.4.9.3.1.b or 4.4.8.3.1.b for CPSES) and the ITS (SR 3.4.12.9) require CHANNEL CALIBRATIONs of the LTOP/COMS PORV actuation channels every 18 months to support this function. The second of these, CTS 3.4.4 and ITS 3.4.11, governs the operability of the PORVs and their block valves as isolable relief valves. While the ability to open the PORVs manually and to isolate a stuck open PORV using its block valve are considered safety-related capabilities, the ability of the PORVs to act as automatic relief valves In MODES 1, 2, and 3 is not a safety function in the current licensing basis. The pressurizer safeties fulfill both the RCS Code overpressure protection function and the automatic pressure relief function assumed in the accident analyses. For this reason, STS 3.4.11 does not have a CHANNEL CALIBRATION surveillance requirement. SR 4.4.4.1.b is therefore moved out of the technical specifications by DOC 4-04-LG. This is appropriate since automatic actuation of the PORVs is not a safety function in MODES 1, 2, or 3. Requirements that are not needed to support the safety analyses are moved out of the Technical Specifications, reflecting the philosophy and content of NUREG-1431.

The premature opening of a PORV is considered to be a small break LOCA. A LOCA is an unisolable leak or break in the RCS. A stuck open pressurizer safety valve would constitute a LOCA. One of the design functions of the PORVs is, however, to reduce the risk of a stuck open safety by having actuation set points below those of the safeties. As stated in the STS Bases for ITS 3.4.11, LCO, a stuck open PORV could be isolated by closing its safety-related block valve, thus avoiding a LOCA. The automatic actuation of a PORV at a pressure lower than its nominal design set point is not desirable, but is not outside the safety analysis.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.11-2 **APPLICABILITY:** CA, CP, DC, WC

REQUEST Change 4-08 LS 34 and Difference 3.4-35

Comment: WOG-60 has not yet become a TSTF.

FLOG Response: WOG-60 has been approved by the TSTF and is designated as TSTF-288. This traveler has been submitted to the NRC and is under review. The proposed wording in TSTF-288 was modified from WOG-60, Rev. 1 , and these modifications have been incorporated into the ITS (editorial Bases note changes). The FLOG continues to pursue the changes proposed by this traveler.

ATTACHED PAGES:

Encl. 3A	8
Encl 3B	5
Encl. 5A	Traveler Status page
Encl. 5B	B 3.4-52 & B 3.4-55, B 3.4-55a and B 3.4-56
Encl. 6A	6

CHANGE NUMBER	NSHC	DESCRIPTION
4-08	LS-34	Consistent with traveler ~~WOG-60~~ TSTF-288, the requirement to perform the 92 day surveillance of the pressurizer PORV block valves and the 18 month surveillance of the pressurizer PORVs (i.e., perform one complete cycle of each valve) is revised to indicate that the surveillance is only required to be performed in MODES 1 and 2. This is consistent with the recommendations of Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10CFR50.54(f)," June 25, 1990. Q-3.4.11-2
4- [illegible]	LS-36	The requirement to perform the 92 day surveillance of pressurizer PORV block valves (i.e., perform one complete cycle of each block valve) is revised such that it is not required if the block valve is closed to meet ACTION a of the current TS LCO 3.4.4. This change is acceptable because no credit is taken for the automatic actuation of the PORV in Modes 1, 2, or 3. Credit is taken for manual operation of the PORVs during the Steam Generator Tube Rupture (SGTR) accident. However, the capability to manually cycle the PORVs will be unaffected by this change. This change will not affect the ability of the block valve to open, if closed to meet ACTION a, in the mitigation of an SGTR. Deferral of the block valve cycling surveillance will not diminish the design capability of the block valve to open against differential pressures that would be present after an SGTR since the block valves are capable of opening against 2485 psig, the safety valve lift pressure, whereas pressurizer pressure decreases after an SGTR. Q-3.4.11-4 In addition, a Note is added to ACTION [d] stating that it does not apply when the block valve(s) are inoperable solely as a result of its power being removed per ACTIONS [b or c] as a result of an inoperable PORV(s). In this scenario ACTION [b or c] is entered as a result of an inoperable PORV(s). If the PORV were inoperable and incapable of being manually cycled (per the change discussed under DOC 4-02-LS-6), ACTION [b] would be entered at time zero, t_0. ACTION [b] would close the associated block valve and remove its power within time t_0 + 1 hour. If, as a result of block valve power removal per ACTION [b], ACTION [d] were then entered, ACTION [d] would require the associated PORV to be placed in manual control within time t_0 + 2 hours. However, the reason for originally entering ACTION [b] is that the associated PORV is inoperable and can't be manually cycled, thus there is

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
4-04 LG	This change moves the requirement to perform channel calibration of PORV actuation instrumentation to a licensee controlled document.	Yes - Moved to the FSAR.	Yes - Moved to the TRM.	Yes - Moved to the USAR.	Yes - Moved to the FSAR.
4-05 LS-31	The shutdown requirement of CTS 3.4.4 would require the plant to go to reduceT_{avg} to <500°F within 12 hours, rather than go to MODE 4, to address the concern of entering [LTOP] LCO Applicability with inoperable PORVs. For consistency the shutdown requirements of CTS LCO [3.4.7] would be similarly revised.	Yes	Yes	Yes	Yes
4-06 LS-32	This change provides a 72 hour completion time to restore an inoperable block valve, with the PORV placed in manual control mode. The current TS requires the block valve to be restored within one hour, or remove power from the solenoid.	Yes	No - Already part of CTS.	No - Already part of CTS.	No - Already part of CTS.
4-07 LS-33	This change provides a two hour completion time for restoring an inoperable block valve when more than one block valve is inoperable, and 72 hours to restore the remaining valves. The current TS requires the block valve to be restored within one hour for one or more valves inoperable.	Yes	No - Already part of CTS.	No - Already part of CTS.	No - Already part of CTS.
4-08 LS-34	Consistent with traveler TSTF-288 WOG-60, the requirement to perform the PORV and block valve cycling surveillances is revised such that the surveillances are only required to be performed in MODES 1 and 2.	Yes	Yes	Yes	Yes Q-3.4.11-2
4-09 LS-36	Consistent with traveler WOG-87, the requirement to perform the 92 day surveillance of the pressurizer block valves (i.e., perform one complete cycle of each block valve) is revised such that it is not required if the block valve is closed to meet ACTION a of the current TS LCO 3.4.4. In addition, a Note is added to ACTION [d] to prevent entry solely due to inoperable PORV(s) under Actions [b or c].	Yes	Yes	Yes	Yes Q-3.4.11-4
5-01 A	This change moves the Steam Generator Tube Surveillances to SR 3.4.13.2 and the Administrative Controls Sections 5.5.9 and 5.6.10.	Yes	No - Same as CPSES change 1-14-A for CTS Section 3/4.0.	Yes	Yes

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-151 Rev. 1	Incorporated	NA	TR-3.4-009
TSTF-153	Incorporated	3.4-01	Approved by NRC TR-3.4-009
TSTF-162	Incorporated	NA	Approved by NRC. TR-3.4-006
~~WOG-51, Rev.1~~ TSTF-285	Incorporated	3.4-45 & 3.4-~~23~~52	See also CNs 3.4-18 and 3.4-20. Q-3.4.12-2
~~WOG-60~~ TSTF-288	Incorporated	3.4-35	Q-3.4.11-2
~~WOG-67, Rev. 1~~ TSTF-233	Incorporated	3.4-10	Approved by NRC. DCPP only. TR-3.4-009
WOG-87 Rev. 2	Incorporated	3.4-47	Q-3.4.11-4
~~WOG-99~~ TSTF-282	Incorporated	3.4-40	Applicable to Callaway and Wolf Creek only. Q-3.4.1-2
~~WOG-100~~TSTF-280	Incorporated	3.4-49	Q-3.4.12-1

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). The exception for LCO 3.0.4, Note 2, permits ~~MODE changes with inoperable PORVs or block valves as one possible recourse to remaining in the Applicability of LCO 3.4.12. entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status. Testing is not performed in lower MODES.~~ entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status, in the event that testing was not satisfactorily performed in lower MODES.

Q-3.4.11-2

A.1

~~With the~~ PORVs may be inoperable and capable of being manually cycled (e.g., excessive seat leakage). In this condition, either the PORVs must be restored or the flow path isolated within 1 hour. The associated block valve~~s should~~ is required to be closed, but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. ~~Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV inoperability may be due to seat leakage, instrumentationproblems, automatic control problems, or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve.~~ This ~~Condition is only intended to~~ permits operation of the plant ~~for a limited period of time not to exceed~~ until the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. ~~Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2).~~

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2, and B.3

If one ~~or two~~ PORV~~s~~ is inoperable and not capable of being manually cycled, it must be either restored, or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are

(continued)

BASES (continued)

system during this time and provide the operator time to correct the situation. The Required Actions are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. Q-3.4.11-4

G.1 and G.2

If the Required Actions of Condition F are not met, then 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 with T_{avg} <500°F within 6 hours ~~and to MODE 4 within 12 hours~~. Additional action is required to be immediately initiated to restore the inoperable valve(s) to OPERABLE status. This will ensure expedient measures are taken to re-establish OPERABLE PORVs and block valves while maintaining plant conditions above MODE 4 but less than 500°F. 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. In MODES 4 ~~and~~ 5 and 6 (with the reactor vessel head on), automatic ~~maintaining~~ PORV OPERABILITY may be required. See LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed~~, except when the block valve(s) is closed in accordance with the Required Actions of Condition B or E, if needed.~~ The basis for the Frequency of 92 days is the ASME Code, Section XI (Ref. 3). ~~If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an otherwise inoperable PORV that is incapable of being manually cycled, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status. (i.e., completion of the Required Actions fulfills the SR).~~ Q-3.4.11-4

This SR is modified by two Notes. ~~The~~ Note 1 modifies this SR by stating that it is not required to be ~~met~~performed with the block valve closed, in accordance with the Required Q-3.4.11-2 Q-3.4.11-4

BASES (continued)

Actions of this LCO. ~~Condition A.this LCO~~ Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable. Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the surveillance to be performed in ~~any MODE 3 under~~ operating temperature and pressure conditions, prior to entering MODE 1 or 2. ~~below MODE 2~~. In accordance with References 4, 5 and 6, administrative controls require this test be performed in MODES 3, 4 or 5 to adequately simulate operating temperature and pressure effects on PORV operation.

Q-3.4.11-4

Q-3.4.11-2

BASES (continued)

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequen‹y of 18 months is based on a typical refueling cycle and industry accepted practice. The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the surveillance to be performed ~~in any~~ MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. ~~below MODE 2.~~ In accordance with References 4, 5 and 6, administrative controls require this test be performed in MODES 3, 4 or 5 to adequately simulate operating temperature and pressure effects on PORV operation.

Q-3.4.Gen-1

Q-3.4.11-2

~~SR 3.4.11.3~~

~~Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. The Frequency of [18] months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.~~

~~SR 3.4.11.4~~

~~This Surveillance is not required for plants with permanent 1E power supplies to the valves.~~

~~The surveillance demonstrates that emergency power can be provided and is performed by transferring power from normal to emergency supply and cycling the valves. The frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

REFERENCES

1. Regulatory Guide 1.32, February 1977.

2. FSAR, Chapter ~~Section~~ 15.

3. ASME, Boiler and Pressure Vessel Code, Section XI.

4. Generic Letter 90-06, "resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and generic issue 94, 'Additional Low-Temperature Overpressure for Light-Water Reactors,' Pursuant to 10CFR50.54(f)," June 25, 1990.

5. CPSES License Amendment 11, July 15, 1992

6. NUREG-0797, Supplement 25, September 1992

Q-3.4.11-2

CHANGE NUMBER	JUSTIFICATION
3.4-33	The Frequency of SR 3.4.2.1 to verify operating RCS loop average temperature at or above [551]°F is changed to "12 hours" from the current surveillance frequency of 30 minutes. The SR to verify operating loop average temperatures every 12 hours ~~is sufficiently frequent to prevent inadvertent violation of the LCO and~~ considers indications and alarms that are continuously available to the operator in the control room. This change is based on industry traveler TSTF-27. Q-3.4.2-1
3.4-34	The current CPSES licensing basis, which does not require that the precision RCS flow measurement be current until after exceeding 85% RTP following each refueling outage, is retained. As discussed in the Bases, the analyses supporting this requirement are predicated on the performance of a gross flow measurement prior to entry into Mode 1 and the maintenance of a reduced power range neutron flux - high reactor trip setpoint until the RCS flow has been verified. Through the use of the Transit Time Flow Meter and other precision instrumentation which has been installed by TU Electric, a sufficiently accurate RCS flow measurement may be made prior to 85% RTP. This capability allows for the use of a single power plateau below 85% RTP for performing required surveillances during the post-refueling power ascension. It also allows the plant to remain below 85% RTP following a refueling outage until the RCS flow is verified to meet the required flow. This change affects the Conditions (a separate condition is required for the precision flow measurement prior to exceeding 85% RTP), and the note and frequency associated with SR 3.4.1.4.
3.4-35	This change adds a note to SR 3.4.11.1 and SR 3.4.11.2 stating that the SRs are only required to be performed in Modes 1 and 2. The Actions Note "LCO 3.0.4 is not applicable" is intended to allow Mode changes for testing purposes (per Bases). This allowance is properly presented as an SR Note. A properly placed exception (i.e., an SR Note exception) would not allow the SR to be considered to be met until the appropriate conditions were available for it to be performed without entering the actions. The Note to these SRs would allow startup in Mode 3 if the SR had not been performed during the required frequency, but would limit the exception to prior to entering Mode 2. The change is consistent with traveler TSTF-288 ~~WOG-60~~ Q-3.4.11-2

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.11-3 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: Change 4-05 LS 31 and Difference 3.4-39

Comment: TSTF-113 (presently Rev. 4) has not yet been approved by the NRC staff.

FLOG Response: TSTF-113 Rev. 4 revises the shutdown requirements of ITS 3.4.11 to allow the plant to reduce T_{avg} to <500ºF within 12 hours, rather than MODE 4, to address the concern of entering LCO 3.4.12 Applicability with one or more inoperable PORVs. The shutdown requirements of ITS 3.4.16 are also revised, for consistency, to allow 12 hours to reduce T_{avg} to < 500°F. ITS 3.4.11 Condition B and C Bases changes have been made to the Callaway submittal to reflect Rev. 4 of the traveler; no changes are required for any other plants' submittals. The FLOG continues to pursue the changes proposed by this traveler.

ATTACHED PAGES:

Encl 5A Traveler Status sheet

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev.3 ~~2~~	Incorporated	3.4-33	Approved by NRC Q-3.4.2-1
TSTF-28	Incorporated	3.4-22	Approved by NRC.
TSTF-54, Rev. 1	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-60	Incorporated	3.4-15	Approved by NRC.
TSTF-61	Not incorporated	NA	Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. 2~~1~~	Incorporated	3.4-31	Approved by NRC. TR-3.4-004
TSTF-93, Rev. 3~~1~~	Incorporated	3.4-17	Approved by NRC. Q-3.4.9-3
TSTF-94, Rev.1	Not incorporated	NA	Retained current TS. TR-3.4-005
~~TSTF-105~~	~~Incorporated~~	~~3.4-38~~	Q-3.4.1-1
TSTF-108, Rev. 1	Not incorporated	NA	LCO 3.4.19 does not apply.
TSTF-113, Rev. 4 3	Incorporated	3.4-39	Q-3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev.2 ~~1~~	Incorporated	3.4-36	Q-3.4.13-2
TSTF-136	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-137	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-138	Not incorporated	NA	Inconsistent with RCS loops requirements of ITS 3.4.5 and 3.4.6.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.11-4 **APPLICABILITY:** CA, CP, DC, WC

REQUEST Change 4-09 LS-36, Difference 3.4-47, Change 3-04 and Difference 3.4-31

Comment: WOG-87 has not yet become a TSTF.

FLOG Response: As discussed during a telecon with NRC Staff on July 30, 1998, the above references to DOC 3-04 and JFD 3.4-31 apply to NRC-approved traveler TSTF-87 and were not intended to be questioned here. Additional changes have recently been added per Revision 2 of WOG-87 and are included in the attached pages below. The addition of the Note to the block valve Action Statement is considered to be administrative in nature as it reflects current plant practice. WOG-87, Revision 2, has been approved by the TSTF group and is expected to be submitted to the NRC expeditiously. Given the nature of the Notes added to the PORV block valve Required Actions and Surveillance Requirement, the FLOG continues to pursue the changes proposed by this traveler.

ATTACHED PAGES:

Encl 2	3/4 4-11
Encl 3A	8 and 8a
Encl 3B	5
Encl 4	68 and 69
Encl 5A	Traveler Status Sheet and pages 3.4-24 thru 3.4-26
Encl 5B	B 3.4-53, B 3.4-55, and B 3.4-55a
Encl 6A	8 and 8a
Encl 6B	6

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

Note: Separate Condition entry is allowed for each valve. 4-01-LS

ACTION:

a. With one or both PORV(s) inoperable ~~because of excessive seat leakage~~ and capable of being manually cycled, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and reduce T_{avg} to <500°F ~~in HOT SHUTDOWN~~ within the following 6 hours. 4-02-LS 4-05-LS

b. With one PORV inoperable ~~due to causes other than excessive seat leakage~~ and not capable of being manually cycled, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within ~~the following~~ 72 hours or be in HOT STANDBY within the next 6 hours and reduce T_{avg} to <500°F ~~in HOT SHUTDOWN~~ within the following 6 hours. 4-02-LS 4-03-M 4-05-LS

c. With both PORV(s) inoperable ~~due to causes other than excessive seat leakage~~ and not capable of being manually cycled, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within ~~the next~~ 6 hours and reduce T_{avg} to <500°F ~~in HOT SHUTDOWN~~ within the following 6 hours. 4-02-LS 4-03-M 4-05-LS

d. With one or both block valves inoperable* within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore any remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and reduce T_{avg} to <500°F ~~in HOT SHUTDOWN~~ within the following 6 hours. 4-05-LS

e. The provisions of Specification 3.0.4 are not applicable.

* ACTION d does not apply when block valve(s) is inoperable solely as a result of complying with ACTIONS b or c.

4-09-LS
Q-3.4.11-4

CHANGE NUMBER	NSHC	DESCRIPTION
4-08	LS-34	Consistent with traveler ~~WOG-60~~ TSTF-288, the requirement to perform the 92 day surveillance of the pressurizer PORV block valves and the 18 month surveillance of the pressurizer PORVs (i.e., perform one complete cycle of each valve) is revised to indicate that the surveillance is only required to be performed in MODES 1 and 2. This is consistent with the recommendations of Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10CFR50.54(f)," June 25, 1990. Q-3.4.11-2
4-09	LS-36	The requirement to perform the 92 day surveillance of pressurizer PORV block valves (i.e., perform one complete cycle of each block valve) is revised such that it is not required if the block valve is closed to meet ACTION a of the current TS LCO 3.4.4. This change is acceptable because no credit is taken for the automatic actuation of the PORV in Modes 1, 2, or 3. Credit is taken for manual operation of the PORVs during the Steam Generator Tube Rupture (SGTR) accident. However, the capability to manually cycle the PORVs will be unaffected by this change. This change will not affect the ability of the block valve to open, if closed to meet ACTION a, in the mitigation of an SGTR. Deferral of the block valve cycling surveillance will not diminish the design capability of the block valve to open against differential pressures that would be present after an SGTR since the block valves are capable of opening against 2485 psig, the safety valve lift pressure, whereas pressurizer pressure decreases after an SGTR. Q-3.4.11-4 In addition, a Note is added to ACTION [d] stating that it does not apply when the block valve(s) are inoperable solely as a result of its power being removed per ACTIONS [b or c] as a result of an inoperable PORV(s). In this scenario ACTION [b or c] is entered as a result of an inoperable PORV(s). If one PORV were inoperable and incapable of being manually cycled (per the change discussed under DOC 4-02-LS-6), ACTION [b] would be entered at time zero, t_0. ACTION [b] would close the associated block valve and remove its power within time t_0 + 1 hour. If, as a result of block valve power removal per ACTION [b], ACTION [d] were then entered, ACTION [d] would require the associated PORV to be placed in manual control within time t_0 + 2 hours. However, the reason for originally entering ACTION [b] is that the associated PORV is inoperable and can't be manually cycled, thus there is

Q-3.4.11-4

CHANGE NUMBER	NSHC	DESCRIPTION
		nothing to be gained by placing the PORV in manual control. The PORV inoperability may be such that the PORV can't be placed in manual control (e.g., blown control power fuse), in which case neither this portion of ACTION [d] nor block valve restoration can be met. In addition, the portion of ACTION [d] requiring block valve restoration can't be satisfied as long as power is removed from the block valve. Restoring the PORV to OPERABLE status within time t_0 + 72 hours allows the plant to exit ACTION [b]. If power were not restored to the block valve at this time, the new Note on ACTION [d] would have no standing and ACTION [d] would be entered. Similar conclusions can be drawn for the relationship between ACTIONS [c and d]. If ACTION [c] is the original ACTION entered, there is nothing to be gained by placing both PORVs in manual control and the block valves can't be restored with their power removed. With ACTION [d] not satisfied, the plant must go to MODE 3, but ACTION [c] would have already had the plant in MODE 3 two hours earlier. Therefore, there is no compensatory action associated with cascading to the block valve ACTION [d] when the sole inoperability is with the PORV(s). This change is consistent with traveler WOG-87.
5-01	A	Not Applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
5-02	A	Not Applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
5-03	A	Not Applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
6-01	M	Consistent with NUREG-1431, this change adds the requirement to perform an RCS water inventory balance every 24 hours when the [sump level detector] is inoperable. This is a new requirement and is more restrictive than the current TS.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
4-04 LG	This change moves the requirement to perform channel calibration of PORV actuation instrumentation to a licensee controlled document.	Yes - Moved to the FSAR.	Yes - Moved to the TRM.	Yes - Moved to the USAR.	Yes - Moved to the FSAR.
4-05 LS-31	The shutdown requirement of CTS 3.4.4 would require the plant to go to reduce T_{avg} to <500°F within 12 hours, rather than go to MODE 4, to address the concern of entering [LTOP] LCO Applicability with inoperable PORVs. For consistency the shutdown requirements of CTS LCO [3.4.7] would be similarly revised.	Yes	Yes	Yes	Yes
4-06 LS-32	This change provides a 72 hour completion time to restore an inoperable block valve, with the PORV placed in manual control mode. The current TS requires the block valve to be restored within one hour, or remove power from the solenoid.	Yes	No - Already part of CTS.	No - Already part of CTS.	No - Already part of CTS.
4-07 LS-33	This change provides a two hour completion time for restoring an inoperable block valve when more than one block valve is inoperable, and 72 hours to restore the remaining valves. The current TS requires the block valve to be restored within one hour for one or more valves inoperable.	Yes	No - Already part of CTS.	No - Already part of CTS.	No - Already part of CTS.
4-08 LS-34	Consistent with traveler TSTF-288 ~~WOG-60~~, the requirement to perform the PORV and block valve cycling surveillances is revised such that the surveillances are only required to be performed in MODES 1 and 2.	Yes	Yes	Yes	Yes Q-3.4.11-2
4-09 LS-36	Consistent with traveler WOG-87, the requirement to perform the 92 day surveillance of the pressurizer block valves (i.e., perform one complete cycle of each block valve) is revised such that it is not required if the block valve is closed to meet ACTION a of the current TS LCO 3.4.4. In addition, a Note is added to ACTION [d] to prevent entry solely due to inoperable PORV(s) under Actions [b or c].	Yes	Yes	Yes	Yes Q-3.4.11-4
5-01 A	This change moves the Steam Generator Tube Surveillances to SR 3.4.13.2 and the Administrative Controls Sections 5.5.9 and 5.6.10.	Yes	No - Same as CPSES change 1-14-A for CTS Section 3/4.0.	Yes	Yes

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS-36
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with traveler WOG-87, the requirement to perform the 92 day surveillance of pressurizer PORV block valves (i.e., perform one complete cycle of each block valve) is revised such that it is not required if the block valve is closed to meet ACTION a of the current TS LCO 3.4.4. In addition, a Note is added to ACTION [d] to prevent entry solely due to inoperable PORV(s) under Actions [b or c]. Q-3.4.11-4

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?

The proposed change adds a relaxation to the surveillance associated with the pressurizer PORV block valves. The quarterly valve cycling will no longer be required if the block valve is closed per any ACTION of the LCO. No credit is taken for the automatic actuation of the PORV in Modes 1, 2, or 3. Credit is taken for manual operation of the PORVs during the Steam Generator Tube Rupture (SGTR) accident. However, the capability to manually cycle the PORVs will be unaffected by this change. This change will not affect the ability of the block valve to open, if closed to meet ACTION a, in the mitigation of an SGTR. Deferral of the block valve cycling surveillance will not diminish the design capability of the block valve to open against differential pressures that would be present after an SGTR since the block valves are capable of opening against

V. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS-36
(continued)

2485 psig, the safety valve lift pressure, whereas pressurizer pressure decreases after an SGTR [ ]. The lack of quarterly block valve cycling, which could extend to a complete cycle since ACTION a allows continued operation with the block valves closed, does not decrease the likelihood of successful pressurizer relief since power remains available to the block valve motor operator(s) and the surveillance frequency for the PORVs can be as long as 18 months (tested during each cold shutdown per the IST plan). Quarterly cycling could make PORV seat leakage worse; if the block valve were to subsequently be unable to close, this surveillance could unnecessarily challenge RCS and PRT integrity. The addition of the Note to ACTION [d] stating that it does not apply when the block valve(s) is inoperable solely as a result of power being removed per ACTIONS [b or c] as a result of an inoperable PORV(s) eliminates operator distraction caused by the required performance of activities with no safety benefit. This change has no effect on the successful mitigation of an SGTR since the initial premise is that the PORV(s) is unavailable. Elimination of operator actions that have no safety benefit result in an overall benefit to plant safety. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. Q-3.4.11-4

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

The only accidents that are potentially associated with this proposed change are those related to a loss of pressurizer relief function. This change does not introduce any new overpressure accidents and the existing analyses remain valid. Thus, the proposed change does not create the possibility of a new or different kind of accident from those 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. The automatic actuation of the PORVs is not credited in the accident analyses for Modes 1, 2, or 3. The PORVs will remain capable of being manually cycled. The margin of safety established by the LCOs also remains unchanged. Thus there is no reduction in the margin of safety from that previously established.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS-36" 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.

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-151 Rev. 1	Incorporated	NA	TR-3.4-009
TSTF-153	Incorporated	3.4-01	Approved by NRC TR-3.4-009
TSTF-162	Incorporated	NA	Approved by NRC. TR-3.4-006
~~WOG-51, Rev.1~~ TSTF-285	Incorporated	3.4-45 & 3.4-~~23~~52	See also CNs 3.4-18 and 3.4-20. 3.4-12-2
~~WOG-60~~ TSTF-288	Incorporated	3.4-35	Q-3.4.11-2
~~WOG-67, Rev. 1~~ TSTF-233	Incorporated	3.4-10	Approved by NRC. DCPP only. TR-3.4-009
WOG-87 Rev. 2	Incorporated	3.4-47	Q-3.4.11-4
~~WOG-99~~ TSTF-282	Incorporated	3.4-40	Applicable to Callaway and Wolf Creek only. Q-3.4.1-2
~~WOG-100~~TSTF-280	Incorporated	3.4-49	Q-3.4.12-1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One block valve inoperable.	-----------NOTE----------- Required Actions do not apply when block valve is inoperable solely as a result of complying with Required Actions B.2 or E.3. ------------------------------	
	C.1 Place associated PORV in manual control.	1 hour
	AND	
	C.2 Restore block valve to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Initiate action to restore PORV(s) and block valve to OPERABLE status	Immediately
	AND	
	D.~~1~~2 Be in MODE 3.	6 hours
	AND	
	D.~~23 Be in MODE 4.~~ Reduce T_{avg} to <500°F.	12 hours

3.4-47
Q-3.4.11-4

3.4-39

CONDITION	REQUIRED ACTION	COMPLETION TIME	
E. Two ~~[or three]~~ PORVs inoperable and not capable of being manually cycled.	E.1 Initiate action to restore one PORV to OPERABLE status	Immediately	3.4-39 B-PS
	AND		
	E.~~1~~2 Close associated block valves.	1 hour	
	AND		
	E.~~2~~3 Remove power from associated block valves.	1 hour	
	AND		
	E.~~3~~4 Be in MODE 3	6 hours	
	AND		
	E.~~4~~5 ~~Be in MODE 4.~~ Reduce T_{avg} to <500°F.	12 hours	3.4-39
F. More than one block valve inoperable.	-----NOTE----- Required Actions do not apply when block valve is inoperable solely as a result of complying with Required Actions B.2 or E.3. ------------		3.4-47 Q-3.4.11-4
	F.1 Place associated PORVs in manual control.	1 hour	
	AND		
	F.2 Restore one block valve to OPERABLE status ~~[if three block valves are inoperable]~~.	2 hours	B-PS
	~~AND~~		
	~~F.3 Restore remaining block valve(s) to OPERABLE status.~~	~~72 hours~~	B-PS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
G. Required Action and associated Completion Time of Condition F not met.	G.1	Initiate action to restore one block valve to OPERABLE status.	Immediately	3.4-39
	AND			
	G.~~1~~2	Be in MODE 3.	6 hours	
	AND			
	G.~~2~~3	~~Be in MODE 4.~~ Reduce T_{avg} to $<$500°F.	12 hours	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.11.1	------------------NOTES------------------ 1. Not required to be ~~met~~performed with block valve closed in accordance with the Required Action of Condition this LCO. ~~A.~~ ~~B or E.~~ 2. Only required to be performed in MODES 1 and 2 --- Perform a complete cycle of each block valve. ~~except with the block valve closed in accordance with the Required Actions of Conditions B or E.~~	92 days	3.4-47 3.4-47 Q-3.4.11-4 3.4-35 3.4-47 3.4-47 Q-3.4.11-4
	------------------NOTE------------------ Only required to be performed in MODES 1 and 2 --- Perform a complete cycle of each PORV.		3.4-35
SR 3.4.11.2	Perform a complete cycle of each PORV.	18 months	B
~~SR 3.4.11.3~~	~~Perform a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems.~~	~~[18] months~~	B-PS

reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to ~~a MODE in which the LCO does not apply.~~ at least MODE 3 with T_{avg} < 500°F, as required by Condition D.

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs ~~are~~ may not be capable of mitigating an ~~overpressure~~ event if the inoperable block valve is not fully open .~~when placed in manual control.~~ If the block valve is restored within the Completion Time of 72 hours, the power will be restored and the PORV restored to OPERABLE status. If it cannot be restored within this additional time, the plant must be brought to ~~a MODE in which the LCO does not apply,~~ at least MODE 3 with T_{avg} < 500°F, as required by Condition D. The Required Actions are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. Q-3.4.11-4

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then 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 with T_{avg} <500°F within 6 hours ~~and to MODE 4 within 12 hours~~. Additional action is required to be initiated

(continued)

BASES (continued)

system during this time and provide the operator time to correct the situation. The Required Actions are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition.

Q-3.4.11-4

G.1 and G.2

If the Required Actions of Condition F are not met, then 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 with T_{avg} <500°F within 6 hours ~~and to MODE 4 within 12 hours~~. Additional action is required to be immediately initiated to restore the inoperable valve(s) to OPERABLE status. This will ensure expedient measures are taken to re-establish OPERABLE PORVs and block valves while maintaining plant conditions above MODE 4 but less than 500°F. 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. In MODES 4 ~~and~~ 5 and 6 (with the reactor vessel head on), automatic ~~maintaining~~ PORV OPERABILITY may be required. See LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed. ~~, except when the block valve(s) is closed in accordance with the Required Actions of Condition B or E, if needed.~~ The basis for the Frequency of 92 days is the ASME Code, Section XI (Ref. 3). ~~If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an otherwise inoperable PORV that is incapable of being manually cycled, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status. (i.e., completion of the Required Actions fulfills the SR).~~

Q-3.4.11-4

This SR is modified by two Notes. ~~The~~ Note 1 modifies this SR by stating that it is not required to be ~~met~~performed with the block valve closed, in accordance with the Required

Q-3.4.11-2

Q-3.4.11-4

BASES (continued)

~~Actions~~ of this LCO. ~~Condition A.this LCO~~ Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable. Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the surveillance to be performed in ~~any~~ MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. ~~below MODE 2~~. In accordance with References 4, 5 and 6, administrative controls require this test be performed in MODES 3, 4 or 5 to adequately simulate operating temperature and pressure effects on PORV operation.

Q-3.4.11-4

Q-3.4.11-2

CHANGE NUMBER	JUSTIFICATION
3.4-46	Consistent with current TS 3/4.1.1.4, "Minimum Temperature for Criticality," ITS LCO 3.4.2 and its Condition A and SR 3.4.2.1 are modified to refer to "operating" RCS loops. Adopting the current TS wording is acceptable since valid T_{avg} measurements are not obtainable for a non-operating loop.
3.4-47	ITS SR 3.4.11.1 contains a Note which exempts the cycling of the block valve when it is closed in accordance with Required Actions of Condition B or E of LCO 3.4.11. However, Required Action A.1 also directs closure of the block valve when one or more PORVs are inoperable and capable of being manually cycled. The SR Note should also exempt performance when the block valve is closed in accordance with Required Action A.1 as the block valve should not be opened when the PORV is inoperable. This change is consistent with NUREG-1430 and NUREG-1432 in as much as the block valve ~~cycling is exempted under Conditions A, B, and E.~~ ~~Since power to the block valve(s) is maintained in Required Action A.1~~ The Note to SR 3.4.11.1 will be revised to not require the surveillance performance if the block valve(s) is closed per the LCO. ~~Condition A.1. Since power to the block valve(s) is removed in Required Actions B.2 and E.3, the surveillance can not be met. Given the wording change "met" to "performed" in the Note, the wording of SR 3.4.11.1 is revised to accommodate the Condition B and E exception.~~ In addition, Notes are added to Conditions C and F stating that these Required Actions don't apply when the block valve(s) is inoperable solely as a result of its power being removed per Required Actions B.2 or E.3 as a result of an inoperable PORV(s). In this scenario Condition B or E is entered as a result of an inoperable PORV(s). If one PORV were inoperable and incapable of being manually cycled, Condition B would be entered at time zero, t_0. Required Actions B.1 and B.2 would close the associated block valve and remove its power within time t_0 + 1 hour. If, as a result of block valve power removal per Required Action B.2, Condition C were then entered, Required Action C.1 would require the associated PORV to be placed in manual control within time t_0 + 2 hours. However, the reason for originally entering Condition B is that the associated PORV is inoperable and can't be manually cycled, thus there is nothing to be gained by placing the PORV in manual control. The PORV inoperability may be such that the PORV can't be placed in manual control (e.g., blown control power fuse), in which case Required Actions C.1 and C.2 can't be met. In addition, Required Action C.2 (restore block valve to OPERABLE status) can't be satisfied as long as power is removed from the block valve. Restoring the PORV to OPERABLE Q-3.4.11-4

CHANGE NUMBER	JUSTIFICATION
	status within time t_0 + 72 hours allows the plant to exit Condition B. If power were not restored to the block valve at this time, the new Note on Condition C would have no standing and Condition C would be entered. Similar conclusions can be drawn for the relationship between Conditions E and F. If Condition E is the original Condition entered, there is nothing to be gained by Required Action F.1 and Required Action F.2 can't be satisfied with block valve power removed. With F.2 not satisfied, Required Action G.2 would require the plant to be in MODE 3, but Required Action E.4 would have already had the plant in MODE 3 two hours earlier. This change is consistent with ~~traveler WOG-87~~. Q-3.4.11-4
3.4-48	A note is added to ITS 3.4.8 ACTIONS, indicating that entry into MODE 5 Loops Not Filled from MODE 5 Loops Filled is not permitted while LCO 3.4.8 is not met. The addition of this note is based on the performance of a plant specific LCO 3.0.4 matrix (see CN 1-02-LS-1 of the CTS 3/4.0 package). 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 3.4.8 is modified by a Note stating: "While this LCO is not met, entry into MODE 5 Loops Not Filled from MODE 5 Loops Filled is not permitted." The transition from MODE 5 (loops filled) to MODE 5 (loops not filled) removes the steam generators as a decay heat removal system while the RHR System is potentially degraded. Therefore, the Note ensures that the transition is precluded if LCO 3.4.7.b (two SGs) were chosen (in lieu of the second RHR loop) to ensure decay heat removal capability prior to draining the RCS. Q-3.4.8-1
3.4-49	LCO 3.4.12 "[LTOP] System", provides four different methods for pressure relief. Any of the four methods may be used. However, Surveillance Requirement 3.4.12.5 requires testing whether or not the equipment is being credited to meet the LCO. The proposed change adds the words "required" to the Surveillance to exempt its performance if the equipment to be tested is not being used to meet the LCO. In addition, two editorial changes were made. The LCO requirement presentation was clarified. Also, the Note to SR 3.4.12.8 was revised to replace "required" to be met" with "required to be performed" since the "performed" nomenclature is appropriate here, consistent with the CTS. This change is consistent with traveler TSTF-280 ~~WOG-100~~. Q-3.4.12-1

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-47	ITS SR 3.4.11.1 contains a Note which exempts the cycling of the block valve when it is closed in accordance with Required Actions of Condition B or E of LCO 3.4.11. However, Required Action A.1 also directs closure of the block valve when one or more PORVs are inoperable and capable of being manually cycled. The SR Note should also exempt performance when the block valve is closed in accordance with Required Action A.1 ~~as the block valve should not be opened when the PORV is inoperable.~~ In addition, Notes are added to Condition C and F to prevent entry solely due to inoperable PORV(s) under Conditions B or E.	Yes	Yes	Yes	Yes Q-3.4.11-4
3.4-48	A note is added to ITS 3.4.8 ACTIONS, indicating that entry into MODE 5 Loops Not Filled from MODE 5 Loops Filled is not permitted while LCO 3.4.8 is not met.	Yes	Yes	Yes	Yes
3.4-49	This change reorganizes the presentation of ITS LCO 3.4.12, adds the word "required" to ITS SR 3.4.12.5, and changes the word "met" to "performed" in ITS SR 3.4.12.8.	Yes	Yes	Yes	Yes
3.4-50	This change is consistent with current TS SR 4.4.9.3.3. The 12 hour frequency applies to vent pathway(s) that are "not locked, sealed, or otherwise secured in the open position. The wording added to ITS SR 3.4.12.5 is also consistent with the format used in similar ITS 3.6 SRs. The 31 day frequency is also revised to be consistent with current TS SR 4.4.9.3.3.	No - adopting ITS format.	No - adopting ITS format.	Yes	Yes
3.4-51	The Note for SR 3.4.1.4 is removed. This is consistent with DCPP CTS 4.2.3.5. DCPP conducts a measured RCS total flowrate verification on the ~~18~~ 24 month frequency.	Yes	No	No	No DC-ALL-005
3.4-52	Consistent with traveler TSTF-285 ~~WOG-51~~, the Note concerning accumulator isolation is reworded for clarity and is moved from the APPLICABILITY to the LCO.	No - See CN 3.4-45	Yes	No - See CN 3.4-45	No - See CN 3.4-45 Q-3.4.12-2

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.11-6 **APPLICABILITY:** CA, CP, WC

REQUEST Difference 3.4-[39] (Wolf Creek, Comanche Peak and Callaway)

Comment: This difference does not address the addition of the "Immediately" in Required Actions D.1, E.1, and G.1 of ITS 3.4.11

FLOG Response: This RAI refers to JFD 3.4-49; it should refer to JFD 3.4-39. Difference 3.4-39 is revised with the addition of the following:

> "New initial Required Actions are added to Conditions D, E and G to immediately initiate actions for restoration of the inoperable PORV(s) (and or PORV block valves) to OPERABLE status. These immediate actions will ensure expedient measures are taken to re-establish the operability of the PORV(s) (and PORV block valves) while maintaining plant conditions above MODE 4 but less than 500°F."

ATTACHED PAGES:

Encl 6A 7

CHANGE NUMBER	JUSTIFICATION
3.4-36	SR 3.4.13.1 and ACTIONS for LCO 3.4.15 are revised with the addition of a note per TSTF-116. The note addresses the concern that an RCS water inventory balance cannot be meaningfully performed unless the unit is operating at or near steady state conditions. The note added to the surveillance provides an exception for operation at less than steady state conditions. The RCS water inventory balance will only be allowed to be deferred for 12 hours after re-establishing steady state conditions.
3.4-37	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
3.4-38	~~Consistent with TSTF-105, the details on the method by which the RCS flow rate are verified is moved from the SR 3.4.1.4 to the Bases. Moving this information to the Bases, allows the use of precision heat balances, elbow taps, and other acceptable methods in order to perform this verification and is consistent with the NUREG-1431 philosophy of moving clarifying information and descriptive details out of the TS to the Bases.~~ Not used Q-3.4.1-1
3.4-39	The shutdown requirements of ITS 3.4.11 would require the plant to reduce T_{avg} to <500°F within 12 hours, rather than MODE 4, to address the concern of entering [LTOP] LCO 3.4.12 Applicability with inoperable PORVs. New initial Required Actions are added to Conditions D, E and G to immediately initiate actions for restoration of the inoperable PORV(s) (and or PORV block valves) to OPERABLE status. These immediate actions will ensure expedient measures are taken to re-establish the operability of PORV(s) (and PORV block valves) while maintaining plant conditions above MODE 4 but less than 500°F. For consistency, the shutdown requirements of ITS 3.4.16 are also revised to allow 12 hours to reduce T_{avg} to <500°F. This change is consistent with TSTF-113. Q-3.4.11-6
3.4-40	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
3.4-41	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
3.4-42	Not applicable to CPSES. See Conversion Comparison Table (enclosure 6B).
3.4-43	Not applicable to CPSES. See Conversion Comparison Table (enclosure 6B).

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.12-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: Difference 3.4-49

Comment: WOG-100 has not yet become a TSTF.

FLOG Response: WOG-100 has been approved by the TSTF and is designated as TSTF-280. This traveler has been submitted to the NRC and is under review. The proposed wording in TSTF-280 was modified from WOG-100, and these modifications have been incorporated into the ITS (added "or" to LCO list and SR 3.4.12.5 Note was deleted). The FLOG continues to pursue the changes proposed by this traveler.

ATTACHED PAGES:

Encl. 5A	Traveler Status page, 3.4-28 and 3.4-32
Encl 5B	B 3.4-64, B 3.4-65 and B 3.4-69
Encl. 6A	8a

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-151 Rev. 1	Incorporated	NA	TR-3.4-009
TSTF-153	Incorporated	3.4-01	Approved by NRC TR-3.4-009
TSTF-162	Incorporated	NA	Approved by NRC. TR-3.4-006
~~WOG-51, Rev.1~~ TSTF-285	Incorporated	3.4-45 & 3.4-~~23~~52	See also CNs 3.4-18 and 3.4-20. Q-3.4-12-2
~~WOG-60~~ TSTF-288	Incorporated	3.4-35	Q-3.4.11-2
~~WOG-67, Rev. 1~~ TSTF-233	Incorporated	3.4-10	Approved by NRC. DCPP only. TR-3.4-009
WOG-87 Rev. 2	Incorporated	3.4-47	Q-3.4.11-4
~~WOG-99~~ TSTF-282	Incorporated	3.4-40	Applicable to Callaway and Wolf Creek only. Q-3.4.1-2
~~WOG-100~~ TSTF-280	Incorporated	3.4-49	Q-3.4.12-1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of ~~one high pressure injection (HPI) pump~~ zero safety injection pumps and ~~one~~ two charging pumps capable of injecting into the RCS and the accumulators isolated and ~~either a or b below.~~ one of the following pressure relief capabilities: B-PS 3.4-49

~~a. Two RCS relief valves, as follows:~~

~~1.~~a. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

~~2.~~b. Two residual heat removal (RHR) suction relief valves with setpoints ≥ 436.5 psig and ≤ 463.5 psig, or B

~~3.~~c. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint ≥ 436.5 psig and ≤ 463.5 psig~~.~~, or B 3.4-49 Q-3.4.12-1

~~b.~~d. The RCS depressurized and an RCS vent of ≥ ~~2.07~~ 2.98 square inches. B-PS

-----NOTE-----

Accumulator may be unisolated~~ion is only required~~ when accumulator pressure is less than ~~greater than or equal to~~ the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. 3.4-52 Q-3.4.12-2

APPLICABILITY: MODE 4 ~~when all RCS cold leg temperature is ≤ [275]°F~~, 3.4-05
MODE 5,
MODE 6 when the reactor vessel head is on.

-----NOTE----- 3.4-52

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

The LCO is not applicable when all RCS cold leg temperatures are > 320°F and the following conditions are met: 3.4-05

a. At least one reactor coolant pump is in operation, and

b. Pressurizer level is ≤ 92%, and

SURVEILLANCE		FREQUENCY	
SR 3.4.12.4	Verify RHR suction isolation valves ~~is~~ are open for each required RHR suction relief valve.	72~~12~~ hours	3.4-08 ~~3.4-09~~ B-PS Q-3.4.12-3
SR 3.4.12.5	~~-------------------NOTE-------------------~~ ~~Only required to be performed when complying with LCO 3.4.12.db.~~ ~~--~~ Verify required RCS vent ≥ ~~2.07~~ 2.98 square inches open.	12 hours for unlocked open vent valve(s) AND 31 days for locked open vent valve(s)	3.4-49 Q-3.4.12-1 B-PS 3.4-49
SR 3.4.12.6	Verify PORV block valve is open for each required PORV.	72 hours	
SR 3.4.12.7	Not Used ~~Verify associated RHR suction isolation valve is locked open with operator power removed for each required RHR suction relief valve~~.	~~31 days~~	~~3.4-09~~ B-PS Q-3.4.12-3

APPLICABLE SAFETY ANALYSES

RCS Vent Performance (continued)

The LTOP System satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10CFR50.36(c)(2)(ii).

LCO

This LCO requires that the LTOP System is OPERABLE. ~~The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE.~~ Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires ~~one HPI~~ zero safety injection pumps and ~~one~~ a maximum of two charging pumps be capable of injecting into the RCS, and all accumulator discharge isolation valves be closed and immobilized~~. When~~, when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. Q-3.4.Gen-1 Q-3.4.12-2

The LCO is modified by a Note stating that the accumulator ~~isolation is only required~~ may be unisolated when the accumulator pressure is less than ~~more than or at~~ the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

~~a. Two RCS relief valves, as follows:~~ Q-3.4.12-1

a.~~1~~. Two OPERABLE PORVs; or

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

b~~2~~. Two OPERABLE RHR suction relief valves; or

An RHR suction relief valve is OPERABLE for LTOP when its RHR suction isolation valves ~~and its RHR suction valve~~ are open, its setpoint is at or between 436.5 psig and 463.5 psig, and testing has proven its ability to open at this setpoint.

(continued)

BASES

LCO (continued)

c~~3~~. One OPERABLE PORV and one OPERABLE RHR suction relief valve; or

Q-3.4.12-1

d~~b~~. A ~~depressurized RCS and an RCS~~ vent.

An RCS vent is OPERABLE when open with an area of ≥ ~~2.07~~ 2.98 square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 ~~when any RCS cold leg temperature is ≤ [275]°F~~, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above ~~275~~ 320°F. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above ~~275~~ 320°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

~~The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions~~.

Q-3.4.12-2

The Applicability is modified by a Note stating that the LCO is not applicable above 320°F when at least one reactor coolant pump is in operation, pressurizer level is ≤ 92%, and the plant heatup rate is limited to 60°F in any one hour period. These conditions are included in the LTOP analysis allowing LTOP to be inoperable above 320°F.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.4.12.4

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its ~~RHR suction valve and~~ RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. ~~(Refer to SR 3.4.12.7 for the RHR suction isolation valve Surveillance.)~~ This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

The RHR suction isolation valves ~~is~~ are verified to be opened every ~~12~~ 72 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valve remains open.

The ASME Code, Section XI (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.5

The RCS vent of ≥ ~~2.07~~ 2.98 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that ~~cannot be~~ is not locked, sealed, or otherwise secured in the open position.

b. Once every 31 days for other vent paths (e.g., a valve that is locked, sealed, or otherwise secured in position). A removed pressurizer safety valve or open manway also fits this category.

~~The~~ Any passive vent path arrangement must only be open when required to be OPERABLE. This Surveillance is required ~~to be performed~~ if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12db.

Q-3.4.12-1

SR 3.4.12.6

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open

(continued)

CHANGE NUMBER | JUSTIFICATION

status within time t_0 + 72 hours allows the plant to exit Condition B. If power were not restored to the block valve at this time, the new Note on Condition C would have no standing and Condition C would be entered. Similar conclusions can be drawn for the relationship between Conditions E and F. If Condition E is the original Condition entered, there is nothing to be gained by Required Action F.1 and Required Action F.2 can't be satisfied with block valve power removed. With F.2 not satisfied, Required Action G.2 would require the plant to be in MODE 3, but Required Action E.4 would have already had the plant in MODE 3 two hours earlier. This change is consistent with traveler WOG-87. Q-3.4.11-4

3.4-48 A note is added to ITS 3.4.8 ACTIONS, indicating that entry into MODE 5 Loops Not Filled from MODE 5 Loops Filled is not permitted while LCO 3.4.8 is not met. The addition of this note is based on the performance of a plant specific LCO 3.0.4 matrix (see CN 1-02-LS-1 of the CTS 3/4.0 package). 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 3.4.8 is modified by a Note stating: "While this LCO is not met, entry into MODE 5 Loops Not Filled from MODE 5 Loops Filled is not permitted." The transition from MODE 5 (loops filled) to MODE 5 (loops not filled) removes the steam generators as a decay heat removal system while the RHR System is potentially degraded. Therefore, the Note ensures that the transition is precluded if LCO 3.4.7.b (two SGs) were chosen (in lieu of the second RHR loop) to ensure decay heat removal capability prior to draining the RCS. Q-3.4.8-1

3.4-49 LCO 3.4.12 "[LTOP] System", provides four different methods for pressure relief. Any of the four methods may be used. However, Surveillance Requirement 3.4.12.5 requires testing whether or not the equipment is being credited to meet the LCO. The proposed change adds the words "required" to the Surveillance to exempt its performance if the equipment to be tested is not being used to meet the LCO. In addition, two editorial changes were made. The LCO requirement presentation was clarified. Also, the Note to SR 3.4.12.8 was revised to replace "required" to be met" with "required to be performed" since the "performed" nomenclature is appropriate here, consistent with the CTS. This change is consistent with traveler TSTF-280 ~~WOG-100~~. Q-3.4.12-1

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.12-2 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: Differences 3.4-23 and 3.4-45

Comment: WOG-51 Rev. 1 has not yet become a TSTF.

FLOG Response: WOG-51, Rev.2 is designated as TSTF-285. This traveler has been submitted to the NRC and is currently under review. The proposed wording in TSTF-285 was modified from WOG-51, Rev 2, and these modifications have been incorporated into the ITS. The FLOG continues to pursue the changes proposed by this traveler.

ATTACHED PAGES:

Encl 5A	Traveler Status sheet, 3.4-28
Encl 5B	B 3.4-64 and B 3.4-65
Encl 6A	9
Encl 6B	6

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS	
TSTF-151 Rev. 1	Incorporated	NA		TR-3.4-009
TSTF-153	Incorporated	3.4-01	Approved by NRC	TR-3.4-009
TSTF-162	Incorporated	NA	Approved by NRC.	TR-3.4-006
~~WOG-51, Rev.1~~ TSTF-285	Incorporated	3.4-45 & 3.4-~~23~~52	See also CNs 3.4-18 and 3.4-20.	Q-3.4-12-2
~~WOG-60~~ TSTF-288	Incorporated	3.4-35		Q-3.4.11-2
~~WOG-67, Rev. 1~~ TSTF-233	Incorporated	3.4-10	Approved by NRC. DCPP only.	TR-3.4-009
WOG-87 Rev. 2	Incorporated	3.4-47		Q-3.4.11-4
~~WOG-99~~ TSTF-282	Incorporated	3.4-40	Applicable to Callaway and Wolf Creek only.	Q-3.4.1-2
~~WOG-100~~TSTF-280	Incorporated	3.4-49		Q-3.4.12-1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12

An LTOP System shall be OPERABLE with a maximum of ~~one high pressure injection (HPI) pump~~ zero safety injection pumps and ~~one~~ two charging pumps capable of injecting into the RCS and the accumulators isolated and ~~either a or b below.~~ one of the following pressure relief capabilities: B-PS 3.4-49

~~a. Two RCS relief valves, as follows:~~

~~1.~~a. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

~~2.~~b. Two residual heat removal (RHR) suction relief valves with setpoints ≥ 436.5 psig and ≤ 463.5 psig, or B

~~3.~~c. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint ≥ 436.5 psig and ≤ 463.5 psig~~.~~, or B 3.4-49 Q-3.4.12-1

~~b.~~d. The RCS depressurized and an RCS vent of ≥ ~~2.07~~ 2.98 square inches. B-PS

-----NOTE-----

Accumulator may be unisolat~~ion is only required~~ when accumulator pressure is less than ~~greater than or equal to~~ the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. 3.4-52 Q-3.4.12-2

APPLICABILITY:

MODE 4 ~~when all RCS cold leg temperature is ≤ [275]°F~~, 3.4-05
MODE 5,
MODE 6 when the reactor vessel head is on.

-----NOTE-----

~~Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.~~ 3.4-52

The LCO is not applicable when all RCS cold leg temperatures are > 320°F and the following conditions are met: 3.4-05

BASES

APPLICABLE SAFETY ANALYSES

RCS Vent Performance (continued)

The LTOP System satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10CFR50.36(c)(2)(ii).

LCO

This LCO requires that the LTOP System is OPERABLE. ~~The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE.~~ Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires ~~one HPI~~ zero safety injection pumps and ~~one~~ a maximum of two charging pumps be capable of injecting into the RCS, and all accumulator discharge isolation valves be closed and immobilized~~. When~~, when accumulator pressure is greater than or equal to the maximum RCS ~~pressure~~ for the existing RCS cold leg temperature allowed in the PTLR.

Q-3.4.Gen-1

Q-3.4.12-2

The LCO is modified by a Note stating that the accumulator ~~isolation is only required~~ may be unisolated when the accumulator pressure is less than ~~more than or at~~ the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

~~a. Two RCS relief valves, as follows:~~

Q-3.4.12-1

a.~~1~~. Two OPERABLE PORVs; or

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

b~~2~~. Two OPERABLE RHR suction relief valves; or

An RHR suction relief valve is OPERABLE for LTOP when its RHR suction isolation valves ~~and its RHR suction valve~~ are open, its setpoint is at or between 436.5 psig and 463.5 psig, and testing has proven its ability to open at this setpoint.

(continued)

BASES

LCO (continued)

~~c~~3. One OPERABLE PORV and one OPERABLE RHR suction relief valve; or — Q-3.4.12-1

d~~b~~. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of ≥ ~~2.07~~ 2.[illegible]3 square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 ~~when any RCS cold leg temperature is ≤ [275]°F~~, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above ~~275~~ 320°F. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above ~~275~~ 320°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

~~The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions~~. — Q-3.4.12-2

The Applicability is modified by a Note stating that the LCO is not applicable above 320°F when at least one reactor coolant pump is in operation, pressurizer level is ≤ 92%, and the plant heatup rate is limited to 60°F in any one hour period. These conditions are included in the LTOP analysis allowing LTOP to be inoperable above 320°F.

(continued)

CHANGE NUMBER	JUSTIFICATION	
3.4-50	Not applicable to CPSES. See conversion comparison table (enclosure 6B).	
3.4-51	Not applicable to CPSES. See conversion comparison table (enclosure 6B).	
3.4-52	Consistent with traveler TSTF-285 ~~WOG-51~~, the Note concerning accumulator isolation is reworded for clarity and is moved from the APPLICABILITY to the LCO.	Q-3.4.12-2

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-47	ITS SR 3.4.11.1 contains a Note which exempts the cycling of the block valve when it is closed in accordance with Required Actions of Condition B or E of LCO 3.4.11. However, Required Action A.1 also directs closure of the block valve when one or more PORVs are inoperable and capable of being manually cycled. The SR Note should also exempt performance when the block valve is closed in accordance with Required Action A.1 as the block valve should not be opened when the PORV is inoperable. In addition, Notes are added to Condition C and F to prevent entry solely due to inoperable PORV(s) under Conditions B or E.	Yes	Yes	Yes	Yes Q-3.4.11-4
3.4-48	A note is added to ITS 3.4.8 ACTIONS, indicating that entry into MODE 5 Loops Not Filled from MODE 5 Loops Filled is not permitted while LCO 3.4.8 is not met.	Yes	Yes	Yes	Yes
3.4-49	This change reorganizes the presentation of ITS LCO 3.4.12, adds the word "required" to ITS SR 3.4.12.5, and changes the word "met" to "performed" in ITS SR 3.4.12.8.	Yes	Yes	Yes	Yes
3.4-50	This change is consistent with current TS SR 4.4.9.3.3. The 12 hour frequency applies to vent pathway(s) that are "not locked, sealed, or otherwise secured in the open position. The wording added to ITS SR 3.4.12.5 is also consistent with the format used in similar ITS 3.6 SRs. The 31 day frequency is also revised to be consistent with current TS SR 4.4.9.3.3.	No - adopting ITS format.	No - adopting ITS format.	Yes	Yes
3.4-51	The Note for SR 3.4.1.4 is removed. This is consistent with DCPP CTS 4.2.3.5. DCPP conducts a measured RCS total flowrate verification on the ~~18~~ 24 month frequency.	Yes	No	No	No DC-ALL-005
3.4-52	Consistent with traveler TSTF-285 ~~WOG-51~~, the Note concerning accumulator isolation is reworded for clarity and is moved from the APPLICABILITY to the LCO.	No - See CN 3.4-45	Yes	No - See CN 3.4-45	No - See CN 3.4-45 Q-3.4.12-2

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.12-3 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: Difference 3.4-09

Comment: The difference does not adequately justify not adopting STS SR 3.4.12.7. The SR is intended to apply to valves besides manual valves. Performing SR 3.4.12.4 does not verify the same status as that verified by SR 3.4.12.7.

FLOG Response: JFD 3.4-09 is not applicable to DCPP.

JFD 3.4-09 provides an incorrect justification for not adopting SR 3.4.12.7. The Surveillance Requirement to verify the RHR suction isolation valves are locked open every 31 days (when the RHR relief valves are being used for cold overpressure protection) was removed from the CTS as part of a license amendment implementing the Generic Letter 88-17 recommendation to delete the RHR autoclosure interlock (ACI). The 31 day surveillance was determined to be no longer necessary since removal of the ACI eliminates the single failure that could have isolated both RHR suction relief valves. ACI removal also reduces the probability of closure of the RHR suction isolation valves when power is available.

Also, SR 3.4.12.7 is bracketed in the STS. NUMARC 93-03, "Writer's Guide for the Restructured Technical Specifications" indicates that brackets are used in the generic Technical Specifications and Bases to indicate where plant specific input is needed. As identified in the "Methodology for Markup of NUREG-1431 Specifications" in Enclosure 5A, changes to bracketed information involve the insertion of plant specific information which is presently located in the current TS. The methodology applied by the FLOG was that a JFD was not required if the bracketed requirement/information was not in the current TS. Therefore, no justification was provided since the STS SR 3.4.12.7 was not in current TS. SR 3.4.12.4 is also bracketed in the STS. The changes being made to that surveillance involve plant specific wording changes (i.e., "isolation valves are"), which require no justification per the FLOG methodology, and the SR Frequency as discussed in JFD 3.4-08 (not applicable to DCPP).

Based on the above, JFD 3.4-09 is no longer necessary and will be replaced by "B-PS" in the Enclosure 5A markup. JFD 3.4-09 will be shown as "not used" in the Enclosure 6A and 6B markups.

Plant Specific Discussion

ACI deletion and elimination of the subject surveillance requirement was approved for CPSES in OL Amendment No. 4 dated October 21, 1991.

ATTACHED PAGES:

Encl 5A	3.4-32
Encl 6A	2
Encl 6B	1

SURVEILLANCE		FREQUENCY	
SR 3.4.12.4	Verify RHR suction isolation valves ~~is~~ are open for each required RHR suction relief valve.	72~~12~~ hours	3.4-08 ~~3.4-09~~ B-PS Q-3.4.12-3
SR 3.4.12.5	~~-----NOTE----- Only required to be performed when complying with LCO 3.4.12.db.~~ Verify required RCS vent ≥ ~~2.07~~ 2.98 square inches open.	12 hours for unlocked open vent valve(s) AND 31 days for locked open vent valve(s)	3.4-49 Q-3.4.12-1 B-PS 3.4-49
SR 3.4.12.6	Verify PORV block valve is open for each required PORV.	72 hours	
SR 3.4.12.7	Not Used ~~Verify associated RHR suction isolation valve is locked open with operator power removed for each required RHR suction relief valve.~~	~~31 days~~	~~3.4-09~~ B-PS Q-3.4.12-3

CHANGE NUMBER	JUSTIFICATION
	and does allow injection from [two] charging pumps. The changes are made consistently in the LCO, ACTIONS and SURVEILLANCE REQUIREMENTS.
3.4-07	Not applicable to CPSES. See Conversion Comparison Table (enclosure 6B).
3.4-08	The existing licensing basis as contained in the Technical Specifications requires performance of this surveillance on a frequency of 72 hours. The Westinghouse STS used to develop the plant specific TS did not address the use of RHR relief valves. The requirement in the current TS was developed as part of an LAR to remove the autoclosure interlock which, in part, proposed 72 hours as it was consistent with the SR for the pressurizer PORV block valves. The 72 hours was found to be acceptable in the SER which was enclosed in the license amendment. Plant experience has not indicated that the existing requirement is unsafe or unacceptable. The surveillance frequency does not require ~~reduction to 12 hours.~~
3.4-09	~~The plant does not have manual RHR suction isolation valves. The motor-operated suction isolation valves (2 per relief valve line) are surveilled in accordance with SR 3.4.12.4. Therefore this surveillance requirement is not used.~~ Not used Q-3.4.12-3
3.4-10	Not applicable to CPSES. See Conversion Comparison Table (enclosure 6B).
3.4-11	The plant does not have the RHR autoclosure portion of the RHR System interlock as the system was deleted from the plant design. However, the portion of the interlock which prevents the valves from opening when system pressure is in excess of the setpoint has been retained. As such the note referring the autoclosure interlock has been deleted from improved TS 3.4.14 Condition C and SR 3.4.14.2 [and SR 3.4.14.2 is modified to be consistent with LCO 3.4.12]. SR 3.4.14.3 is not used.
3.4-12	In conformance with the current TS, the RHR Isolation Valves which are RCS PIVs are excluded from being retested following an extended period of operation in MODE 5. The valves are tested on an 18 month frequency which is acceptable based on valve type, interlocks, position indication and system alarm functions. These valves meet the criteria for testing at least every 18 months but do not require additional testing based on when the unit is in cold shutdown or based upon being moved from a closed position.

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431 - SECTION 3.4

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-01	Clarifies intent of wording for the allowance to remove pumps from operation by changing "de-energized" to "removed from operation" consistent with TSTF-153.	Yes	Yes	Yes	Yes
3.4-02	This change revises Condition A of ITS 3.4.6 to cover any required loop's inoperability and adds Required Action A.2 indicating that cooldown to MODE 5 is only required if an RHR loop is OPERABLE. Condition B is deleted.	Yes	Yes	Yes	Yes
3.4-03	The current technical specifications allow 1 hour for the de-energization of all RHR pumps. [ ]	Yes	Yes	Ye	Yes
3.4-04	The symbol ">" in DCPP LCO 3.4.8 Note 1.a., is replaced by the words "at least".	Yes	No	No	No
3.4-05	This change is being made consistent with the current assumptions used in the analysis. The analysis credits three operational restrictions below 350°F, to ensure that the reactor vessel is protected.	No. These operational restrictions are not part of the current TS.	Yes	No-Similar analysis assumptions not contained in CTS.	No-Similar analysis assumptions not contained in CTS.
3.4-06	Plant specific safety analyses do not allow injection from safety injection pumps but do allow CCP injection .	Yes - DCPP analysis assumes one CCP only.	Yes - CPSES analysis assumes two CCPs.	Yes - WCNOC analysis assumes one CCP only.	Yes - Callaway analysis assumes one CCP only.
3.4-07	The word "all" in the DCPP LCO 3.4.12 APPLICABILITY is replaced by the word any.	Yes	No	No	No
3.4-08	The current licensing basis as contained in the Technical Specifications requires performance of this surveillance on a frequency of 72 hours.	No - DCPP LTOP design does not use RHR relief valves.	Yes	Yes - See Amendment No. 49.	Yes - See OL Amendment NO. 42
3.4-09	~~The plant does not have manual RHR suction isolation valves. The motor-operated suction isolation valves (2 per relief valve line) are surveilled in accordance with SR 3.4.12.4.~~ Not used	~~No - DCPP LTOP design does not use RHR relief valves.~~ NA	~~Yes~~ NA	~~Yes~~ NA	~~Yes~~ NA Q-3.4.12-3

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.12-4 **APPLICABILITY:** CA, CP, WC

REQUEST: ITS Bases 3.4.12 Applicability (Comanche Peak, Wolf Creek, and Callaway)

Comment: The intent of the addition to the end of the first paragraph of the Applicability Bases is unclear. The LCO applies if the head is on. The added discussion essentially states LTOP (COMS) protection is not needed with the head on and the bolts fully detensioned. If that is the argument then rather than adding it to the Bases discussion, the case should be made for modifying the LCO Applicability.

FLOG Response: This comment is not applicable to Comanche Peak as this information was not in the CPSES ITS Bases. The Applicability for ITS LCO 3.4.12 includes MODE 6 when the reactor vessel head is on. With no fuel in the reactor vessel the plant is not in MODE 6. The statement was placed in the ITS Bases to indicate that low temperature overpressure protection (LCO 3.4.12) is not required to be OPERABLE with no fuel in the reactor vessel. There may be some plant conditions when the reactor is defueled that warrant placing the reactor vessel head on the vessel for radiological concerns. In these situations, the requirements of LCO 3.4.12 are not required to be met. The inserted Bases words are being deleted and plant procedures will provide the appropriate guidance for the plant conditions when no fuel is in the reactor vessel.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.12-7 **APPLICABILITY:** CP

REQUEST ITS 3.4.12 Required Action D.1 (Comanche Peak)

Comment: Is there an approved analysis that demonstrates that this new action is sufficient protection from an accumulator discharge?

FLOG Response: The additional actions in Condition D are equivalent to depressurizing the accumulator to less than the maximum RCS pressure for the existing cold leg temperature as allowed in the PTLR; instead of depressurizing the accumulator, the cold leg temperature is increased to the point where the maximum allowable pressure is greater than the accumulator pressure. Because the maximum allowable pressure is greater than the accumulator pressure, any accumulator discharge transient has no significant adverse effects; therefore, this condition is not explicitly analyzed. The additional actions are consistent with CTS 3.4.8.3.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.12-8 **APPLICABILITY:** CP

REQUEST: ITS 3.4.12 Required Action D.2 (Comanche Peak)

Comment: What RCS temperature has to be greater than 350 degrees F? Tave (enter Mode 3)? One or more cold leg temperature(s)?

FLOG Response: The RCS temperature > 350°F refers to T_{ave} for the purpose of entering Mode 3 and thus exiting the Applicability of the specification. For CPSES, in order to exit the Applicability by satisfying Required Action D.1 (raising RCS cold leg temperature to 320°F), it is also necessary to satisfy the three additional requirements of RCP operation, pressurizer level and heatup rate. Because of these constraints and because of the close proximity of RCS cold leg temperature requirement to the Mode 3 temperature requirement, Mode 3 may achieved before all cold legs are above 320°F. New CPSES Required Action D.2 specifies this option.

Required Action D.2 is revised to refer to RCS <u>average</u> temperature.

ATTACHED PAGES:

Encl 5A 3.4-30

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Increase RCS cold leg temperature to > ~~275~~ 320°F.	12 hours
	AND Verify at least one reactor coolant pump is in operation.	
	AND Pressurizer level is ≤ 92%.	
	AND The plant heatup rate is limited to 60°F in any one hour period.	
	OR	
	D.2 Increase RCS average temperature to > 350°F	12 hours
	OR	
	D.3 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours
E. One required RCS relief valve inoperable in MODE 4.	E.1 Restore required RCS relief valve to OPERABLE status.	7 days
F. One required RCS relief valve inoperable in MODE 5 or 6.	F.1 Restore required RCS relief valve to OPERABLE status.	24 hours

B-PS

3.4-05

3.4-05
Q-3.4.12-8

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.13-2 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: **Change 6-26 LS 30 and Difference 3.4-36 (Diablo Canyon, Callaway and Wolf Creek)**

Comment: TSTF-116 has not yet been approved by the NRC.

FLOG RESPONSE: TSTF-116, Rev. 2 is currently under NRC review. This change provides assurance that the RCS water inventory balance will provide meaningful results. The proposed wording in TSTF-116, Rev. 2 was modified from TSTF-116, Rev. 1, and these modifications have been incorporated into the ITS. The FLOG continues to pursue the changes proposed by this traveler.

This comment is also applicable to CPSES based on the applicability of JFD 3.4-36.

ATTACHED PAGES:

Encl. 5A	Traveler Status page
Encl. 5B	B 3.4-76

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev.3 ~~2~~	Incorporated	3.4-33	Approved by NRC Q-3.4.2-1
TSTF-28	Incorporated	3.4-22	Approved by NRC.
TSTF-54, Rev. 1	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-60	Incorporated	3.4-15	Approved by NRC.
TSTF-61	Not incorporated	NA	Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. 2~~1~~	Incorporated	3.4-31	Approved by NRC. TR-3.4-004
TSTF-93, Rev. 3~~1~~	Incorporated	3.4-17	Approved by NRC. Q-3.4.9-3
TSTF-94, Rev.1	Not incorporated	NA	Retained current TS. TR-3.4-005
~~TSTF-105~~	~~Incorporated~~	~~3.4-38~~	Q-3.4.1-1
TSTF-108, Rev. 1	Not incorporated	NA	LCO 3.4.19 does not apply.
TSTF-113, Rev. 4 ~~3~~	Incorporated	3.4-39	Q-3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev.2 ~~1~~	Incorporated	3.4-36	Q-3.4.13-2
TSTF-136	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-137	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-138	Not incorporated	NA	Inconsistent with RCS loops requirements of ITS 3.4.5 and 3.4.6.

ACTIONS

B.1 and B.2 (continued)

must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows) ~~and near operating pressure~~. Therefore, a Note is added allowing that this SR is not required to be performed ~~in MODES 3 and 4~~ until 12 hours ~~of~~ after establishing steady state operation near operating pressure. ~~have been established.~~ The 12 hour allowance provides sufficient time to collect and process necessary data after stable plant conditions are established. Q-3.4.13-2

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful ~~and a Note requires the Surveillance to be met when steady state is established.~~ For RCS operational LEAKAGE determination by water

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.13-3 **APPLICABILITY:** CA, CP, WC

REQUEST: ITS 3.4.13 Bases LCO c. (Wolf Creek, Callaway, and Comanche Peak)

Comment: How is the addition of what does not constitute identified leakage consistent with the definition in ITS Section 1.1?

FLOG Response: The three categories in the definition of identified leakage do not include leakage outside containment. The added text in the ITS 3.4.13 LCO Bases on what does not constitute identified leakage is unnecessary and will be removed.

ATTACHED PAGES:

Encl 5B B 3.4-74

BASES

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. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

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 level monitoring equipment~~ Containment Sump Level and Flow Monitoring System 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. Q-3.4.15-4

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 ~~identified~~ unidentified 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). ~~Identified LEAKAGE does not include leakage from portions of the Chemical and Volume Control System outside of containment which can be isolated from the RCS. Leakage of this nature should be reviewed for possible impact on the Primary Coolant Sources Outside Containment Program.~~ Violation of this LCO could result in continued degradation of a component or system. Q-3.4.13-3

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the ~~SLB~~ accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.13-4 **APPLICABILITY:** CP, DC

REQUEST: ITS 3.4.13 Bases SR 3.4.13.1 (Comanche Peak and Diablo Canyon)

Comment: The Bases for SR 3.4.13.1 define steady state as Tavg changing by less than 5 degrees F/hr (Comanche Peak) and Tavg changing by less than 5 degrees/hr and stable RCS pressure etc. (Diablo Canyon). The text for Diablo Canyon then goes on to define steady state as changing less than 5 degrees/hr and for Comanche Peak ITS Bases 3.4.15 Required Action B.1.1 and B.1.2 and B.2 defines steady state in terms of stable RCS pressure and then refers back to SR 3.4.13.1. Which statement or statements define steady state?

FLOG Response: The term "steady state" is defined as "stable RCS pressure, temperature (T_{avg} changes less than 5°F per hour), power level, pressurizer and makeup levels, makeup and letdown, and RCP seal injection and return flows." The Bases are revised to consistently use this definition.

ATTACHED PAGES:

Encl. 5B B 3.4-77, B 3.4-88 and B 3.4-89

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

inventory balance, steady state is defined as stable RCS pressure, temperature (T_{avg} changing by less than 5°F/hour), power level, pressurizer and makeup ~~tank levels, makeup~~ and letdown, and RCP seal injection and return flows. An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Q-3.4.13-4

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. ~~A Note under the Frequency column states that this SR is required to be performed during steady state operation.~~ When non steady state operation precludes surveillance performance, the surveillance should be performed in a reasonable time period commensurate with the surveillance performance length, once steady state operation has been achieved, provided greater than 72 hours have elapsed since the last performance. ~~Steady state is defined as T_{avg} changing by less than 5°F/hour.~~

Q-3.4.13-4

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions. This surveillance does not tie directly to any of the leakage criteria in the LCO or of the CONDITIONS; therefore failure to meet this surveillance is considered failure to meet the integrity goals of the LCO and LCO 3.0.3 applies.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 4 and 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.
4. FSAR, Section 3.
5. NUREG-1061, Volume 3, November 1984.
6. 10 CFR 100.

BASES

ACTIONS

Actions are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the required ~~containment monitor~~ Containment Sump Level and Flow Monitoring System, the required atmospheric particulate monitor, the required atmospheric gaseous monitor or the required air cooler condensate flow rate monitor are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage. Q-3.4.15-4

A.1 and A.2

With the required ~~containment monitor~~ Containment Sump Level and Flow Monitoring System inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (as defined in the Bases of SR 3.4.13.1). ~~stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows).~~ The 12 hour allowance provides sufficient time to collect and process necessary data after stable plant conditions are established. Q-3.4.15-4 Q-3.4.13-4

Restoration of the required ~~sump monitor~~ Containment Sump Level and Flow Monitoring System to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1. Q-3.4.15-4

~~Required Action A.1 is modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the required containment sump monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.~~

B.1.1, B.1.2, and B.2~~.1, and B.2.2~~

With ~~both gaseous and~~ the particulate containment atmosphere radioactivity monitoring instrumentation channel~~s~~ inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment

BASES

atmosphere radioactivity monitors. ~~Alternatively, continued operation is allowed if the air cooler condensate flow rate monitoring system is OPERABLE, provided grab samples are taken every 24 hours.~~

The 24 hour interval provides periodic information that is adequate to detect leakage. A note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (as defined in the Bases of SR 3.4.13.1). ~~stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows).~~ The 12 hour allowance provides sufficient time to collect and process necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

Q-3.4.13-4

~~Required Action B.1 and Required Action B.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the required gaseous and particulate containment atmosphere radioactivity monitor channel is inoperable. This allowance is provided because other instrumentation is available to monitor for RCS LEAKAGE.~~

~~C.1 and C.2~~

~~With the required containment air cooler condensate flow rate monitor inoperable, alternative action is again required. Either SR 3.4.15.1 must be performed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a CHANNEL CHECK is performed every 8 hours or a water inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the containment air cooler condensate flow rate monitor to OPERABLE status.~~

~~The 24 hour interval provides periodic information that is adequate to detect RCS LEAKAGE.~~

~~D.1 and D.2~~ C.1.1, C.1.2, C.2.1 and C.2.2

With the required containment atmosphere gaseous radioactivity monitor and the required containment air cooler condensate flow rate monitor inoperable, the ~~only~~ means of detecting leakage ~~is~~

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.14-1 **APPLICABILITY:** CA, CP, WC

REQUEST: Difference 3.4-13 (Callaway, Wolf Creek and Comanche Peak)

Comment: What is the justification for restricting the testing to check valves with the addition of the term "check" in three places in SR 3.14-1 and its Bases? All PIVs at a plant may be check valves however, the addition is not consistent with the "or isolation valve" part of the first sentence of the SR Bases or with the words of required Action A of ITS 3.4.14. For Callaway and Wolf Creek simple deletion [sic] of "check" causes a problem with CTS 4.4.6.2.2.d and 4.4.5.2.2.d for Comanche Peak.

FLOG Response: CTS 4.4.5.2.2 for Comanche Peak requires surveillances be performed on each RCS PIV listed in Table 3.4-1. The valves listed in this table are not all check valves. All the valves listed are subject to the testing frequency of items SR 4.4.5.2.2.a, b, c, and e. In addition, testing of the <u>check</u> valves within 24 hours of actuation was specifically addressed in item d. This CTS surveillance does not contain a 24 hour test requirement for non-check valve PIVs. The STS equivalent of 4.4.5.2.2 for Comanche Peak is SR 3.4.14.1. However the STS SR does not appear to limit the 24 hour test requirement to check valves only. Therefore that portion of the STS surveillance was modified to be consistent with the CTS. The Bases were similarly modified.

CTS 4.4.6.2.2.d for Callaway and Wolf Creek is similar to CTS 4.4.5.2.2.d for Comanche Peak. However, CTS 4.4.6.2.2.d for Callaway and Wolf Creek does not specify check valves only (as does the Comanche Peak counterpart). Nevertheless, all the valves subject to CTS 4.4.6.2.2.d in Table 3.4-1 for Callaway and Wolf Creek are check valves, given that the CTS SR wording excludes the RHR suction isolation valves. It was decided that the STS wording should be revised consistent with the wording for Comanche Peak to reference only check valves rather than bring forward the CTS list of PIVs and the RHR suction isolation valve exclusion.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.14-2 **APPLICABILITY**: CP, DC, WC

REQUEST: Change 6-11 LS-11 (Wolf Creek, Diablo Canyon and Comanche Peak)

Comment: The change justifies isolation by a single valve within 4 hours and the use of check valves as isolations. However, the change does not justify the practice of using a second isolation valve.

FLOG Response: Change 6-11-LS-11 for Comanche Peak and Diablo Canyon is modified to include the bracketed information from NSHC LS-11. DOC 6-11-LS-11 provides justification for isolation by a single valve within 4 hours, the use of check valves as isolation valves, the use of using a second series isolation valve within 72 hours to isolate a leaking PIV. Comanche Peak and Diablo Canyon take credit for a second series isolation valve to isolate a leaking PIV.

The Enclosure 3A description for Comanche Peak and Diablo Canyon of 6-11-LS-11 (in part) provides:

> "This change in conformance with NUREG-1431 Rev. 1, allows for the flow path to be isolated by one valve within 4 hours and [by a second in series valve] within 72 hours. This change is less restrictive and is acceptable because the first valve has been surveilled as meeting the same leakage criteria as the inoperable PIV and the small probability of a failure during the 72 hour period..."

For Comanche Peak and Diablo Canyon Enclosure 4 NSHC LS-11 provides additional justification for use two series valves which has been added to DOC 06-11-LS-11 as follows:

> "The valve used to isolate the inoperable PIV will be leak tested in accordance with the surveillance requirements. With the successful completion of this leak test requirement, there is sufficient assurance that a single valve can provide adequate isolation for the following 72 hours. [The requirement to employ a second series isolation valve within 72 hours restores the two valve isolation required by the current TS.] The interval during which only single valve isolation of high-to-low pressure interface is provided, is sufficiently short so as to not involve a significant increase in the probability or consequences of an accident previously evaluated."

Wolf Creek has evaluated this issue further and determined that the design of the plant is such that the Reactor Coolant Pressure Boundary only contains two qualified pressure isolation valves in series. Therefore, the bracketed STS Required Action allowing the use of a second series isolation valve to isolate a leaking PIV is not required.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.14-3 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: ITS 3.4.14 Actions Notes 1 and 2

Comment: The adoption of the STS notes (especially #1 which is a less restrictive change) is not discussed/justified.

FLOG Response: A new DOC (6-29-LS-38) has been added to include ITS 3.4.14 Action Note 1 to the CTS markup. This note allows separate Condition entry for each pressure isolation valve (PIV) flow path made inoperable by an inoperable PIV. Also new DOC 6-30-A is added to include ITS 3.4.14 Action Note 2 which specifies entry into applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

ATTACHED PAGES:

Encl 2	3/4 4-15
Encl 3A	13 and 13a
Encl 3B	9
Encl 4	1, 72 and 73

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

(b. cont.) With RCP seal injection flow greater than the above limit, verify ≥ 100% flow equivalent to a single OPERABLE ECCS charging train is available within 4 hours and reduce the flow rate to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. 6-09-LS 6-21-LS

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least ~~two~~ one closed manual, ~~or~~ deactivated automatic ~~valves~~, or check valve## and within 72 hours by the use of a second closed manual, deactivated automatic, or check valve##; or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.*** ### 6-11-LS 6-12-M 6-30-A Q-3.4.14-3 6-29-LS Q-3.4.14-3

(New) With the RHR suction isolation valve interlock function inoperable, isolate the affected penetration by use of one closed manual or deactivated automatic valve within 4 hours. 6-22-M

SURVEILLANCE REQUIREMENTS

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

~~a. Monitoring the Reactor Coolant System Leakage Detection System required by Specification 3.4.5.1 at least once per 12 hours;~~ 6-13-LS

b. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4; 6-14-A

c. Performance of a Reactor Coolant System water inventory balance at least within 12 hours after achieving steady state operation* and at least once per 72 hours thereafter during steady state operation~~, except that no more than 96 hours shall elapse between any two successive inventory balances~~. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4; and 6-15-LS

~~d. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.~~ 6-16-LS 6-17-LG 6-12-M

~~* T_{avg} being changed by less than 5°F/hour.~~

\## Each valve used to satisfy this action must have been verified to meet surveillance requirement 4.4.5.2.2. 6-30-A Q-3.4.14-3

\### Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

*** Separate Condition entry is allowed for each PIV flow path. 6-29-LS Q-3.4.14-3

CHANGE NUMBER	NSHC	DESCRIPTION
		remaining charging flow (with some inoperability in the charging system) is greater than or equal to 100% of the assumed post-LOCA charging flow, 72 hours is allowed to restore Operability. This change is consistent with traveler WOG-84.
6-22	M	This change adds a new ACTION to isolate the affected RHR penetration within 4 hours if the RHR suction isolation valve interlock function is inoperable. The function of the RHR suction valve interlock is to protect the RHR system from an intersystem LOCA by preventing the RCS hot leg suction isolation valves from inadvertently opening when the RCS pressure exceeds the interlock setpoint. Upon failure of the interlock, the current TS permits continued operation for 72 hours for restoration of the affected subsystem. The improved TS requires action within 4 hours to isolate the affected RHR subsystem. Thus the new ACTION decreases the probability of an intersystem LOCA upon the failure of the interlock. This is a more restrictive change and the new ACTION is in LCO 3.4.14 Condition C of the improved TS.
6-23	LS-25	Not Applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
6-24	M	Not Applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
6-25	LS-26	Not Applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
6-26	LS-30	Not Applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
6-27	A	Not Applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
6-28	LG	Not Applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
6-29	LS-38	Consistent with NUREG-1431, separate Condition entry is allowed for each flow path with excessive leakage from RCS PIVs. Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. Each flow

Q-3.4.14-3

CHANGE NUMBER	NSHC	DESCRIPTION
		path is allowed separate Condition entry based upon the functional independence of the flow paths. That is, the required actions to isolate the high pressure portions of the affected flow paths, in order to protect the connecting low pressure systems are not affected by leaking PIVs in other flow paths. Q-3.4.14-3
6-30	A	ITS 3.4.14 Action Note 2 which specifies entry into applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV is added to the CTS. This is an administrative change because it only makes explicit a general requirement that is already implicit in the CTS. Q-3.4.14-3
7-01	R	This change in conformance with NUREG-1431 Rev. 1, removes the reactor coolant system chemistry specification from the Technical Specifications. The limits and required actions on reactor coolant system chemistry is being relocated to a licensee controlled document.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
6-25 LS-26	The Operational Leakage LCO has been modified to change allowed limit for RCS pressure isolation valves.	Yes	No - Leakage limit of $\leq$ .5 gpm is already part of current TS.	Yes	No - Already part of current TS per Amendment 66.
6-26 LS-30	The CTS surveillance requirement for performing an RCS water inventory balance is modified to allow deferral of the water inventory balance such that it would be performed within 12 hours after achieving steady state conditions.	Yes	No - Already part of the CPSES current TS.	Yes	Yes
6-27 A	RCS leakage detection system descriptions are revised for consistency with current TS LCO 3.3.3.1 and FSAR Sections 5.2.5.2.2 and 11.5.2.3.2.2.	No - Current systems are applicable.	No - Current systems are applicable.	Yes	Yes
6-28 LG	The current TS definition of controlled leakage is deleted. The RCP seal water return flow limit is moved to a licensee controlled document.	No - Retaining CTS.	No - Retaining CTS.	Yes - Moved to USAR	Yes - Moved to FSAR
6-29 LS-38	Separate Condition entry is allowed each flow path with excessive leakage from RCS PIVs.	Yes	Yes	Yes	Yes Q-3.4.14-3
6-30 A	ITS 3.4.14 Action Note 2 which specifies entry into applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV is added to the CTS.	Yes	Yes	Yes	Yes Q-3.4.14-3
7-01 R	This change relocates the reactor coolant system chemistry specification from the Technical Specifications to a licensee controlled document.	No. Amendment 98/97 relocated to Equipment Control Guidelines (ECG).	Yes - To be relocated TRM.	No - Amendment 89 relocated to USAR Chapter 16.	No - Amendment 103 relocated to FSAR Chapter 16.

NO SIGNIFICANT HAZARDS CONSIDERATIONS (NSHC)

CONTENTS

Q-3.4.14-3

Q-3.4.14-3

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS-38
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, separate Action entry is allowed for each flow path with excessive leakage from RCS PIVs. Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. Each flow path is allowed separate Condition entry based upon the functional independence of the flow paths. That is, the required actions to isolate the high pressure portions of the affected flow paths, in order to protect the connecting low pressure systems are not affected by leaking PIVs in other flow paths.

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?

The proposed change adds a relaxation to the LCO by allowing separate Condition entry for each PIV flowpath. Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. Each flow path is allowed separate Condition entry based upon the functional independence of the flow paths. Thus the required actions, to isolate the high pressure portions of the affected flow paths, in order to protect the

NSHC LS-38
(continued)

connecting low pressure systems are not affected by other leaking PIVs. 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?

The only accidents that are potentially associated with this proposed change, are those related to the potential for an interfacing systems LOCA causing a failure of the low pressure portion of a system outside of containment with the resulting escape of radioactive material. The Required Actions for this LCO provide for the isolation of the flow path with valves that meet the same leakage requirements as the PIVs and which must be within the RCPB or for DCPP and CPSES, the high pressure portion of the system. The protection provided for the low pressure system continues to be maintained and is independent of the actions required to protect other flowpaths that may also be affected. This change does not introduce any new overpressure accidents and the existing analyses remain valid. Thus, the proposed change does not create the possibility of a new or different kind of accident from those 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. Overpressure protection of each affected low pressure system continues to be provided by leak tested isolation valves which are independent of other flow paths. The margin of safety established by the LCO remains unchanged. Thus there is no reduction in the margin of safety from that previously established.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS-38" 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: Q3.4.15-4 **APPLICABILITY:** CP, DC

REQUEST: ITS 3.4.15.3 (Comanche Peak and Diablo Canyon)

Comment: The SR requires a Channel Calibration of the sump monitors. However, ITS LCO 3.4.15.a only requires one monitor (level and discharge flow) [Comanche Peak] or one monitor system [Diablo Canyon] to be operable. What other monitor(s) is the SR referencing?

FLOG Response: For Comanche Peak the CTS title of the subject monitor is the "Containment Sump Level and Flow Monitoring System." The ITS markup for the system title is not quite accurate and will be revised to reflect the verbatim CTS title. The revised SR 3.4.15.3 will become "Perform CHANNEL CALIBRATION of the required Containment Sump Level and Flow Monitoring System." Also the Bases will be revised to better describe the system.

For Diablo Canyon, each of the three containment sumps into which unidentified leakage can flow has its own monitor. The sump monitoring system is made up of the three individual monitors. The surveillance requirement applies to each of them.

ATTACHED PAGES:

Encl 5A 3.4-40 and 3.4-43
Encl 5B B 3.4-74, B 3.4-87, B 3.4-88 and B 3.4-90

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One ~~containment sump (level or and discharge flow) monitor~~ Containment Sump Level and Flow Monitoring System; PS Q-3.4.15-4

b. One containment atmosphere particulate radioactivity monitor ~~(gaseous or particulate)~~; and 3.4-14

c. One containment air cooler condensate flow rate monitor or one containment atmosphere radioactivity monitor (gaseous). 3.4-14

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

ACTIONS

------------NOTE------------
LCO 3.0.4 is not applicable. 3.4-15

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required ~~Containment sump monitor~~ Containment Sump Level and Flow Monitoring System inoperable.	~~------------NOTE------------ LCO 3.0.4 is not applicable. ----------------------------~~	
	A.1 --------NOTE-------- Not required until 12 hours after establishment of steady state operation. ---------------------- Perform SR 3.4.13.1.	Once per 24 hours
	AND	
	A.2 Restore ~~containment sump monitor~~ Containment Sump Level and Flow Monitoring System to OPERABLE status.	30 days

PS Q-3.4.15-4
3.4-15
3.4-36
PS Q-3.4.15-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.15.1	Perform CHANNEL CHECK of the required containment atmosphere particulate and gaseous radioactivity monitors.	12 hours	3.4-14
SR 3.4.15.2	Perform COT of the required containment atmosphere particulate and gaseous radioactivity monitors.	92 days	3.4-14
SR 3.4.15.3	Perform CHANNEL CALIBRATION of the required Containment Sump Level and Flow Monitoring System ~~containment sump monitors~~.	18 months	B PS Q-3.4.15-4 3.4-14
SR 3.4.15.4	Perform CHANNEL CALIBRATION of the required containment atmosphere particulate and gaseous radioactivity monitors.	18 months	B 3.4-14
SR 3.4.15.5	Perform CHANNEL CALIBRATION of the required containment air cooler condensate flow rate monitor.	18 months	B

BASES

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. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

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 level monitoring equipment~~ Containment Sump Level and Flow Monitoring System 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. Q-3.4.15-4

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 ~~identified~~ unidentified 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). ~~Identified LEAKAGE does not include leakage from portions of the Chemical and Volume Control System outside of containment which can be isolated from the RCS. Leakage of this nature should be reviewed for possible impact on the Primary Coolant Sources Outside Containment Program.~~ Violation of this LCO could result in continued degradation of a component or system. Q-3.4.13-3

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the ~~SLB~~ accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR (Ref. 3). Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of ~~the NRC Policy Statement.~~ 10CFR50.36(c)(2)(ii).

LCO

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the ~~containment monitor~~ Containment Sump Level and Flow Monitoring System, ~~in combination with a gaseous or~~ particulate radioactivity monitor and either a containment air cooler condensate flow rate monitor or a gaseous radioactivity monitor provide an acceptable minimum. Q-3.4.15-4

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

BASES

ACTIONS

Actions are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the required ~~containment monitor~~ Containment Sump Level and Flow Monitoring System, ~~the required atmospheric particulate monitor,~~ the required atmospheric gaseous monitor or the required air cooler condensate flow rate monitor are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage. Q-3.4.15-4

A.1 and A.2

With the required ~~containment monitor~~ Containment Sump Level and Flow Monitoring System inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (as defined in the Bases of SR 3.4.13.1). ~~stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows).~~ The 12 hour allowance provides sufficient time to collect and process necessary data after stable plant conditions are established. Q-3.4.15-4 Q-3.4.13-4

Restoration of the ~~required sump monitor~~ Containment Sump Level and Flow Monitoring System ~~to OPERABLE~~ status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1. Q-3.4.15-4

~~Required Action A.1 is modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the required containment sump monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.~~

B.1.1, B.1.2, and B.2.~~1, and B.2.2~~

With ~~both gaseous and~~ the particulate containment atmosphere radioactivity monitoring instrumentation channel~~s~~ inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment

BASES

ACTIONS

C.1.1, C.1.2, C.2.1 and C.2.2 (continued)

are the ~~containment sump monitor~~ Containment Sump Level and Flow Monitoring System ~~and the~~ containment atmosphere particulate radioactive monitor. This Condition does not provide all the required diverse means of leakage detection. With both gaseous containment atmosphere radioactivity monitoring and containment air cooler condensate flow rate monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Q-3.4.15-4

The followup Required Action is to restore either of the inoperable required monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.

A note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process necessary data after stable plant conditions are established.

~~E~~D.1 and ~~E~~D.2

If a Required Action of Condition A, B or C ~~or D~~ cannot be met, 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 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. Q-3.4.Gen-1

~~F~~E.1

With all required monitors/systems inoperable, no automatic means of monitoring leakage ~~are~~ available, and immediate plant shutdown in accordance with LCO 3.0.3 is required. Q-3.4.15-4

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.16-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: Difference 3.4-39

Comment: TSTF-113 has not yet been approved by the NRC staff.

FLOG Response: See response to Comment number 3.4.11-3

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q3.4.G-1 **APPLICABILITY:** CP

REQUEST CTS 3.4.8.2 and Change 9-05-R (Comanche Peak)

Comment: The CTS Cross Reference Table shows this specification is relocated to the FSAR. Since this is an operational requirement shouldn't it be in the PTLR or a plant procedure?

FLOG Response: The Cross Reference Table was in error. The correct location is the TRM as indicated in enclosure 3B. The TRM contains operational requirements and is the appropriate location. The Cross Reference Table will be revised accordingly.

ATTACHED PAGES:

Encl 1 4 (Sorted by Current TS)

Table 4.4-1	ITEM	4.a	3.4.16	ACTION	A.1	
Table 4.4-1	ITEM	4.a	3.4.16	CONDITION	B	
Table 4.4-1	ITEM	4.a	3.4.16	ACTION	B.1	
Table 4.4-1	ITEM	4.b	3.4.16.2	SR		
3.4.8.1	LCO		3.4.3	LCO		
3.4.8.1	LCO		Relocated	PTLR		
3.4.8.1	ACTION		3.4.3	CONDITION	A	
3.4.8.1	ACTION		3.4.3	ACTION	A.2	
3.4.8.1	ACTION		3.4.3	CONDITION	B	
3.4.8.1	ACTION	New	3.4.3	CONDITION	C	
4.4.8.1.1	SR		3.4.3.1	SR		
4.4.8.1.2	SR		Relocated	FSAR		
Figure 3.4-2			Relocated	PTLR		
Figure 3.4-3			Relocated	PTLR		
Table 4.4-2			Relocated	FSAR		
3.4.8.2	LCO		Relocated	~~FSAR~~ TRM		
3.4.8.2	ACTION		Relocated	~~FSAR~~ TRM		Q-3.4.G-1
4.4.8.2	SR		Relocated	~~FSAR~~ TRM		
3.4.8.3	LCO		3.4.12	LCO		
3.4.8.3	LCO	a	3.4.12	LCO		
3.4.8.3	LCO	a	Relocated	FSAR		
3.4.8.3	APP		3.4.12	APP		
3.4.8.3	ACTION	a	3.4.12	CONDITION	E	
3.4.8.3	ACTION	a	3.4.12	CONDITION	G	
3.4.8.3	ACTION	b	3.4.12	CONDITION	F	
3.4.8.3	ACTION	b	3.4.12	CONDITION	G	
3.4.8.3	ACTION	c	3.4.12	CONDITION	G	
3.4.8.3	ACTION	d	3.4.12.5	SR		
3.4.8.3	ACTION	e			Not Used	
3.4.8.3	ACTION	f	3.4.12	ACTION	Note	
3.4.8.3	ACTION	New	3.4.12	CONDITON	B	
3.4.8.3	ACTION	New	3.4.12	CONDTION	G	
3.4.8.3	ACTION	New	3.4.12	CONDTION	C	
3.4.8.3	ACTION	New	3.4.12	CONDITION	D	
3.4.8.3	ACTION	New	3.4.12	CONDTION	G	
3.4.8.3	ACTION	New	3.4.12	CONDITION	G	
4.4.8.3	SR	New	3.4.12.3	SR		
4.4.8.3.1	SR	a	3.4.12.8	SR		
4.4.8.3.1	SR	b	3.4.12.9	SR		
4.4.8.3.1	SR	c	3.4.12.6	SR		
4.4.8.3.2	SR	a	3.4.12.4	SR		
4.4.8.3.2	SR	b	3.4.12.4	SR		
4.4.8.3.2	SR	c	5.5.8			

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: CA 3.4-004 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: This item covers the following changes:

1. Revise ITS 3.4.14 (and corresponding CTS mark-ups) to reflect that the RHR suction isolation valves from the RCS are remote-manual, not automatic. (Not applicable to DCCP, WCGS and CPSES)

2. Define an OPERABLE RCP in ITS 3.4.4 LCO Bases as defined in ITS 3.4.5 and 3.4.6 LCO Bases.

3. Revise the ITS 3.4.9 Bases for Required Action A.3 to match the Bases changes made for ITS 3.4.5 Required Action C.2 per approved traveler TSTF-87 with regard to ways to make the Rod Control System incapable of rod withdrawal.

4. Revise the Applicability Bases for ITS 3.4.10 to reflect the change to the Note in Enclosure 5A under JFD 3.4-18, i.e., the Note allows entry into MODE 3. MODE 4 should be struck-through as it was in Enclosure 5A since the pressurizer safety LCO is not applicable in MODE 4. (Not applicable to DCCP, WCGS and CPSES)

ATTACHED PAGES:

Encl 5B B 3.4-19 and B 3.4-42

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the limit value, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops – MODES 1 and 2 satisfy Criterion 2 of ~~the NRC Policy Statement.~~ 10CFR50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required to be in operation at ~~rated~~ power.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow. CA-3.4-004

APPLICABILITY

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

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

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level" (MODE 6).

(continued)

cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters, capable of being powered from offsite power or an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable. WC-3.4-002

ACTIONS

A.1 ~~and~~ A.2, A.3 and A.4

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Water Level — High Trip.

If the pressurizer water level is not within the limit, action must be taken to ~~restore~~bring the plant to a MODE in which the LCO does not apply. ~~operation within the bounds of the safety analyses.~~ To achieve this status, within 6 hours the unit must be brought to MODE 3, with all rods fully inserted and incapable of withdrawal (e.g., de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets). Additionally, the unit must be brought ~~the reactor trip breakers open, within 6 hours and~~ to MODE 4 within 12 hours. This takes the unit out of the applicable MODES. ~~and restores the unit to operation within the bounds of the safety analyses.~~ CA-3.4-004

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.

B.1

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand for more than one group of heaters ~~caused by loss of offsite power~~ would be unlikely in this period. ~~Pressure control may be maintained during this time using normal station powered heaters.~~

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: CP 3.4-004 **APPLICABILITY:** CP

REQUEST: Revises the Bases discussion of SR 3.4.1.4 to clarify the meaning the SR note concerning " not required to be performed until...".

ATTACHED PAGES:

Encl. 5B B 3.4-6

indication for this parameter indicates in percent (%). The value in % that will assure compliance with the minimum total flow limit in the SR is determined based on the measured RCS total flow from SR 3.4.1.4. Following each refueling outage and prior to the completion of SR 3.4.1.4, the value in % used to assure compliance with the minimum RCS total flow is based upon the measured RCS total flow (SR 3.4.1.4) from the previous operating cycle or an alternate measurement and assessment of actual RCS total flow.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every ~~[18]~~ 18 months (after each refueling) allows the installed RCS flow instrumentation to be ~~calibrated~~ normalized and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The Frequency of ~~[18]~~ 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance. This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is ~~not required to be performed until~~ not required to be performed until after exceeding 85% RTP after each refueling outage. ~~24 hours after ≥ [90%] RTP required to be performed prior to exceeding 85% RTP after each refueling outage. This exception is appropriate since the heat balance requires the plant to be at a minimum of [90%] RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching [90%] RTP.~~ Using precision instrumentation with multiple indications, the stated RCS flow accuracy may be attained at power levels significantly below 85% RTP, as described in the uncertainty analyses. Requiring the precision flow measurement to be performed prior to 85% RTP allows for a single testing plateau to be used to perform the RCS flow measurement and various other tests described in Section 3.2. Procedures require that the THERMAL POWER, available instrumentation, and calibration intervals be sufficient to ensure that the stated RCS flow accuracy is attained. For feedwater pressure and temperature, the main steam pressure, and feedwater flow differential pressure instruments are calibrated within 90 days of performing the calorimetric flow measurement.

CP-3.4-004

REFERENCES 1. FSAR, Section 15.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NOs: TR-3.4-004
TR-3.4-005
TR-3.4-006
TR-3.4-009

APPLICABILITY: CA, CP, DC, WC

REQUEST: Revise the Traveler Status Sheet to reflect that TSTF-54 Rev. 1, TSTF-87 Rev. 2, TSTF-136, TSTF-137, TSTF-153, TSTF-162, and TSTF-233 (was WOG-67) are approved by NRC. Add Rev. 1 to TSTF-94 and TSTF-151.

ATTACHED PAGES:

Encl. 5A	Traveler Status pages (2)
Encl 6A	5
Encl 6B	4

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev.3 ~~2~~	Incorporated	3.4-33	Approved by NRC Q-3.4.2-1
TSTF-28	Incorporated	3.4-22	Approved by NRC.
TSTF-54, Rev. 1	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-60	Incorporated	3.4-15	Approved by NRC.
TSTF-61	Not incorporated	NA	Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. 2~~1~~	Incorporated	3.4-31	Approved by NRC. TR-3.4-004
TSTF-93, Rev. 3~~1~~	Incorporated	3.4-17	Approved by NRC. Q-3.4.9-3
TSTF-94, Rev.1	Not incorporated	NA	Retained current TS. TR-3.4-005
~~TSTF-105~~	~~Incorporated~~	~~3.4-38~~	Q-3.4.1-1
TSTF-108, Rev. 1	Not incorporated	NA	LCO 3.4.19 does not apply.
TSTF-113, Rev. 4 ~~3~~	Incorporated	3.4-39	Q-3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev.2 ~~1~~	Incorporated	3.4-36	Q-3.4.13-2
TSTF-136	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-137	Incorporated	NA	Approved by NRC TR-3.4-009
TSTF-138	Not incorporated	NA	Inconsistent with RCS loops requirements of ITS 3.4.5 and 3.4.6.

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS	
TSTF-151 Rev. 1	Incorporated	NA		TR-3.4-009
TSTF-153	Incorporated	3.4-01	Approved by NRC	TR-3.4-009
TSTF-162	Incorporated	NA	Approved by NRC.	TR-3.4-006
~~WOG-51, Rev.1~~ TSTF-285	Incorporated	3.4-45 & 3.4-~~23~~52	See also CNs 3.4-18 and 3.4-20.	Q-3.4-12-2
~~WOG-60~~ TSTF-288	Incorporated	3.4-35		Q-3.4.11-2
~~WOG-67, Rev. 1~~ TSTF-233	Incorporated	3.4-10	Approved by NRC. DCPP only.	TR-3.4-009
WOG-87 Rev. 2	Incorporated	3.4-47		Q-3.4.11-4
~~WOG-99~~ TSTF-282	Incorporated	3.4-40	Applicable to Callaway and Wolf Creek only.	Q-3.4.1-2
~~WOG-100~~TSTF-280	Incorporated	3.4-49		Q-3.4.12-1

CHANGE NUMBER	JUSTIFICATION
	gross specific activity limit is added to LCO 3.4.16 Condition B rather than its first reference being in SR 3.4.16.1. This is also consistent with the treatment of DOSE EQUIVALENT I-131.
3.4-26	Not applicable to CPSES. See Conversion Comparison Table (enclosure 6B).
3.4-27	Not applicable to CPSES. See Conversion Comparison Table (enclosure 6B).
3.4-28	Not applicable to CPSES. See Conversion Comparison Table (enclosure 6B).
3.4-29	Not applicable to CPSES. See Conversion Comparison Table (enclosure 6B).
3.4-30	An exception is added to the Actions for LCO 3.4.12 This is consistent with the current licensing basis. Additionally, circumstances could arise where increasing MODE would reduce the risk of a low temperature overpressurization event. In these cases it would be unwise to maintain the plant in a lower MODE configuration. Increasing plant MODE may also be the expedient way to exit a low temperature over pressurization potential when operating within a Condition. This option should be retained as exist in the current Technical Specifications.
3.4-31	These ACTIONS in ITS 3.4.5 and 3.4.9 are modified to reflect their LCO. The position of the reactor trip breakers and the power supply status of the CRDMs are not LCO requirements; therefore CONDITIONS and ACTIONS are revised. As worded in NUREG-1431 Rev 1, these ACTIONS could preclude certain testing in MODE 3. A more generic action, which assures the rods cannot be withdrawn, replaces the specific method of precluding rod withdrawal. The specific methods are added to the Bases as examples. The revised ACTIONS still assure rod withdrawal is precluded and this detail is not required to be in the TS to provide adequate protection to the public health and safety. No technical changes result from this change. These changes are consistent with TSTF-87 ~~Rev. 1~~. TR-3.4-004
3.4-32	In accordance with industry traveler TSTF-26, the ACTION would be changed to specify taking the plant to a MODE for which the LCO is not applicable. This change maintains the consistency between the Mode of Applicability and the Required Action which requires the Mode of Applicability to be exited.

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-27	Applicability for LCO 3.4.12 MODE 6 is revised to include an additional qualification if the head closure bolts are not fully detensioned, per DCCP CTS.	Yes	No	No	No
3.4-28	This change adds a DCPP specific description of a secured open valve.	Yes	No	No	No
3.4-29	The use of Channel Functional Test (CFT) would be retained from the current DCPP TS to the improved TS.	Yes	No	No	No
3.4-30	An LCO 3.0.4 exception is added to the Actions of LCO 3.4.12	No - This change is out of scope for DCPP.	Yes	Yes	Yes
3.4-31	Condition C and REQUIRED ACTION D.1 of ITS 3.4.5 and Condition A of ITS 3.4.9 are modified to reflect generic wording to assure that the rods are fully inserted and cannot be withdrawn. This change is consistent with TSTF-87 Rev. 1	Yes	Yes	Yes	Yes
3.4-32	In accordance with industry traveler TSTF-26, the ACTION would be changed to specify taking the plant to a MODE for which the LCO is not applicable.	Yes	Yes	Yes	Yes
3.4-33	The Frequency of SR 3.4.2.1 is changed to "12 hours". This change is based on industry traveler TSTF-27.	Yes	Yes	Yes	Yes
3.4-34	Retains CPSES current TS which requires that the precision RCS flow measurement be performed prior to exceeding 85% RTP.	No	Yes	No	No
3.4-35	Adds a note to SR 3.4.11.1 and SR 3.4.11.2 stating that the SRs are only required to be performed in Modes 1 and 2.	Yes	Yes	Yes	Yes
3.4-36	SR 3.4.13.1 and ACTIONs for LCO 3.4.15 are revised with the addition of a note per TSTF-116.	Yes	Yes	Yes	Yes
3.4-37	The primary to secondary leakage limits are revised per Callaway OL Amendment No. 116 dated October 1, 1996.	No	No	No	Yes

TR-3.4-004

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 3.4-001 **APPLICABILITY:** CP, WC

REQUEST: ITS LOC 3.4.8, Note 1.a. is revised to "at least 10°F" consistent with CTS 3.4.1.4.2. Note 1.a. is a bracketed ([ ]) note in NUREG-1431, therefore, this change is incorporating CTS inside the brackets.

ATTACHED PAGES:

Encl. 5A 3.4-17

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops – MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

---------------------------NOTES---------------------------

1. All RHR pumps may be ~~de-energized~~ removed from operation for ≤ ~~15 minutes when switching from one loop to another~~ 1 hour provided: 3.4-01 3.4-03

   a. The core outlet temperature is maintained ≥ at least 10°F below saturation temperature. B WC-3.4-001

   b. No operations are permitted that would cause a reduction of the RCS boron concentration; and

   c. No draining operations to further reduce the RCS water volume are permitted.

2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

---------------------------NOTE---------------------------

While this LCO is not met, entry into MODE 5, Loops Not Filled from MODE 5, Loops filled is not permitted. 3.4-48

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 3.4-002 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: Clarify ITS 3.4.9 Applicability Bases to state the pressurizer heaters are capable of being powered from either the offsite power source or the emergency power supply.

ATTACHED PAGES:

Encl. 5B B 3.4-42

cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters, capable of being powered from offsite power or an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable. WC-3.4-002

ACTIONS

A.1 ~~and~~ A.2, A.3 and A.4

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Water Level – High Trip.

If the pressurizer water level is not within the limit, action must be taken to ~~restore~~bring the plant to a MODE in which the LCO does not apply. ~~operation within the bounds of the safety analyses.~~ To achieve this status, within 6 hours the unit must be brought to MODE 3, with all rods fully inserted and incapable of withdrawal (e.g., de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets). Additionally, the unit must be brought ~~the reactor trip breakers open, within 6 hours and~~ to MODE 4 within 12 hours. This takes the unit out of the applicable MODES. ~~and restores the unit to operation within the bounds of the safety analyses.~~ CA-3.4-004

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.

B.1

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand for more than one group of heaters ~~caused by loss of offsite power~~ would be unlikely in this period. ~~Pressure control may be maintained during this time using normal station powered heaters.~~

JLS CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS

CTS 6.0 - ADMINISTRATIVE CONTROLS

ITS 5.0 - ADMINISTRATIVE CONTROLS

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION AND LICENSEE INITIATED ADDITIONAL CHANGES

INDEX OF ADDITIONAL INFORMATION

ADDITIONAL INFORMATION NUMBER	APPLICABILITY	ENCLOSED
5.1-1	CA	NA
5.2-1	CA, CP, DC, WC	YES
5.3-1	CA, DC, WC	NA
5.5-1	CA, DC	NA
5.5-2	CA, CP, DC, WC	YES
5.5-3	CA, CP, DC, WC	YES
5.5-4	CA, CP, DC, WC	YES
5.5-5	CA	NA
5.5-6	CA	NA
5.5-7	WC	NA
5.5-8	CA, CP, DC, WC	YES
5.5-9	CA, DC WC	NA
5.5-10	CA, WC	NA
5.5-11	DC	NA
5.5-12	WC	NA
5.5-13	DC	NA
5.5-14	CP	YES
5.6-1	CA, CP, DC, WC	YES
5.6-2	DC	NA
5.7-1	CA, CP, DC, WC	YES
CA 5.0-002	CA	NA
CA 5.0-003	CA, DC, WC	NA
CA 5.0-004	CA	NA
CA 5.0-005	CA	NA
DC 5.0-ED	DC	NA
DC 5.0-001	DC	NA
DC 5.0-002	DC	NA
DC 5.0-003	DC	NA
DC 5.0-004	DC	NA
TR 5.0-003	CA, CP, DC, WC	YES
TR 5.0-005	CP	YES
TR 5.0-006	CA, CP, DC, WC	YES
WC 5.0-ED	WC	NA
WC 5.0-001	WC	NA
WC 5.0-002	WC	NA
WC 5.0-003	WC	NA
WC 5.0-004	WC	NA
WC 5.0-005	WC	NA
WC 5.0-006	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
(continued)

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 5.2-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: STS 5.2.2 b and Difference 5.2-2

Comment: TSTF-121 has been withdrawn for modification, combination and resubmission. Use current ITS.

FLOG RESPONSE: Traveler TSTF-258 has been submitted to the NRC for review. This traveler superseded travelers, TSTF-86, TSTF-121, and TSTF-167. TSTF-258 is based on the recommendations in the April 9, 1997 letter from C. Grimes (NRC) to J. Davis (NEI), with some exceptions. The FLOG submittals have been revised to incorporate TSTF-258. The latest industry status on TSTF-258 is that the NRC has requested changes to Section 5.7, High Radiation Area. See response to Comment Number 5.7-1 for how the FLOG has addressed the NRC comments on TSTF-258.

ATTACHED PAGES:

Encl. 2	6-2, 6-3, 6-7, 6-11 and 6-15
Encl. 3A	2, 2a, 4, 4a, 6 and 7
Encl. 3B	2, 2a, 5, 5a and 8
Encl. 4	1, 19 and 20
Encl. 5A	Traveler Status sheet, 5.0-4, 5.0-5, 5.0-6, 5.0-11, 5.0-31, 5.0-39 thru 5.0-44
Encl. 6A	1, 1a, 2, 3, 4 and 5
Encl. 6B	1, 2, 4, 5 and 6

6.2.1 ONSITE AND OFFSITE ORGANIZATION (Continued)

d. The individuals who train the operating staff and those who carry out the radiation protection and quality assurance functions may report to the appropriate manager onsite; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 UNIT STAFF

The unit organization shall include:

a. An Auxiliary Operator shall be assigned to each reactor containing fuel and an additional Auxiliary Operator be assigned if either unit is operating;

With both units shutdown or defueled, a total of three Auxiliary operators for the two units are required. 1-10-M

b. ~~At least one licensed Operator for each unit shall be in the control room when fuel is in either reactor. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room~~; 1-05-A

c. A Radiation Protection Technician* and a Chemistry Technician* shall be onsite when fuel is in the reactor**; 1-11-A

d. ~~A site Fire Brigade of at least five members* shall be maintained on site at all times. The Fire Brigade shall not include the Shift Supervisor and the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency;~~ 1-08-LG

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, Radiation Protection Technicians, auxiliary operators, and key maintenance personnel).

~~The amount of overtime worked by unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).~~ The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime. Any deviation from the above guidelines shall be authorized in advance by the [Plant Manager] or the [Plant Manager's] designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized. ; and 1-09-A Q-5.2-1

f. The Shift Operations Manager shall hold a Senior Reactor Operator license.

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* The Radiation Protection and the Chemistry Technicians ~~and Fire Brigade~~ composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions. 1-08-LG

** a single Radiation Protection Technician and a single Chemistry Technician can fulfill this requirement for both units. 1-11-A

6.2.2 UNIT STAFF (Continued)

g. ~~The Shift Technical Advisor (STA)~~ An individual shall provide advisory technical support to the unit operations shift crew ~~Shift Manager~~ in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. ~~The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.~~ 1-15-A Q-5.2-1 1-12-A

NEW Shift crew compositions may be one less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. 1-13-A

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 2, 1987.

6.4 NOT USED

6.5 NOT USED

6.6 NOT USED

6.7 ~~SAFETY LIMIT VIOLATION~~ 2-02-LS

~~6.7.1 The following actions shall be taken in the event a Safety Limit is violated~~:

~~a. In accordance with 10CFR50.72, the NRC Operations Center, shall be notified by telephone as soon as practical and in all cases within one hour after the violation has been determined. The Plant Manager*, Vice President of Nuclear Operations* and the Operations Review Committee (ORC) shall be notified within 24 hours~~.

~~b. A Licensee Event Report shall be prepared in accordance with 10CFR50.73.~~

~~c. The Licensee Event Report shall be submitted to the Commission in accordance with 10CFR50.73, and to the Plant Manager*, Vice President of Nuclear Operations*, Station Operations Review Committee (SORC) and Operations Review Committee (ORC) within 30 days after discovery of the event~~.

~~* If that organizational position is assigned~~

10) Limitations on the annual ~~dose or~~ dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity ~~and to radiation~~ from uranium fuel cycle sources conforming to 40CFR190. 2-18-A Q-5.2-1

11) The provisions of Technical Specification 4.0.2 and 4.0.3 are applicable to the Radiological Effluent Controls Program. 2-22-A Q-5.2-1

f. Not used

g. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September, 1995"

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.3 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program, with the exception of the containment ventilation isolation valves, which is specified in specification 4.6.1.7.2 and 4.6.1.7.3.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

NEW Technical Specifications (TS) Bases Control Program 2-11-M

~~ANNUAL~~ RADIOACTIVE EFFLUENT RELEASE REPORT** 3-06-A

6.9.1.4 The ~~Annual~~ Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10CFR50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10CFR50.36a and Section IV.B.1 of Appendix I to 10CFR50.

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience~~, including documentation of all challenges to the pressurizer PORVs or pressurizer safety valves,~~ shall be submitted on a monthly basis ~~to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC,~~ no later than the 20th of each month following the calendar month covered by the report. 3-18-LS Q-5.2-1 3-08-A

CORE OPERATING LIMITS REPORT

6.9.1.6a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

1). Moderator temperature coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2). Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

3). Control Rod Insertion Limits for Specification 3/4.1.3.6,

4). AXIAL FLUX DIFFERENCE Limits and target band for Specification 3/4.2.1.,

5). Heat Flux Hot Channel Factor, K(Z), W(Z), F_Q^{RTP}, and the $F_Q^C(Z)$ allowances for Specification 3/4.2.2,

6). Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

7) SHUTDOWN MARGINS values in Specifications 3/4.1.1.1, 3/4.1.1.2, 3/4.1.2.2, 3/4.1.2.4, and 3/4.1.2.6.

8) Refueling Boron Concentration limits in Specification 3/4.9.1. 3-15-M

6.9.1.6b The following analytical methods used to determine the core operating limits are for Units 1 and 2, unless otherwise stated, and shall be those previously approved by the NRC in:

** A single submittal may be made for a multi-unit station. The submittal ~~should~~ shall combine ~~those~~ sections ~~that are~~ common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit. 3-06-A

6.12 HIGH RADIATION AREA

qualified in radiation protection procedures (e.g., Radiation Protection Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with dose rates equal to or less than 1000 mrem/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and individuals have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the RWP.

3-11-A Q-5.2-1

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to individuals with radiation levels greater than or equal to 1000 mrem/h at 30 ~~cm (12 in.) but less than 500 rads in one hour at one meter from the radiation source or from any surface which the radiation penetrates~~ shall be provided with locked doors or continuously guarded to prevent ~~unauthorized~~ unauthorized ~~inadvertent~~ entry, and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or radiation protection supervision. Doors shall remain locked except during periods of access by individuals under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by individuals qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

3-20-LS

~~3-19-A~~ Q-5.2-1

For isolated high radiation areas accessible to individuals with radiation levels of greater than 1000 mrem/h at 30 cm but less than 500 rads in one hour at one meter that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking or cannot be continuously guarded , and where no enclosure can be reasonably constructed around the isolated area, that isolated area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device. 3-20-LS

6.13 NOT USED

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be 3-09-LG

CHANGE NUMBER	NSHC	DESCRIPTION
01-09	A	~~Not used.~~ The CTS requirements concerning overtime being in accordance with the NRC Policy Statement is replaced by referring to administrative procedures for the control of working hours. The proposed change provides reasonable assurance that safe plant operations will not be jeopardized by impaired performance caused by excessive working hours. Specific controls for working hours of reactor plant staff are described in procedures that require a deliberate decision making process to minimize the potential for impaired personnel performance, and that established procedure control processes will provide sufficient controls for changes to that procedure. Replacement of the CTS reference to referring to administrative controls does not change the requirements associated with working hours and is therefore considered an administrative change. These changes are consistent with the NUREG-1431 as modified by TSTF-258. Q-5.2-1
01-10	M	Adds requirement for three Auxiliary Operators for the two unit sites with both units shutdown or defueled. This change is consistent with NUREG-1431. Although more restrictive, this change is consistent with current plant practice.
01-11	A	For clarity, a note is added to state that one Radiation Protection Technician and one Chemistry Technician can fulfill the staffing requirements for both units.
01-12	A	Deletes the shift technical advisor (STA) qualifications. The additional NUREG-1431 requirement that the STA shall meet the qualifications specified by "Policy Statement of Engineering Expertise on Shift" was not inserted because this requirement is redundant in that it is embedded in RG 1.8, Revision 2 (see section on Unit Staff Qualifications).
01-13	A	Adds new statement to accommodate unexpected absences of on-duty crew member. This change is consistent with NUREG-1431. This criteria was in the CTS but was subsequently moved to the Final Safety Analysis Report (FSAR). The criteria is consistent with current practice and 10 CFR 50.54(m) and, as such, reinsertion in the TS is considered administrative only.
01-14	A	Not applicable to CPSES. See conversion comparison table (enclosure 3B).

CHANGE NUMBER	NSHC	DESCRIPTION	
01-15	A	This change revises the CTS to eliminate the title of "Shift Technical Advisor (STA)." STAs are not used at all plants (the function may be fulfilled by one of the other on-shift individuals). This Section is revised so that it does not imply that the STA and the Shift Supervisor must be different individuals. Option 1 of the Commission Policy Statement on Engineering Expertise on Shift is satisfied by assigning an individual with specified educational qualifications to each operating crew as one of the SROs (preferably the Shift Supervisor) required by 10 CFR 50.54(m)(2)(i) to provide the technical expertise on shift. Eliminating the title of STA is considered an administrative change since the requirement for engineering expertise on shift is maintained. This change is consistent with NUREG-1431 as modified by TSTF-258.	Q-5.2-1
01-16	LG	A generic title has replaced the plant specific utility title for the corporate officer having responsibility for overall plant safety. The plant specific title is moved to the FSAR. This change is consistent with TSTF-65	TR-5.0-005
02-01	A	Not applicable to CPSES. See conversion comparison table (enclosure 3B).	
02-02	LS-4	CTS Section [6.7], "Safety Limit Violation," requirements to notify the NRC within 1 hour following a violation of a safety limit (SL), submit a Safety Limit Violation Report and not resume plant operation until authorized by the Commission are being deleted. These requirements are a duplication of 10 CFR 50.36(c)(1), 10 CFR 50.72 and 10 CFR 50.73.[ ] Since the plant must meet the applicable requirements contained in the regulations, sufficient regulatory controls are maintained to allow removing these duplicate regulatory requirements from the current TS. The notification requirement to management and the review committees is an after-the-fact notification and is not necessary to assure safe operation of the facility. As such, this requirement is not necessary to be included in	

CHANGE NUMBER	NSHC	DESCRIPTION
02-11	M	New program requirements, "Safety Function Determination Program" and "Bases Control Program" would be added, consistent with NUREG-1431. Although these new programs reflect current plant practice, delineating them in the ITS would be more restrictive.
02-12	LG	Not applicable to CPSES. See conversion comparison table (enclosure 3B).
02-13	LG	Revises Section 6.14 item b to move the requirement that ODCM (or similar programs and procedures) changes require review and acceptance by onsite review committees to the ODCM. The onsite review of ODCM changes is currently required per [procedures]. This change is consistent with NUREG-1431.
02-14	M	Not applicable to CPSES. See conversion comparison table (enclosure 3B).
02-15	A	Not applicable to CPSES. See conversion comparison table (enclosure 3B).
02-16	A	Not applicable to CPSES. See conversion comparison table (enclosure 3B).
02-17	LS-1	Not applicable to CPSES. See conversion comparison table (enclosure 3B).
02-18	A	~~Not applicable to CPSES. See conversion comparison table (enclosure 3B)~~ Revises the Radioactive Effluent Controls Program Dose rate limits released to areas beyond the site boundary to reflect new 10 CFR Part 20 requirements. After issuance of Generic Letter 89-01, 10 CFR 20 was updated. The NRC issued a draft Generic Letter, 93-XX, on proposed changes to STS NUREGS based on the new 10 CFR 20. The proposed changes are consistent with the draft generic letter, the April 9, 1997 letter from C. Grimes to J. Davis (with some exceptions). The proposed changes maintain the same overall level of effluent control while retaining the operational flexibility that exists with current TS under the previous 10 CFR 20. These changes are intended to eliminate possible confusion or improper implementation of the revised 10 CFR 20 requirements. The proposed changes are consistent with NUREG-1431 as modified by TSTF-258. For DCPP and CPSES, portions of TSTF-258 were adopted that were not already incorporated into the CTS based on previous license amendments. Q-5.2-1

CHANGE NUMBER	NSHC	DESCRIPTION	
02-19	LS-2	Not applicable to CPSES. See conversion comparison table (enclosure 3B).	
02-20	A	Not applicable to CPSES. See conversion comparison table (enclosure 3B).	
02-21	A	Not applicable to CPSES. See conversion comparison table (enclosure 3B).	WC-5.0-004
02-22	A	The Radioactive Effluents Controls Program is revised to include clarification statements denoting that the provisions of CTS 4.0.2 and 4.0.3 are applicable to these activities. These statements of applicability clarify the allowance for surveillance frequency extensions and allowance to perform missed surviellances. Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications and the Relocation of Details of RETS to the Offsite Dose Calculation Manual or the Process Control Program" allowed licensees to relocate the Radiological Effluent Technical Specifications and establish the Radioactive Effluents Control Program in the Administrative Controls Section of the Technical Specifications. This change effectively implements the CTS requirements that were relocated per Generic Letter 89-01. This change is considered an administrative change since the changes are in the presentation method only. This change is consistent with NUREG-1431 as modified by TSTF-258.	Q-5.2-1
03-01	A	Revises "Routine Reports" section to be consistent with NUREG-1431. The method for submitting all reports is revised to be in accordance with 10 CFR 50.4. Since this change merely makes the TS consistent with the regulations, it is considered administrative.	
03-02	A	Not applicable to CPSES. See conversion comparison table (enclosure 3B).	

CHANGE NUMBER	NSHC	DESCRIPTION
03-08	A	CTS Specifications [6.9.1.5, 6.9.1.6 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.
03-09	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 procedures and licensee controlled documents, any changes in these record retention requirements will be adequately controlled under the provisions of 10 CFR 50.59 and the applicable regulations.
03-10	LG	Not applicable to CPSES. See conversion comparison table (enclosure 3B).
03-11	A	~~The High Radiation Area is revised to be consistent with NUREG-1431 and the new Part 20 requirements. Changes are non-technical to add clarification and conform with NUREG-1431 and RG 8.38.~~ CTS 6.12, which provides high radiation area access control alternatives pursuant to 10 CFR 20.203(c)(2) has been revised as a result of the change to 10 CFR 20 and the guidance in Regulatory Guide 8.3.8. Since the plant requirements remain the same, except as identified in specific Description of Changes, the change is considered administrative. This change is consistent with NUREG-1431 as modified by TSTF-258. Q-5.2-1
03-12	LG	Not applicable to CPSES. See conversion comparison table (enclosure 3B).
03-13	M	The following report[ ] will be added to the ITS Administrative Controls section: "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" [ ].

CHANGE NUMBER	NSHC	DESCRIPTION
03-15	M	Refueling boron concentration limits will also be added to the COLR. The addition of these limits to the COLR is considered to be more restrictive.
03-16	A	Deletes one of the allowed ECCS evaluation models for CPSES Unit 2 which is no longer used. Although this change is more restrictive with respect to the TS (the model may no longer be used), there is no impact on current operations.
03-17	A	Deletes the methodology section references in the COLR. These references are adequately defined by the analytical methods themselves as approved bu the NRC and it is redundant to repeat the information in the ITS. This change is consistent with NUREG-1431.
03-18	ALS-5	~~Not applicable to CPSES. See conversion comparison table (enclosure 3B).~~ The CTS requirement to provide documentation of all challenges to the PORV's or safety valves is deleted. The reporting of pressurizer safety and relief valve failures and challenges is based on the guidance in NUREG-0694, "TMI-Related Requirements for New Operating Licensees." The guidance of NUREG-0694 states: "Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report." NRC Generic Letter 97-02, "Revised Contents of the Monthly Operating Report" requests the submittal of less information in the monthly operating report. The generic letter identifies what needs to be reported to support the NRC Performance Indicator Program, and availability and capacity statistics. The generic letter does not specifically identify the need to report challenges to the pressurizer safety and relief valves. This change is consistent with NUREG-1431 as modified by TSTF-258. Q-5.2-1
03-19	~~A~~	Not used. ~~The term "unauthorized" is changed to "inadvertent" in the High Radiation Area section. The prevention of inadvertent entry is discussed in section 1.5 of RG 8.38. This RG reflects the NRC's position regarding physical barriers for high radiation areas. Radiation areas within the limits listed shall be locked or continuously guarded to prevent inadvertent entry as discussed in RG 8.38. Furthermore, the distinction between unauthorized versus inadvertent is important based on a Notice of Violation that Callaway received on this interpretation of terms.~~ Q-5.2-1

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-07 LG	Revises Section 6.2.2a, Unit Staff Organization, to reflect the non-licensed operator staffing requirements for a single unit site. The minimum shift crew composition as described in Table 6.2-1 has been moved to a licensee controlled document.	No DCPP is a multi-unit plant	No CPSES is a multi-unit plant	Yes moved to USAR Chapter 13	Yes moved to FSAR
01-08 LG	Moves the fire brigade requirements to a licensee controlled document. These requirements can be found in BTP ASB 9.5-1 and their duplication on the ITS is not required.	No LA 75/74	Yes Move to FSAR	Yes Move to USAR	Yes Move to FSAR
01-09	~~Not used.~~ The CTS requirements concerning overtime being in accordance with the NRC Policy Statement is replaced by referring to administrative procedures for the control of working hours.	~~NA~~ Yes	~~NA~~ Yes	~~NA~~ Yes	~~NA~~ Yes Q-5.2-1
01-10 M	Adds requirement for three auxiliary operators for the two unit sites with both units shutdown or defueled.	No Already DCPP requirement	Yes	No Wolf Creek is a single unit site	No Wolf Creek is a single unit site
01-11 A	For clarity, a note is added to state that one radiation protection technician and one chemistry technician can fulfill the staffing requirements for both units.	No DCPP procedure and operational requirements differ.	Yes	No Wolf Creek is a single unit site	No Wolf Creek is a single unit site
01-12 A	Deletes the Comanche Peak STA qualifications based on use of RG 1.8, Revision 2.	No	Yes	No	No
01-13 A	Adds new statement to accommodate unexpected absences of on-duty crew member.	No already in CTS	Yes	No already in CTS	No already in CTS
01-14 A	Deletes the shift supervisors and operating supervisor from section 6.2 as required to hold a senior reactor operator license.	No DCPP procedure and operational requirements differ.	No Not in CTS	No Wolf Creek has different requirements	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-15 A	This change revises the CTS to eliminate the title of "Shift Technical Advisor (STA)."	Yes	Yes	Yes	Yes Q-5.2-1
01-16 LG	A generic title has replaced the plant specific utility title for the corporate officer having responsibility for overall plant safety and the plant specific title is moved to the FSAR.	No- Retained CTS	Yes	No- Retained CTS	No- Retained CTS TR-5.0-005

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-16 A	Change the Diesel Fuel Oil Testing Program description for sampled properties of new fuel oil from "within limits" to "analyzed" within 30 days following sampling and addition of the fuel oil to storage tanks. This wording more clearly defines that within 30 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in table 1 of ASTM D975-81 are met. This change is consistent with the Bases for SR 3.8.3.3.	No Not in CTS	No Not in CTS	Yes	Yes
02-17 LS-1	"Reactor Coolant Pump Flywheel" is being revised consistent with TSTF-237. ~~WOG-85.~~ The proposed changes provide an exception to the examination requirements in Regulatory Guide 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity."	Yes	No - see Section 3/4.4, CN 10-03-LS	No LAR submitted 12/3/96	Yes TR-5.0-006
02-18 A	Revise the Radiological Effluent Controls Program dose rate limits to reflect changes to 10 CFR Part 20. ~~Draft Generic Letter, and proposed traveler.~~	~~No~~ Yes ~~already in CTS~~	~~No~~ Yes ~~already in CTS~~	Yes	Yes Q-5.2-1
02-19 LS-2	The surveillance interval for verifying that other properties are with limits for ASTM 2D fuel oil is changed from "within 30 days" to within 31 days" after obtaining a sample.	No addressed in 3/4.8 (CN 01-60-LS24)	No addressed in 3/4.8 (CN 01-60-LS24)	Yes	Yes
02-20 A	Add the provisions of Specifications 3.0.2 and 3.0.3 are applicable to the Diesel Fuel Oil Testing program. This change is consistent with TSTF-118.	No not in CTS	No not in CTS	Yes	Yes
02-21 A	Amendment No. 106 for Wolf Creek incorporated a footnote to allow the volumetric and surface examination of the RCP "D" motor flywheel for the first 10-year ISI interval be delayed for one operating cycle. The examinations were completed during the ninth refueling outage. Since the footnote is a one time exception and has been satisfied, the footnote is no longer applicable and can be deleted.	No	No	Yes	No WC-5.0-004

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-22 A	The Radioactive Effluents Controls Program is revised to include clarification statements denoting that the provisions of CTS 4.0.2 and 4.0.3 are applicable to these activities.	Yes	Yes	Yes	Yes Q-5.2-1
03-01 A	The method for submitting all reports is revised to be in accordance with 10 CFR 50.4.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-16 A	Deletes one of the allowed ECCS evaluation models for CPSES Unit 2 which is no longer used.	No	Yes	No	No
03-17 A	Deletes the methodology section references in the COLR.	No References do not exist in DCPP CTS	Yes	Yes	Yes
03-18 ~~A~~LS-5	~~Moves the reporting requirement for documentation of all challenges to the PORVs or safety valves to the WCNOC Monthly Operating Report.~~ The CTS requirement to provide documentation of all challenges to the PORV's or safety valves is deleted.	~~No~~Yes	~~No~~Yes	Yes	~~No~~Yes Q-5.2-1
03-19 A	~~The term "unauthorized" is changed to "inadvertent" in the High Radiation Area section. The prevention of inadvertent entry is discussed in section 1.5 of RG 8.38.~~ Not used	~~Yes~~NA	~~Yes~~NA	~~Yes~~NA	~~Yes~~NA Q-5.2-1
03-20 LS-3	The use of a continuous guard is provided as an additional option for preventing inadvertent entry into high radiation areas that are accessible to individuals.	Yes	Yes	Yes	No Maintaining CTS

NO SIGNIFICANT HAZARDS CONSIDERATIONS (NSHC)
CONTENTS

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Q-5.2-1

NSHC LS-5
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS requirement to provide documentation of all challenges to the PORV's or safety valves is deleted. The reporting of pressurizer safety and relief valve failures and challenges is based on the guidance in NUREG-0694, "TMI-Related Requirements for New Operating Licensees." The guidance of NUREG-0694 states: "Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report." NRC Generic Letter 97-02, "Revised Contents of the Monthly Operating Report" requests the submittal of less information in the monthly operating report. The generic letter identifies what needs to be reported to support the NRC Performance Indicator Program, and availability and capacity statistics. The generic letter does not specifically identify the need to report challenges to the pressurizer safety and relief valves. This change is consistent with NUREG-1431 as modified by TSTF-258.

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?

The proposed change would not affect the method of operation of plant systems and involves only the deletion of reporting any challenges to the PORVs or safety valves. Reporting of challenges to the PORVs or safety valves has not impact on any accident previously evaluated.

Therefore, the proposed change would not result in a significant increase in the probability or consequences of a previously evaluated accident.

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

The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiators are introduced by the deletion of this reporting requirement. Therefore, this 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?

The accident analyses are assumed to be initiated from conditions which are consistent with the Technical Specifications Limiting Condition for Operation. The proposed change does not affect any LCO. Therefore, there is no change in the accident analyses and all relevant event acceptance criteria remain valid. Further, the proposed change has no affect on any actual or regulated failure point which is protected by an event acceptance criterion. Because there is no change in any failure point nor in any event acceptance criteria, there is no reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS-5" 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 CTS Section 6.0/ITS 5.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS	
TSTF-9, Rev. 1	Incorporated	B-PS	NRC Approved.	
TSTF-37, Rev. 1	Incorporated	5.6-2	DCPP only	
TSTF-52, Rev. 1	Incorporated	5.5-4	Incorporated draft Rev. 1 per Q3.6.1-6z	Q-3.6.1-6
TSTF-65, Rev. 1	~~Not~~ Incorporated	~~NA~~5.2-9	~~Not~~ NRC approved ~~as of traveler cut-off date~~.	TR-5.0-005
TSTF-106, Rev. 1	Not Incorporated	NA	Retain CTS	
TSTF-115	Not Incorporated	NA	Not NRC approved as of traveler cut-off date.	
TSTF-118	Incorporated	5.5-8	NRC Approved	TR-5.0-006
~~TSTF-119~~	~~Not Incorporated~~	~~NA~~	~~Retain CTS~~	TR-5.0-006
TSTF-120, Rev. 1	Not Incorporated	NA	Retain CTS	TR-5.0-006
~~TSTF-121~~	~~Incorporated~~	~~5.2-2~~		Q-5.2-1
TSTF-152	Incorporated	5.6-4	NRC Approved	TR-5.0-006
~~TSTF-167~~	~~Incorporated~~	~~5.7-2~~		Q-5.2-1
~~WOG-67, Rev.1~~ TSTF -233	Incorporated	5.6-5		TR-5.0-003
WOG-72	Incorporated	5.5-13		
~~WOG-85~~TSTF-237	Incorporated	5.5-14		Q-5.5-2
~~Proposed Traveler~~ TSTF-258	Incorporated	5.2-2, 5.2-3, 5.2-6, 5.3-2, 5.5-1, 5.5-16, 5.6-6, 5.7-1		Q-5.2-1

5.2 Organization

5.2.2 Unit Staff (continued)

de. Administrative procedures shall be developed and implemented to limit the working hours of personnel ~~unit staff~~ who perform safety related functions (e.g. ~~(~~ licensed Senior Reactor Operators (SROs), licensed Reactor Operators (ROs), ~~health physicists~~ Radiation Protection Technicians, auxiliary operators, and key maintenance personnel ~~)~~. 5.2-3 Q-5.2-1 ~~ED~~ B-PS

~~[Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:~~ B-PS

1. ~~An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;~~

2. ~~An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;~~

3. ~~A break of at least 8 hours should be allowed between work periods, including shift turnover time;~~

4. ~~Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.~~

~~Any deviation from the above guidelines shall be authorized in advance by the [Plant Superintendent] or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.~~ B-PS

(continued)

5.2.2 Unit Staff (continued)

~~Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.~~

~~OR~~

~~The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).~~ The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime. Any deviation from the above guidelines shall be authorized in advance by the [Plant Manager*] or the [Plant Manager's*] designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.

5.2-3
Q-5.2-1

~~e~~f. The Shift Operations Manager ~~or Assistant Operations Manager]~~ shall hold an SRO license. B-PS

~~f~~g. ~~The Shift Technical Advisor (STA)~~ An individual shall provide advisory technical support to the ~~Shift Supervisor (SS) Manager~~ unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. ~~The STA~~ This shall be assigned to both units when either unit is in MODE 1, 2, 3, or 4. The STA position may be filled by the shift manager or an on-shift SRO providing the individuals meet the dual role ~~In addition, the STA shall meet~~ qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. ~~PS~~

5.2-6
Q-5.2-1

* Duties may be performed by the Vice President of Nuclear Operations if that organizational position is assigned. B-PS

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

~~[Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.]~~ B-PS

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 2, 1987~~, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff]~~. 5.3-1

5.3.2 For the purposes of 10CFR55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the rquirements of TS 5.3.1, perform the functions described in 10CFR50.54(m). 5.3-2 Q-5.2-1

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1 following:~~shall be in accordance with the following: | 5.5-1 Q-5.2-1

1. For noble gases: ≤ to a dose rate of 500 mrem/yr to the total body and ~~≤ to~~ a dose rate of ≤ 3000 mrem/yr to the skin, and | 5.5-1 Q-5.2-1

2. For iodine-131, for iodine-133, for tritium, and for all radionuclides in particulate form with half lives ≥ greater than 8 days: ~~≤ to~~ a dose rate ~~of~~ ≤ 1500 mrem/yr to any organ. | 5.5-1 Q-5.2-1

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public, beyond the site boundary, from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and | 5.5-1 Q-5.2-1

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

k. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls program surveillance frequency. | 5.5-16 Q-5.2-1

(continued)

5.6 Reporting Requirements

consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in ~~the~~ a format ~~of~~ similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. ~~[The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.]~~ In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible. B-PS B-PS

5.6.3 Radioactive Effluent Release Report

-----------------------------NOTE----------------------------- B-PS
A single submittal may be made for a multiple unit station. The submittal ~~should~~ shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1. 5.6-4

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, ~~including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves,~~ shall be submitted on a monthly basis no later than the ~~15~~ 20th of each month following the calendar month covered by the report. B 5.6-6 Q-5.2-1 5.6-1

(continued)

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.7 High Radiation Area~~ B-PS

5.7-1
Q-5.2-1

~~5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr at 30 cm, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Radiation Protection Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.~~ 5.7-1 B-PS

~~Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:~~

~~a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.~~

~~b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.~~

~~c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Manager in the RWP.~~ PS B-PS

~~5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr at 30 cm but < 500 rads in one hour at one meter from the radiation source or from any surface which the radiation penetrates shall be provided with locked or continuously guarded doors to prevent unauthorized inadvertant entry and the keys shall be maintained under the administrative control of the Shift Foreman Manager on duty or health physics radiation protection supervisor. Doors shall remain locked except during access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in~~ 5.7-1 5.7-2 PS

(continued)

~~5.7 High Radiation Area~~

~~those areas. In lieu of the stay time specification of the RWP, direct or remote such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.~~ 5.7-1 Q-5.2-1

~~5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr at 30 cm but < 500 rads in one hour at one meter, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.~~ 5.7-1

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20: 5.7-1 Q-5.2-1

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation:

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously, transmits dose rate information and cumulative dose to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure with the area, or

5.7 High Area Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued) 5.7-1 Q-5.2-1

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area: who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at I Meter from the Radiation Source or from any Surface Penetrated by the Radiation:

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the [shift ~~supervisor, radiation protection~~ manager], or his or her designee.

5.7 High Area Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation: (continued) 5.7-1 Q-5.2-1

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection proced~res, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

5.7 High Area Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated bythe Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation: (continued) 5.7-1 Q-5.2-1

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area, or

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. 5.7-2 Q-5.2-1

f. Such individual areas that are within a larger area ~~that is controlled as a high radiation area~~, such as PWR containment, where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device. 5.7-4 Q-5.2-1

JUSTIFICATIONS FOR DIFFERENCES FROM NUREG-1431 - Section 5.0

This enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, Revision 1, 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. For enclosures 3A, 3B, 4, 6A and 6B, text in brackets "[ ]" indicates the information is plant 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	JUSTIFICATION
5.1-1	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.1-2	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.2-1	This change revises Section 5.2.2a to reflect the current Technical Specifications (CTS). This change clarifies the application of the unit staff provisions to both units.
5.2-2	This change deletes Section 5.2.2.b since the requirement for the presence of a reactor operator (RO) or a senior reactor operator (SRO) in the control room is adequately controlled by 10 CFR 50.54(m)(2)(iii) and 50.54(k). The ITS 5.2.2.b requirement that is being deleted will be met through compliance with these regulations and is ~~not~~ required in the TS. This is consistent with traveler TSTF-~~121~~258. Q-5.2-1
5.2-3	~~Not used.~~ ITS Section 5.2.2d (ISTS 5.2.2e) is revised from specific working hour limits to administrative procedures to control working hours. The proposed changes will provide reasonable assurance that safe plant operations will not be jeopardized by impaired performance caused by excessive working hours. Specific working hour limitations are not otherwise required to be in the technical specifications under 10 CFR 50.36(c)(5). Specific controls for working hours of reactor plant staff are described in procedures that require a deliberate decision making process to minimize the potential for impaired personnel performance, and that established procedure control processes will provide sufficient controls for changes to that procedure. These changes are consistent with the recommendation in the April 9, 1997 letter from C. Grimes to J Davis. Additionally, the ISTS statement "Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned." is being deleted. There is no guidance in Generic Letter 82-12 that discusses these additional controls. The additional requirement to have the Plant Superintendent (or his designee) review individual overtime on a monthly basis is unnecessary since sufficient administrative controls and policies exist, as well as the role of the individuals supervisors in supervising personnel prevent excessive or abuse of overtime. These changes are consistent with TSTF-258. Q-5.2-1

CHANGE NUMBER	JUSTIFICATION
5.2-4	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.2-5	Not used.
5.2-6	~~This change revises Section 5.2.2f to describe the current [licensing basis] for the Shift Technical Advisor (STA).~~ This change revises ITS Section 5.2.2.f (ISTS Section 5.2.2.g) to describe the current [TS] and to eliminate the title of "Shift Technical Advisor (STA)." STAs are not used at all plants (the function may be fulfilled by one of the other on-shift individuals). This Section is revised so that it does not imply that the STA and the Shift Supervisor must be different individuals. Option 1 of the Commission Policy Statement on Engineering Expertise on Shift is satisfied by assigning an individual with specified educational qualifications to each operating crew as one of the SROs (preferably the Shift Supervisor) required by 10 CFR 50.54(m)(2)(i) to provide the technical expertise on shift. However, the ISTS 5.2.2.g wording of, "the STA shall provide ... support to the Shift Supervisor...," is considered to be easily misinterpreted to require separate individuals. Therefore, the wording is revised so that the STA function may be provided by either a separate individual or the individual who also fulfills another role in the shift command structure. This change is consistent with TSTF-258. Q-5.2-1
5.2-7	This change revises 5.2.2c to add a note that a single Radiation Protection Technician and a single Chemistry Technician may fulfill the requirements for both units. This statement was added to clarify operational practices consistent with current licensing basis.
5.2-8	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.2-9	A generic title has replaced the plant specific utility title for the corporate officer having responsibility for overall plant safety. A reviewer's note was added describing when the use of generic titles is allowed. A statement was also added indicating that the plant specific titles for any generic titles used will be provided in the FSAR. This change is consistent with TSTF-65. DC-5.0-002 TR-5.0-005
5.3-1	This change revises Section 5.3.1 to be consistent with the current TS regarding plant staff qualifications and training.
5.3-2	New paragraph 5.3.2 is added to ensure that there is not misunderstanding when complying with 10 CFR 55.4 requirements. The Definitions in 10 CFR 55.4 state: "Actively performing the functions of an operator or senior operator means that an individual has a position on the shift crew that requires the individual to be licensed as defined in the facility's technical specifications, and that ...." Placing this paragraph in the ITS meets the 10 CFR 55.4 requirement for defining in the facility's technical specifications the function performed by licensed individuals per 10 CFR 50.54(m). Adding this paragraph is consistent with the recommendations in the April 9, 1997 letter from C. Grimes to J. Davis and TSTF-258. Q-5.2-1

CHANGE NUMBER	JUSTIFICATION
5.5-1	These changes revise Section 5.5.4, "Radioactive Effluent Controls Program," to reflect new 10 CFR Part 20 requirements and NRC letter dated 7/28/95 (Christopher I. Grimes to Owners Groups). ~~A proposed traveler is being prepared to reflect changes required for NUREG-1431 to be consistent with 10CFR Part 20.~~ After issuance of Generic Letter 89-01, 10 CFR 20 was updated. The NRC issued a draft Generic Letter, 93-XX, on proposed changes to STS NUREGS based on the new 10 CFR 20. The proposed changes are consistent with the draft generic letter, the April 9, 1997 letter from C. Grimes to J. Davis (with some exceptions) and traveler TSTF-258. The proposed changes maintain the same overall level of effluent control while retaining the operational flexibility that exists with current TS under the previous 10 CFR 20. These changes are intended to eliminate possible confusion or improper implementation of the revised 10 CFR 20 requirements. Q-5.2-1
5.5-2	This change revises Section 5.5.3," Post Accident Sampling," to ensure capability to obtain and analyze radioactive "iodines" in lieu of "gases." This change is consistent with the current TS and plant practices.
5.5-3	This change revises Section 5.5.8, "Inservice Testing Program," to delete "including applicable supports." This change is consistent with the current TS.
5.5-4	The Containment Leakage Rate Testing Program is added to the improved Technical Specifications (ITS) consistent with the current TS. The Containment Leakage Rate Testing Program is consistent with traveler TSTF-52.
5.5-5	This change revises Section 5.5.13, "Diesel Fuel Oil Testing Program," to be consistent with the current TS. The details of the method applied to this test are discussed in the associated SR 3.8.3.3 Bases. [ ]
5.5-6	Additional programs are added to the ITS (other than Containment Leakage Rate Testing Program discussed in CN 5.5-4). [This change adds new Section 5.5.17, "Technical Requirements Manual," to the CPSES TS. A program description has been added to describe the licensee control of this licensing basis document and any changes to it. This program is subject to control by 10CFR50.59.]
5.5-7	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.5-8	A sentence is added to Section 5.5.9 ("The provisions of SR 3.0.2 are applicable to the Steam Generator Tube Surveillance Program test frequencies") and Section 5.5.13 ("The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies") to provide consistency with current application of these requirements. This is consistent with the use of current TS and alleviates potential confusion in the program descriptions. This change is consistent with traveler TSTF-118.

CHANGE NUMBER	JUSTIFICATION
5.5-9	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.5-10	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.5-11	The documents referenced for the testing frequency for the VFTP do not provide frequencies for combined pressure drop tests or the heater power rating test. The CTS frequency is added for these tests.
5.5-12	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.5-13	This change revises Radioactive Effluent Controls Program dose projections to meet original intent of TS prior to implementation of GL 89-01. GL 89-01 provided the wording for the STS (Section 5.5.4.e) which combined the requirements for cumulative and projected dose. This requires a plant to make projected doses for the quarter and year on a 31 day basis. It is only necessary and reasonable to make a projection for the next 31 days. A cumulative dose projection is still required for the current calender quarter and year in accordance with the ODCM. This is consistent with WOG-72.
5.5-14	Section 5.5.7 is being revised consistent with TSTF-237 ~~WOG-85~~ [ ]. Q-5.5-2 The proposed changes to Section 5.5.7 provide an exception to the examination requirements in Regulatory Guide 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity." The proposed exception to the recommendations of Regulatory Position C.4.b would allow for an acceptable inspection method of either an ultrasonic volumetric or surface examination. The acceptable inspection method would be conducted at approximately 10 year intervals. This change is consistent with the NRC Safety Evaluation Report associated with WCAP-14535, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination."
5.5-15	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.5-16	The Radioactive Effluents Controls Program is revised to include clarification statements denoting that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to these activities. These statements of applicability clarify the allowance for surveillance frequency extensions and allowance to perform missed surviellances. Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications and the Relocation of Details of RETS to the Offsite Dose Calculation Manual or the Process Control Program" allowed licensees to relocate the Radiological Effluent Technical Specifications and establish the Radioactive Effluents Control Program in the Administrative Controls Section of the Technical Specifications. This change effectively implements the CTS requirements that were relocated per Generic Letter 89-01. Since this change adopts previous CTS requirements, it is considered a change of presentation method only. This change is consistent with TSTF-258. Q-5.2-1
5.6-1	This change revises Section 5.6.4, "Monthly Operating Report," to reflect a revised submittal date. This change is consistent with the current TS.
5.6-2	Not applicable to CPSES. See conversion comparison table (enclosure 6B).

CHANGE NUMBER	JUSTIFICATION
5.6-3	This change revises the report date in Section 5.6.2 "Annual Radiological Environmental Operating Report" to be consistent with current TS.
5.6-4	This change revises Sections 5.6.1 and 5.6.3, "Occupational Radiation Exposure Report" and "Radioactive Effluent Release Report", respectively, per NRC letter dated July 28, 1995, "Changes to Technical Specifications Resulting from 10CFR20 and 50.36a Changes" (From Christopher I. Grimes to Owners Group Chairs). This change is consitent with TSTF-152.
5.6-5	[ ] PORV lift settings are referenced in PTLR section per ~~WOG-67, Rev. 1~~ TSTF-233. TR.5.0-003
5.6-6	The ITS requirement to provide documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves is deleted. The reporting of pressurizer safety and relief valve failures and challenges is based on the guidance in NUREG-0694, "TMI-Related Requirements for New Operating Licensees." The guidance of NUREG-0694 states: "Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report." NRC Generic Letter 97-02, "Revised Contents of the Monthly Operating Report" requests the submittal of less information in the monthly operating report. The generic letter identifies what needs to be reported to support the NRC Performance Indicator Program, and availability and capacity statistics. The generic letter does not specifically identify the need to report challenges to the pressurizer safety and relief valves. This change is consistent with TSTF-258. Q-5.2-1
5.7-1	~~This change revises High Radiation Area to incorporate changes consistent with [the current TS]. Specifically, distances from the radiation source.~~ Section 5.7 is revised in accordance with 10 CFR 20.1601(c) and updates the acceptable alternate controls to those given in 10 CFR 20.1601. These changes are consistent with the draft Generic Letter (93-XX) on proposed changes to STS NUREGs based on the new 10 CFR 20 and the letter from C. Grimes, NRC, to J. Davis, NEI dated April 9, 1997. This change is consistent with TSTF-258 and encompasses the NRC comments on 6/11/98. Additional technical changes made to Section 5.7 are identified and justified. Q-5.2-1
5.7-2	~~This change revises "unauthorized" to "inadvertent" in the High Radiation Area section to reflect NRC's position as stated in RG 8.38, Section 1.5 regarding physical barriers for High Radiation Areas. This is consistent with traveler TSTF-167.~~ ITS 5.7.2.e is revised consistent with CTS 6.12 that allows any individual or group of individuals to enter a high-high radiation area (dose rates greater than 1.0 rem/hour at 30 cm) accompanied by an individual qualified in radiation protection procedures with a radiation dose rate monitoring device. The qualified individual is responsible for providing positive control and shall perform periodic radiation surveillances at the frequency specified in the RWP. The CTS requirements allow the qualified individual to enter a locked high radiation area with plant workers without first having to enter the area to Q-5.2-1

CHANGE NUMBER	JUSTIFICATION	
	determine dose rates and then exit the area to provide dose rate information to the plant workers and then reenter the area. This flexibility is in keeping with the "As Low As Reasonably Achievable" principle while maintaing appropriate radiation worker practices.	Q-5.2-1
5.7-3	Not applicable to CPSES. See conversion comparison table (enclosure 6B).	
5.7-4	ITS 5.7.2.f is revised consistent with CTS 6.12 to delete the phrase "that is controlled as a high radiation area". The proposed change would preclude having to post an area around the high-high radiation area as a high radiation area when the area may not meet the definition of a high radiation area.	Q-5.2-1

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 5.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.1-1	Revises Section 5.1.1 to maintain WCNOC current technical specifications (CTS). The Plant Manager does not currently approve prior to implementation each proposed test, experiment or modification to systems or equipment that affect nuclear safety.	No	No	Yes	No
5.1-2	Revises Section 5.1.1 to maintain Callaway CTS that the plant manager approves prior to implementation each proposed test, experiment or modification to systems or equipment that affect nuclear safety and are not addressed in the FSAR or TS.	No	No	No	Yes
5.2-1	Revises Section 5.2.2.a to reflect the CTS. This change clarifies the application of the unit staff provisions to both units.	Yes	Yes	No Wolf Creek is a single unit site	No Callaway is a single unit site
5.2-2	The requirement for the presence of a RO or a SRO in the control room may be deleted from the ITS since this requirement is adequately controlled by 10 CFR 50.54(m)(2)(iii). This change is consistent with traveler TSTF ~~121~~ 258.	Yes	Yes	Yes	Yes Q-5.2-1
5.2-3	~~Not Used~~ ITS Section 5.2.2d (ISTS 5.2.2e) is revised from specific working hour limits to administrative procedures to control working hours.	~~NA~~Yes	~~NA~~Yes	~~NA~~Yes	~~NA~~Yes Q-5.2-1
5.2-4	Section 5.2.2.a describes the unit staff requirements for non-licensed operator staffing for multi-unit sites. This change reflects plant specific requirements for a single unit site and is consistent with the current TS.	No DCPP is a multi-unit plant	No CPSES is a multi-unit plant	Yes	Yes
5.2-5	Not Used	NA	NA	NA	NA

TECH SPEC CHANGE		APPLICABILITY				
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY	
5.2-6	Revises Section ~~5.2.2f to~~ describe the current [licensing basis] ~~for STA.~~ and eliminate the title of Shift Technical advisor (STA).	Yes	Yes LA 50/36 moved text to FSAR Section 13.1 which permits on-shift SRO to fill STA position	Yes	Yes	Q-5.2-1
5.2-7	Revises 5.2.2c to add note that a single Radiation Protection Technician and a single Chemistry Technician may fulfill the requirments for both units.	No Not current procedure or operational requirement	Yes	No Wolf Creek is a single unit site	No Callaway is a single unit site	
5.2-8	Revises Sections 5.2.2 and 5.3.1 to reflect License Amendment 128/126 dated 6/11/98 which changed requirements for the DCPP Operations Director.	Yes LA128/126	No	No	No	DC-5.0-002
5.2-9	A generic title has replaced the plant specific utility title for the corporate officer having responsibility for overall plant safety.	No	Yes	No	No	TR-5.0-005
5.3-1	Revises Section 5.3.1 to be consistent with current TS regarding plant staff qualifications and training.	Yes LA 43/42	Yes	Yes	Yes	
5.3-2	New paragraph 5.3.2 is added to ensure that there is not misunderstanding when complying with 10 CFR 55.4 requirements.	Yes	Yes	Yes	Yes	Q-5.2-1
5.5-1	Revises Section 5.5.4, "Radioactive Effluent Controls Program," ~~to reflect new 10 CFR Part 20 requirements. and NRC letter dated 7/28/95 consistent with proposed traveler.~~	Yes	Yes	Yes	Yes	Q-5.2-1
5.5-2	Revises Section 5.5.3, "Post Accident Sampling," to ensure the capability to obtain and analyze radioactive "iodines" in lieu of "gases." This change is consistent with the current TS and plant practices.	Yes	Yes	Yes	Yes	
5.5-3	Revises Section 5.5.8, "Inservice Testing Program," to delete "including applicable supports." This change is consistent with the current TS.	Yes	Yes	Yes	Yes	

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 5.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-11	The documents referenced for the testing frequency for the Ventilation Filter Testing Program (VFTP) do not provide frequencies for combined pressure drop tests or the heater power rating test. The CTS frequency is added for these two tests.	No See CN 5.5-12	Yes	Yes	Yes
5.5-12	The referenced frequencies for the tests listed in the Ventilation Filter Testing Program (VFTP) were evaluated as part of the DCPP 24 month fuel cycle program (see LAR 96-09).	Yes	No	No	No
5.5-13	Revises Radioactive Effluent Controls Program dose projections to meet original intent of TS prior to implementation of GL 89-01. (WOG-72)	Yes	Yes	Yes	Yes
5.5-14	Section 5.5.7 is being revised consistent with TSTF-237 ~~a WOG-85~~. [ ] The proposed changes to Section 5.5.7 provide an exception to the examination requirements in Regulatory Guide 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity."	Yes	Yes	Yes	Yes Q-5.5-2
5.5-15	This change provides a time interval of within 31 days after removal in which a laboratory test of a sample obtained from the charcoal adsorber must be tested. This change is consistent with the Callaway CTS.	No	No	No	Yes
5.5-16	The Radioactive Effluents Controls Program is revised to include clarification statements denoting that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to these activities.	Yes	Yes	Yes	Yes Q-5.2 1
5.6-1	Revises Section 5.6.4, "Monthly Operating Report," to reflect a revised submittal date.	No DCPP CLB consistent with NUREG-1431	Yes LAR 94-14	No Wolf Creek CTS consistent with NUREG-1431	No Callaway CTS consistent with NUREG-1431
5.6-2	Deletes the EDG Report to reflect the recommendations of GL 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," dated May 31, 1994.	Yes	No - not in CTS	No - not in CTS	No - not in CTS

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.6-3	Revises report date in Section 5.6.2, "Annual Radiological Environmental Operating Report" to be consistent with current TS.	Yes Consistent with original TS and LA 78/77.	Yes See LA 42/28	Yes	Yes
5.6-4	Revises Sections 5.6.1 and 5.6.3, "Occupational Radiation Exposure Report" and "Radioactive Effluent Release Report," respectively, per NRC letter dated July 28, 1995, "Changes to Technical Specifications Resulting from 10 CFR 20 and 50.36a Changes" (From Christopher I. Grimes to Owners Groups Chairs). This change is consistent with TSTF-152.	Yes	Yes	Yes	Yes
5.6-5	[ ] PORV lift settings are referenced in PTLR section per TSTF-233 ~~WOG-67, Rev. 1~~.	Yes	Yes	Yes	Yes TR-5.0-003
5.6-6	The ITS requirement to provide documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves is deleted.	Yes	Yes	Yes	Yes Q-5.2-1
5.7-1	~~Revises High Radiation Area to incorporate consistent changes with [10 CFR 20.1601].~~ Section 5.7 is revised in accordance with 10 CFR 20.1601(c) and updates the acceptable alternate controls to those given in 10 CFR 20.1601.	Yes	Yes	Yes	Yes Q-5.2-1
5.7-2	~~Changes "unauthorized" to "inadvertent" in the High Radiation Area section to reflect the NRC's position as stated in RG 8.38, Section 1.5 regarding physical barriers for High Radiation Areas. This change is consistent with TSTF-167.~~ ITS 5.7.2.e is revised consistent with CTS 6.12 that allows any individual or group of individuals to enter a high-high radiation area (dose rates greater than 1.0 rem/hour at 30 cm) accompanied by an individual qualified in radiation protection procedures with a radiation dose rate monitoring device.	Yes	Yes	Yes	Yes Q-5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.7-4	ITS 5.7.2.f is revised consistent with CTS 6.12 to delete the phrase "that is controlled as a high radiation area". The proposed change would preclude having to post an area around the high-high radiation area as a high radiation area when the area may not meet the definition of a high radiation area.	Yes	Yes	Yes	Yes

Q-5.2-1

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 5.5-2 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: Difference 5.5-14

Comment: WOG-85 has not yet become a TSTF. Use current ITS.

FLOG RESPONSE: WOG-85 has been approved by the TSTF and is designated as TSTF-237. This traveler has been submitted to the NRC and is under review. The proposed wording in TSTF-237 was modified from WOG-85 and these modifications have been incorporated into the ITS. The FLOG continues to pursue the changes proposed by this traveler.

For Wolf Creek, this change was approved by the NRC in Amendment No. 106 dated June 24, 1997. Therefore, the wording in ITS 5.5.7 is consistent with Amendment No. 106.

ATTACHED PAGES:

Attachment 10 CTS 3/4.4

Encl 2	3/4.4-33
Encl 3A	18
Encl 3B	13
Encl 4	70

Attachment 18 ITS 5.0

Encl 5A	Traveler Status sheet and 5.0-12
Encl 6A	3
Encl 6B	4

~~3/4.4.9 STRUCTURAL INTEGRITY~~

10-01-LS

LIMITING CONDITION FOR OPERATION

~~3.4.9 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.9.~~

~~APPLICABILITY: All MODES.~~

~~ACTION:~~

~~a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.~~

~~b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.~~

~~c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.~~

SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius ~~represented by the flywheel gauge holes shall be conducted~~ or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may ~~shall~~ be conducted at approximately ten year intervals coinciding with Inservice Inspection schedule required by ASME Section XI. 10-02-A 10-03-LS Q-5.5-2

~~NOTE: Separate Technical Specifications provide specific temperature/pressure limitations for specific components (e.g., 3/4.4.8.1 for the Reactor Coolant System, 3/4.7.2 for the Steam Generators, 3/4.7.13 for the Main Feedwater Isolation Valves, etc.)~~

CHANGE NUMBER	NSHC	DESCRIPTION
		described above and the current individual Technical Specifications which contain the operability requirements for the required components or equipment that meet criterion 3, the RCS Structural Integrity specification is deleted instead of relocated.
10-02	A	Consistent with NUREG-1431, the Reactor Coolant Pump flywheel inspection requirement has been moved to Section ITS 5.5.7.
10-03	LS-37	Consistent with traveler TSTF-237 WOG-85, the Reactor Coolant Pump Flywheel Inspection Program is revised to provide an exception to the examination requirements in Regulatory Guide 1.14, Rev 1. The exception (to Regulatory Position C.4.b(1) and C.4.b(2)) allows for an acceptable inspection method of either an ultrasonic volumetric, or surface examination. The inspection would conducted at approximately ten year intervals coinciding with the Inservice Inspection schedule required by ASME Section XI. The acceptability of the proposed change is established in WCAP-14535, Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination, with Limitations. The NRC's Safety Evaluation of the topical concluded that the inspections should not be completely eliminated but should be conducted during scheduled inservice inspections or RCP maintenance at approximately 10 year intervals. The proposed change is consistent with these recommendations. Q-5.5-2
11-01	R	This change in conformance with NUREG-1431 Rev. 1, removes the reactor coolant system vents specification from the Technical Specifications. The requirements for the reactor coolant system vents will be relocated to a licensee controlled document as identified in the Conversion Comparison Table (enclosure 3B).

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
10-02 A	The Reactor Coolant Pump flywheel inspection requirement has been moved to Section 5.5.7 in the improved TS.	No - Amendment 98/97 relocated RCP flywheel surveillances to CTS 6.8.4.i.	Yes	No - Amendment 89 relocated to USAR Chapter 16 and CTS 6.8.5b..	No - Amendment 103 relocated to FSAR Chapter 16 and CTS 6.8.5.b.
10-03 LS-37	The Reactor Coolant Pump Flywheel Inspection Program is revised to provide an exception to the examination requirements in Regulatory Guide 1.14, Rev 1. The exception (to Regulatory Position C.4.b(1) and C.4.b(2)) allows for an acceptable inspection method of either an ultrasonic volumetric, or surface examination. The inspection would conducted at approximately ten year intervals coinciding with the Inservice Inspection schedule required by ASME Section XI.	No - See CN-02-17-LS-1 in the ITS Section 5.0 package.	Yes	No - See CN-02-17-LS-1 in the ITS Section 5.0 package.	No - See CN-02-17-LS-1 in the ITS Section 5.0 package. Q-5.5-2
11-01 R	Removes the reactor coolant system vents specification from the Technical Specifications. The requirements for the reactor coolant system vents will be relocated to a licensee controlled document.	No. Amendment 98/97 relocated requirement to Equipment Control Guidelines (ECG).	Yes - To be relocated to the TRM.	No - Amendment 89 relocated to USAR Chapter 16.	No - Amendment 103 relocated to FSAR Chapter 16.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS-37
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with traveler TSTF-237-WOG-85, the Reactor Coolant Pump Flywheel Inspection Program is revised to provide an exception to the examination requirements in Regulatory Guide 1.14, Rev 1. The exception (to Regulatory Position C.4.b(1) and C.4.b(2)) allows for an acceptable inspection method of either an ultrasonic volumetric, or surface examination. The inspection would conducted at ten year intervals coinciding with the Inservice Inspection schedule required by ASME Section XI. The acceptability of the proposed change is established in WCAP-14535, Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination, with Limitations. The NRC's Safety Evaluation of the topical concluded that the inspections should not be completely eliminated but should be conducted during scheduled inservice inspections or RCP maintenance at approximately 10 year intervals. The proposed change is consistent with these recommendations. Q-5.5-2

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?

The safety function of the RCP flywheels is to provide a coastdown period during which the RCPs would continue to provide reactor coolant flow to the reactor after loss of power to the RCPs. The maximum loading on the RCP

Industry Travelers Applicable to CTS Section 6.0/ITS 5.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS	
TSTF-9, Rev. 1	Incorporated	B-PS	NRC Approved.	
TSTF-37, Rev. 1	Incorporated	5.6-2	DCPP only	
TSTF-52, Rev. 1	Incorporated	5.5-4	Incorporated draft Rev. 1 per Q3.6.1-6z	Q-3.6.1-6
TSTF-65, Rev. 1	~~Not~~ Incorporated	~~NA~~5.2-9	~~Not~~ NRC approved ~~as of traveler cut-off date~~.	TR-5.0-005
TSTF-106, Rev. 1	Not Incorporated	NA	Retain CTS	
TSTF-115	Not Incorporated	NA	Not NRC approved as of traveler cut-off date.	
TSTF-118	Incorporated	5.5-8	NRC Approved	TR-5.0-006
~~TSTF-119~~	~~Not Incorporated~~	~~NA~~	~~Retain CTS~~	TR-5.0-006
TSTF-120, Rev. 1	Not Incorporated	NA	Retain CTS	TR-5.0-006
~~TSTF-121~~	~~Incorporated~~	~~5.2-2~~		Q-5.2-1
TSTF-152	Incorporated	5.6-4	NRC Approved	TR-5.0-006
~~TSTF-167~~	~~Incorporated~~	~~5.7-2~~		Q-5.2-1
WOG-67, Rev. 1	Incorporated	5.6-5		
WOG-72	Incorporated	5.5-13		
~~WOG-85~~TSTF-237	Incorporated	5.5-14		Q-5.5-2
~~Proposed Traveler~~ TSTF-258	Incorporated	5.2-2, 5.2-3, 5.2-6, 5.3-2, 5.5-1, 5.5-16, 5.6-6, 5.7-1		Q-5.2-1

5.5 Programs and Manuals

5.5.6 ~~Pre-Stressed Concrete Containment Tendon Surveillance Program~~ Not used B-PS

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.~~

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory. Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius ~~represented by the flywheel gauge holes shall be conducted~~ or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may ~~shall~~ be conducted at approximately 10 ~~ten~~ year intervals coinciding with Inservice Inspection schedule as required by ASME Section XI. 5.5-14 Q-5.5-2

~~Reviewer's Note~~

~~1. Licensees shall confirm that flywheels are made of SA 533B material. Further, licensees having Group 15 flywheels (as determined in WCAP-14535, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination") need to demonstrate that material properties of their A516 material is equivalent to SA 533 B material, and its reference temperature,~~ ~~RT_{NDT}~~ ~~is less tha 30°F.~~

~~2. For flywheels not made of SA 533 B or A516 material, licensees need to demonstrate that the flywheel material properties are bounded by those of SA 533 B material, or provide the minimum ultimate tensile stress, the fracture toughness, and the reference temperature,~~ ~~RT_{NDT}~~, ~~for that material. For the latter, the licensee should employ these material properties, and use the methodology in the topical report, as extended in the two response to the staff's RAI, to provide an assessment to justify a change in inspection schedule for their plants.~~ 5.5-14

(continued)

CHANGE NUMBER	JUSTIFICATION
5.5-9	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.5-10	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.5-11	The documents referenced for the testing frequency for the VFTP do not provide frequencies for combined pressure drop tests or the heater power rating test. The CTS frequency is added for these tests.
5.5-12	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.5-13	This change revises Radioactive Effluent Controls Program dose projections to meet original intent of TS prior to implementation of GL 89-01. GL 89-01 provided the wording for the STS (Section 5.5.4.e) which combined the requirements for cumulative and projected dose. This requires a plant to make projected doses for the quarter and year on a 31 day basis. It is only necessary and reasonable to make a projection for the next 31 days. A cumulative dose projection is still required for the current calender quarter and year in accordance with the ODCM. This is consistent with WOG-72.
5.5-14	Section 5.5.7 is being revised consistent with TSTF-237 ~~WOG-85~~ [ ]. Q-5.5-2 The proposed changes to Section 5.5.7 provide an exception to the examination requirements in Regulatory Guide 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity." The proposed exception to the recommendations of Regulatory Position C.4.b would allow for an acceptable inspection method of either an ultrasonic volumetric or surface examination. The acceptable inspection method would be conducted at approximately 10 year intervals. This change is consistent with the NRC Safety Evaluation Report associated with WCAP-14535, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination."
5.5-15	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.5-16	The Radioactive Effluents Controls Program is revised to include Q-5.2-1 clarification statements denoting that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to these activities. These statements of applicability clarify the allowance for surveillance frequency extensions and allowance to perform missed surviellances. Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications and the Relocation of Details of RETS to the Offsite Dose Calculation Manual or the Process Control Program" allowed licensees to relocate the Radiological Effluent Technical Specifications and establish the Radioactive Effluents Control Program in the Administrative Controls Section of the Technical Specifications. This change effectively implements the CTS requirements that were relocated per Generic Letter 89-01. Since this change adopts previous CTS requirements, it is considered a change of presentation method only. This change is consistent with TSTF-258.
5.6-1	This change revises Section 5.6.4, "Monthly Operating Report," to reflect a revised submittal date. This change is consistent with the current TS.
5.6-2	Not applicable to CPSES. See conversion comparison table (enclosure 6B).

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-11	The documents referenced for the testing frequency for the Ventilation Filter Testing Program (VFTP) do not provide frequencies for combined pressure drop tests or the heater power rating test. The CTS frequency is added for these two tests.	No See CN 5.5-12	Yes	Yes	Yes
5.5-12	The referenced frequencies for the tests listed in the Ventilation Filter Testing Program (VFTP) were evaluated as part of the DCPP 24 month fuel cycle program (see LAR 96-09).	Yes	No	No	No
5.5-13	Revises Radioactive Effluent Controls Program dose projections to meet original intent of TS prior to implementation of GL 89-01. (WOG-72)	Yes	Yes	Yes	Yes
5.5-14	Section 5.5.7 is being revised consistent with TSTF-23 & WOG-85. [ ] The proposed changes to Section 5.5.7 provide an exception to the examination requirements in Regulatory Guide 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity."	Yes	Yes	Yes	Yes Q-5.5-2
5.5-15	This change provides a time interval of within 31 days after removal in which a laboratory test of a sample obtained from the charcoal adsorber must be tested. This change is consistent with the Callaway CTS.	No	No	No	Yes
5-5-16	The Radioactive Effluents Controls Program is revised to include clarification statements denoting that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to these activities.	Yes	Yes	Yes	Yes Q-5.2-1
5.6-1	Revises Section 5.6.4, "Monthly Operating Report," to reflect a revised submittal date.	No DCPP CLB consistent with NUREG-1431	Yes LAR 94-14	No Wolf Creek CTS consistent with NUREG-1431	No Callaway CTS consistent with NUREG-1431
5.6-2	Deletes the EDG Report to reflect the recommendations of GL 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," dated May 31, 1994.	Yes	No - not in CTS	No - not in CTS	No - not in CTS

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 5.5-3 **APPLICABILITY:** CA, CP, DC, CA

REQUEST: ITS 5.5.4 b&g and Difference 5.5-1

Comment: Changes are based on a yet unnumbered traveler. Use current ITS.

FLOG RESPONSE: Traveler TSTF-258 has been submitted to the NRC for review. This traveler superseded travelers, TSTF-86, TSTF-121, and TSTF-167. TSTF-258 is based on the recommendations in the April 9, 1997 letter from C. Grimes (NRC) to J. Davis (NEI), with some exceptions. The FLOG submittals have been revised to incorporate TSTF-258. The latest industry status on TSTF-258 is that the NRC has requested changes to Section 5.7, High Radiation Area. See response to Comment Number 5.7-1 for how the FLOG has addressed the NRC comments on TSTF-258.

ATTACHED PAGES:

See markups associated with Comment Number Q 5.2-1.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 5.5-4 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: ITS 5.5.4 e and Difference 5.5-13

Comment: WOG-72 has not yet become a TSTF. Use current ITS.

FLOG RESPONSE: This change to ITS 5.5.4 e was prepared in accordance with WOG-72, Rev. 1 which is currently under TSTF review. The change specifies that the requirement to determine cumulative dose contributions from radioactive effluents need be done on a current quarterly and annual basis instead of every 31 days. We believe there is a strong technical basis for this change to the ITS. We request that the NRC keep this as an open item under the assumption that the traveler will be approved prior to issuance of the SER.

ATTACHED PAGES:

None.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 5.5-8 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: CTS 3.7.6 (3.7.5.1 and 3.7.6.1 - DCPP and 3.7.7.1 and 3.7.8 -CPSES) and Change 10-08-A

Comment: It should be specifically noted as to which CTS requirements were carried over to the VFTP and which were deleted (as well as which section of what standard justified the duplication deletions). Provide explanation and justification.

FLOG RESPONSE: Attached Table 5.5-8 describes where the CTS SRs for plant ventilation systems were moved to in the ITS. The following provides justification and clarification for those CTS SRs that were not moved to either the "Ventilation Filter Testing Program (VFTP)" in the ITS, or the ITS SRs:

DOC 10-07-LG (Not applicable to CPSES) moves the requirement to verify Control Room temperature once every 12 hours to a licensee controlled document. This DOC has been revised to include the following additional justification: "The NRC has previously approved moving this type of detailed information or specific requirements to a licensee controlled document that is maintained in accordance with applicable regulatory requirements. This temperature is not an initial condition or controlled parameter for any licensing-based accident scenarios Also, its inclusion in the ITS is not necessary to adequately protect the health and safety of the public. The basic requirements for maintaining OPERABILITY are still retained in the technical specifications."

Per DOC 10-17-A, the SR to measure ventilation system flow rate is not identified as a separate SR in the ITS because it is verified as part of the other in-place filter tests that are specified in ITS 5.5.11. The same DOC applies to CTS SR 4.7.6.1 b 3 for Diablo Canyon, CTS SR 4.9.13 b 3 for Wolf creek, and CTS SR 4.7.7 b 3 for Callaway for the same reason.

DOC 10-08 A has been revised to show that some CTS SRs were moved to the ITS SRs.

ATTACHED PAGES:

Table 5.5-8

Attachment 13 CTS 3/4.7 - Plant systems

Encl. 3A	12
Encl 3B	11

TABLE Q5.5-8						
DCPP CTS SR	WC CTS SR	CA CTS SR	CP CTS SR	VFTP	ITS SR	Licensee Controlled Document
4.7.5.1 a	4.7.6 a	4.7.6 a	N/A			X
4.7.5.1 b 1	4.7.6 b	4.7.6 b	4.7.7.1 a		3.7.10.1	
4.7.5.1 b 2	N/A	N/A	N/A			3.7.10 Bases
4.7.5.1 b 3	N/A	N/A	N/A			3.7.10 Bases
			4.7.7.1b	ITS 5.5.11	3.7.10.2	
4.7.5.1 c 1	4.7.6 c 1	4.7.6 c 1	4.7.7.1 b 1	ITS 5.5.11a&b		
4.7.5.1 c 2	4.7.6 c 2	4.7.6 c 2	4.7.7.1 b 2	ITS 5.5.11c		
4.7.5.1 c 3	4.7.6 c 3	4.7.6 c 3	4.7.7.1 b 3	See DOC 10-17-A		
4.7.5.1 d	4.7.6 d	4.7.6 d	4.7.7.1 c	ITS 5.5.11 & 5.5.11c	3.7.10.2	
4.7.5.1 e 1	4.7.6 e 1	4.7.6 e 1	4.7.7.1 d 1	ITS 5.5.11d	3.7.10.2	
4.7.5.1 e 2	4.7.6 e 2	4.7.6 e 2	4.7.7.1 i		3.7.10.3	
4.7.5.1 e 3	4.7.6 e 3	4.7.6 e 3	4.7.7.1 j		3.7.10.4	
4.7.5.1 e 4	4.7.6 e 4	4.7.6 e 4	4.7.7.1 d 2	ITS 5.5.11e	3.7.10.2	
4.7.5.1 f	4.7.6 f	4.7.6 f	4.7.7.1 e	ITS 5.5.11 & 5.5.11a	3.7.10.2	
4.7.5.1 g	4.7.6 g	4.7.6 g	4.7.7.1 f	ITS 5.5.11 & 5.5.11b	3.7.10.2	
			4.7.7.1 g	ITS 5.5.11 & 5.5.11a	3.7.10.2	
			4.7.7.1 h	ITS 5.5.11 & 5.5.11b	3.7.10.2	
4.7.6.1 a 1	4.9.13 a	4.7.7 a	4.7.8a		3.7.12.1 DC&CP 3.7.13.1 WC&CA	3.7.12.1 Bases CP
4.7.6.1 a 2	N/A	N/A	N/A			3.7.12.1 Bases
			4.7.8b	ITS 5.5.11	3.7.12.2	
4.7.6.1 b 1	4.9.13 b 1	4.7.7 b 1	4.7.8 b 1	ITS 5.5.11a&b	3.7.12.2 DC 3.7.13.2 WC&CA NA CP	
4.7.6.1 b 2	4.9.13 b 2	4.7.7 b 2	4.7.8 b 2	ITS 5.5.11c	3.7.12.2 DC 3.7.13.2 WC&CA NA CP	
4.7.6.1 b 3	4.9.13 b 3	4.7.7 b 3	N/A	See DOC 10-17-A		
4.7.6.1 c	4.9.13 c	4.7.7 c	4.7.8 c	ITS 5.5.11 & 5.5.11c	3.7.12.2 DC&CP 3.7.13.2 WC&CA	
4.7.6.1 d 1	4.9.13 d 1	4.7.7 d 1	4.7.8 d 1	ITS 5.5.11d	3.7.12.2 DC&CP 3.7.13.2 WC&CA	

DCPP CTS SR	WC CTS SR	CA CTS SR	CP CTS SR	VFTP	ITS SR	Licensee Controlled Document
4.7.6.1 d 2	4.7.7 b 2	4.7.7 d 3	4.7.8 d 2		3.7.12.3 DC&CP 3.7.13.3 WC&CA	
4.7.6.1 d 3	4.9.13 d 2	4.7.7 d 4	4.7.8 d 3	ITS 5.5.11e	3.7.12.2 DC&CP 3.7.13.2 WC&CA	
4.7.6.1 d 4	N/A	N/A	N/A		3.7.12.6	3.7.12.6 Bases
4.7.6.1 e	4.9.13 e	4.7.7 e	4.7.8 e	ITS 5.5.11 & 5.5.11a	3.7.12.2 DC&CP 3.7.13.2 WC&CA	
4.7.6.1 f	4.9.13 f	4.7.7 f	4.7.8 f	ITS 5.5.11 & 5.5.11b	3.7.12.2 DC&CP 3.7.13.2 WC&CA	
N/A	4.7.7 b 1	4.7.7 d 2	4.7.8 d 4		3.7.13.4 WC&CA 3.7.12.4 CP	
			4.7.8d-new		3.7.12.6 CP	

CHANGE NUMBER	NSHC	DESCRIPTION
10-08	A	The description of the ventilation filter specific testing requirements and the required surveillances are moved to the Ventilation Filter Testing Program (VFTP) as defined in the Administrative Controls of the ITS. No technical changes to requirements or test specifics except as noted by separate change numbers are made. A new SR is added that requires [CR ventilation and primary plant ventilation] system filter testing in accordance with the VFTP. The requirements of this specification are: 1) moved to Section 5.5.11 of the ITS~~; or~~ 2) moved to ITS SRs. ~~or 2) deleted since they are duplicated in Regulatory Guide (RG) 1.52, revision 2. [ANSI N510-1980, or ANSI N 509-1980.].~~ Q-5.5-8
10-09	LS-27	Not applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
10-10	TR-1	The SR is revised to allow credit for an actual actuation, if one occurs, to satisfy the SRs. The identification of the initiating signal is moved to the Bases.
10-11	LS-19	The frequency of the surveillance requiring verification of the CR ventilation system capability to maintain a positive pressure is relaxed to [18] months on a STB, consistent with NUREG-1431. The new frequency requires one of the 2 trains to be tested every [18] months instead of both trains every [18] months. The most likely cause of a failure to achieve the required pressure is a failure of the ventilation pressure boundary. Thus when one train successfully demonstrates the ability to maintain the pressure, in all likelihood the other train will also. This results in less testing of the CR ventilation system than is required by the CTS. DC-ALL-001
10-12	LS-32	Not applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
10-13	LG	Not applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
10-14	A	Not applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
10-15	LG	Not applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
10-16	LG	Not applicable to CPSES. See Conversion Comparison Table (enclosure 3B).
10-17	A	The SR to measure ventilation system 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).

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
09-07 A	A note is added to the [SSW] surveillance that clarifies system operability requirements. Isolation of [SSW] flow to individual components does not render the system inoperable.	NO; ASW only supplies CCW heat exchangers.	YES	YES	YES
10-01 LG	The DCPP specific text description, definition of a ventilation train, is deleted from the LCO and moved to the Bases.	YES	NO	NO	NO
10-02 M	The APPLICABILITY and ACTIONS are revised to include "during movement of irradiated fuel assemblies."	YES	NO; part of CTS.	YES	YES
10-03 LS-7	The SR for the control room ventilation system is revised to require the filtration units without electric heaters to be tested for only 15 minutes instead of 10 hours.	NO; Plant configuration includes heaters.	YES	NO; refer to 10-22-M.	NO; refer to 10-22-M.
10-04 A	An ACTION statement is added to require entering 3.0.3 if two trains of the control room (CR) ventilation filter system are inoperable in MODES 1, 2, 3, or 4.	YES	YES	YES	YES
10-05 LS-18	A new option is added to the ACTIONs by NUREG-1431 that allows the suspension of CORE ALTERATIONS or movement of irradiated fuel versus placing the ventilation system in the recirculation mode.	YES	NO; part of CTS.	YES	YES
10-06 LG	The details and description of the required actions and the monthly SRs for train operability are moved to the Bases.	YES	NO; not in CTS.	YES	YES
10-07 LG	The surveillance that verifies control room temperature once per 12 hours is moved to a licensee-controlled document.	YES; moved to ECG.	NO; not in CTS.	YES; moved to USAR.	YES; moved to FSAR.
10-08 A	The description of the ventilation filter specific testing requirements are moved to the VFTP, as defined ~~in the Administrative Controls of the ITS, or~~ moved to ITS SRs. ~~deleted as being duplicated in the applicable RGs or Standards.~~ A SR is added that requires [CR ventilation and primary plant ventilation system] filter testing in accordance with the VFTP.	YES	YES	YES	YES Q-5.5-8

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 5.5-14 **APPLICABILITY:** CP

REQUEST: ITS 5.5.11.e and CTS 4.7.8.d.3 (Comanche Peak)

Comment: The value for the ESF filtration unit is 100 plus or minus 5 kW in the CTS and 100 plus 5 kW in the ITS. Provide correction or justify change.

FLOG Response: The ITS value for the ESF filtration unit is corrected to be 100 plus or minus 5 kw.

ATTACHED PAGES:

Encl 5A 5.0-23

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Penetration	RH
Control Room Emergency filtration unit	0.2%	70%
Control Room Emergency pressurization unit	0.2%	70%
Primary Plant Ventilation System - ESF filtration unit	1.0%	70%

B-PS

~~Reviewer's Note: Allowable penetration = 100% - methyl iodide efficiency for charcoal credited in safety evaluation/(safety factor). safety factor of = 5 for systems with heaters. = [7] for systems without heater.~~

d. Demonstrate at least once per 18 months for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980~~1989~~ at the system flowrate specified below ± 10% 5.5-11 B-PS

ESF Ventilation System	Delta P	Flowrate
Control Room Emergency filtration unit	8.0 in WG	8000 CFM
Control Room Emergency pressurization unit	9.5 in WG	800 CFM
Primary Plant Ventilation System - ESF filtration unit.	8.5 in WG	15000 CFM

B-PS

e. Demonstrate at least once per 18 months that the heaters for each of the ESF systems dissipate the value specified below ~~± 10%~~ when tested in accordance with ANSI/ASME N510-1980~~1989~~. 5.5-11 B-PS

ESF Ventilation System	Wattage
Control Room Emergency pressurization unit	10 ± 1 kW
Primary Plant Ventilation System - ESF filtration unit.	100 ± 5 kW

B-PS Q-5.5-14

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Penetration	RH
Control Room Emergency filtration unit	0.2%	70%
Control Room Emergency pressurization unit	0.2%	70%
Primary Plant Ventilation System - ESF filtration unit	1.0%	70%

B-PS

~~Reviewer's Note: Allowable penetration = 100% - methyl iodide efficiency for charcoal credited in safety evaluation/(safety factor). safety factor of = 5 for systems with heaters. = [7] for systems without heater.~~

d. Demonstrate at least once per 18 months for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980~~1989~~ at the system flowrate specified below ± 10% 5.5-11 B-PS

ESF Ventilation System	Delta P	Flowrate
Control Room Emergency filtration unit	8.0 in WG	8000 CFM
Control Room Emergency pressurization unit	9.5 in WG	800 CFM
Primary Plant Ventilation System - ESF filtration unit.	8.5 in WG	15000 CFM

B-PS

e. Demonstrate at least once per 18 months that the heaters for each of the ESF systems dissipate the value specified below ~~± 10%~~ when tested in accordance with ANSI/ASME N510-1980~~1989~~. 5.5-11 B-PS

ESF Ventilation System	Wattage
Control Room Emergency pressurization unit	10 ± 1 kW
Primary Plant Ventilation System - ESF filtration unit.	100 ± 5 kW

B-PS

Q-5.5-14

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 5.6-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: ITS 5.6.5 a.7&8, Changes 03-14&15 M

Comment: It is true that the additions would make the COLR more restrictive however, the removal of the specific values from the TS is a less restrictive change that needs to be justified. Provide justification.

FLOG RESPONSE: DOC-03-14-M describes the addition of the SHUTDOWN MARGIN (SDM) limits and the Moderator Temperature Coefficient (MTC) limits to the Administrative Program description of the CORE OPERATING LIMITS REPORT (COLR). As stated, this change is more restrictive to the COLR. The change for moving the actual limits from the technical specifications to the licensee controlled COLR are addressed and justified by DOC 01-01-LG (SDM) found in Section 3.1 (not applicable to CPSES) and DOC 03-07-LG (MTC) found in Section 3.1 (applicable to DCPP only).

DOC-03-15-M, in a similar way, adds the Refueling Boron Concentration limits to the Administrative Program description of the COLR. The change moving these limits to the licensee controlled COLR is addressed and justified by DOC 01-02-LG found in Section 3.9.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 5.7-1 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: ITS 5.7.2 and Difference 5.7-2

Comment: TSTF-167 has been rejected by the NRC. Use current ITS.

FLOG RESPONSE: Traveler TSTF-258 has been submitted to the NRC for review. This traveler superseded travelers, TSTF-86, TSTF-121, and TSTF-167. TSTF-258 is based on the recommendations in the April 9, 1997 letter from C. Grimes (NRC) to J. Davis (NEI), with some exceptions. The FLOG submittals have been revised to incorporate TSTF-258 and encompass the NRC comments of 6/11/98. Additional technical changes made to Section 5.7 are identified and justified. (See JFD 5.7-2 which revises ITS 5.7.2e consistent with CTS 6.12, and JFD 5.7-4 which revises ITS 5.7.2f consistent with CTS 6.12) The latest industry status on TSTF-258 is that the NRC has requested changes to Section 5.7, High Radiation Area.

ATTACHED PAGES:

See markups associated with Comment Number Q 5.2-1.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: TR 5.0-003 **APPLICABILITY:** CA, CP, DC, WC

REQUEST: TS 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)", was revised to incorporate changes based upon WOG-67. WOG-67 has been approved by the TSTF and is designated as TSTF-233. This traveler has been submitted to the NRC and the latest traveler reports indicate that TSTF-233 has been approved by the NRC. The attached pages reflect changes associated with WOG-67 being designated as TSTF-233.

ATTACHED PAGES:

Encl 5A	Traveler Status sheet
Encl 6A	4
Encl 6B	5

Industry Travelers Applicable to CTS Section 6.0/ITS 5.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS	
TSTF-9, Rev. 1	Incorporated	B-PS	NRC Approved.	
TSTF-37, Rev. 1	Incorporated	5.6-2	DCPP only	
TSTF-52, Rev. 1	Incorporated	5.5-4	Incorporated draft Rev. 1 per Q3.6.1-6z	Q-3.6.1-6
TSTF-65, Rev. 1	~~Not~~ Incorporated	~~NA~~5.2-9	~~Not~~ NRC approved ~~as of traveler cut-off date~~.	TR-5.0-005
TSTF-106, Rev. 1	Not Incorporated	NA	Retain CTS	
TSTF-115	Not Incorporated	NA	Not NRC approved as of traveler cut-off date.	
TSTF-118	Incorporated	5.5-8	NRC Approved	TR-5.0-006
~~TSTF-119~~	~~Not Incorporated~~	~~NA~~	~~Retain CTS~~	TR-5.0-006
TSTF-120, Rev. 1	Not Incorporated	NA	Retain CTS	TR-5.0-006
~~TSTF-121~~	~~Incorporated~~	~~5.2-2~~		Q-5.2-1
TSTF-152	Incorporated	5.6-4	NRC Approved	TR-5.0-006
~~TSTF-167~~	~~Incorporated~~	~~5.7-2~~		Q-5.2-1
~~WOG-67, Rev.1~~ TSTF -233	Incorporated	5.6-5		TR-5.0-003
WOG-72	Incorporated	5.5-13		
~~WOG-85~~TSTF-237	Incorporated	5.5-14		Q-5.5-2
~~Proposed Traveler~~ TSTF-258	Incorporated	5.2-2, 5.2-3, 5.2-6, 5.3-2, 5.5-1, 5.5-16, 5.6-6, 5.7-1		Q-5.2-1

CHANGE
NUMBER JUSTIFICATION

5.6-3 This change revises the report date in Section 5.6.2 "Annual Radiological Environmental Operating Report" to be consistent with current TS.

5.6-4 This change revises Sections 5.6.1 and 5.6.3, "Occupational Radiation Exposure Report" and "Radioactive Effluent Release Report", respectively, per NRC letter dated July 28, 1995, "Changes to Technical Specifications Resulting from 10CFR20 and 50.36a Changes" (From Christopher I. Grimes to Owners Group Chairs). This change is consitent with TSTF-152.

5.6-5 [ ] PORV lift settings are referenced in PTLR section per ~~WOG-67, Rev. 1~~ TSTF-233. TR.5.0-003

5.6-6 The ITS requirement to provide documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves is deleted. The reporting of pressurizer safety and relief valve failures and challenges is based on the guidance in NUREG-0694, "TMI-Related Requirements for New Operating Licensees." The guidance of NUREG-0694 states: "Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report." NRC Generic Letter 97-02, "Revised Contents of the Monthly Operating Report" requests the submittal of less information in the monthly operating report. The generic letter identifies what needs to be reported to support the NRC Performance Indicator Program, and availability and capacity statistics. The generic letter does not specifically identify the need to report challenges to the pressurizer safety and relief valves. This change is consistent with TSTF-258. Q-5.2-1

5.7-1 ~~This change revises High Radiation Area to incorporate changes consistent with [the current TS]. Specifically, distances from the radiation source.~~ Section 5.7 is revised in accordance with 10 CFR 20.1601(c) and updates the acceptable alternate controls to those given in 10 CFR 20.1601. These changes are consistent with the draft Generic Letter (93-XX) on proposed changes to STS NUREGs based on the new 10 CFR 20 and the letter from C. Grimes, NRC, to J. Davis, NEI dated April 9, 1997. This change is consistent with TSTF-258 and encompasses the NRC comments on 6/11/98. Additional technical changes made to Section 5.7 are identified and justified. Q-5.2-1

5.7-2 ~~This change revises "unauthorized" to "inadvertent" in the High Radiation Area section to reflect NRC's position as stated in RG 8.38, Section 1.5 regarding physical barriers for High Radiation Areas. This is consistent with traveler TSTF-167.~~ ITS 5.7.2.e is revised consistent with CTS 6.12 that allows any individual or group of individuals to enter a high-high radiation area (dose rates greater than 1.0 rem/hour at 30 cm) accompanied by an individual qualified in radiation protection procedures with a radiation dose rate monitoring device. The qualified individual is responsible for providing positive control and shall perform periodic radiation surveillances at the frequency specified in the RWP. The CTS requirements allow the qualified individual to enter a locked high radiation area with plant workers without first having to enter the area to Q-5.2-1

TECH SPEC CHANGE		APPLICABILITY				
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY	
5.6-3	Revises report date in Section 5.6.2, "Annual Radiological Environmental Operating Report" to be consistent with current TS.	Yes Consistent with original TS and LA 78/77.	Yes See LA 42/28	Yes	Yes	
5.6-4	Revises Sections 5.6.1 and 5.6.3, "Occupational Radiation Exposure Report" and "Radioactive Effluent Release Report," respectively, per NRC letter dated July 28, 1995, "Changes to Technical Specifications Resulting from 10 CFR 20 and 50.36a Changes" (From Christopher I. Grimes to Owners Groups Chairs). This change is consistent with TSTF-152.	Yes	Yes	Yes	Yes	
5.6-5	[ ] ~~PORV lift settings~~ are referenced in PTLR section per TSTF-233 ~~WOG-67, Rev. 1~~.	Yes	Yes	Yes	Yes	TR-5.0-003
5.6-6	The ITS requirement to provide documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves is deleted.	Yes	Yes	Yes	Yes	Q-5.2-1
5.7-1	~~Revises High Radiation Area to incorporate consistent changes with [10 CFR 20.1601].~~ Section 5.7 is revised in accordance with 10 CFR 20.1601(c) and updates the acceptable alternate controls to those given in 10 CFR 20.1601.	Yes	Yes	Yes	Yes	Q-5.2-1
5.7-2	~~Changes "unauthorized" to "inadvertent" in the High Radiation Area section to reflect the NRC's position as stated in RG 8.38. Section 1.5 regarding physical barriers for High Radiation Areas. This change is consistent with TSTF-167.~~ ITS 5.7.2.e is revised consistent with CTS 6.12 that allows any individual or group of individuals to enter a high-high radiation area (dose rates greater than 1.0 rem/hour at 30 cm) accompanied by an individual qualified in radiation protection procedures with a radiation dose rate monitoring device.	Yes	Yes	Yes	Yes	Q-5.2-1

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: TR 5.0-005 **APPLICABILITY:**, CP

REQUEST: This change incorporates NRC approved traveler TSTF-65 Rev. 1 which provides for the optional use of generic titles for certain utility positions.

ATTACHED PAGES:

Encl 2	6-1
Encl 3A	1a
Encl 3B	2a
Encl 5A	Traveler Status sheet, 5.0-1 and 5.0-2
Encl 6A	1a
Encl 6B	2

6.1 RESPONSIBILITY

6.1.1 The Vice President of Nuclear Operations* shall be responsible for overall operation of the units and shall delegate in writing the succession to this responsibility during his absence.

The Vice President of Nuclear Operations*, or his designee, in accordance with approved administrative procedures, shall approve prior to implementation, each proposed test or experiment and proposed changes and modifications to unit systems or equipment that affect nuclear safety.

6.1.2 The Shift Manager shall be responsible for the control room command function. ~~A management directive to this effect, signed by the Group Vice President, Nuclear Production shall be reissued annually to all station personnel.~~ 1-01-A During any absence of the Shift Manager from the control room while the unit is in MODE 1, 2, 3 or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the Shift Manager from the Control Room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

6.2 ORGANIZATION

6.2.1 ONSITE AND OFFSITE ORGANIZATION

An onsite and an offsite organization shall be established for unit operation and corporate management, respectively. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in the equivalent forms of documentation. These requirements shall be documented in the FSAR.

b. The Vice President of Nuclear Operations* shall be responsible for overall site safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. ~~The Group Vice President of Nuclear Production~~ A corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety. 1-16-LG TR-5.0-005

* Duties may be performed by the Plant Manager if that organizational position is assigned

CHANGE NUMBER	JUSTIFICATION
5.2-4	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.2-5	Not used.
5.2-6	~~This change revises Section 5.2.2f to describe the current [licensing basis] for the Shift Technical Advisor (STA).~~ This change revises ITS Section 5.2.2.f (ISTS Section 5.2.2.g) to describe the current [TS] and to eliminate the title of "Shift Technical Advisor (STA)." STAs are not used at all plants (the function may be fulfilled by one of the other on-shift individuals). This Section is revised so that it does not imply that the STA and the Shift Supervisor must be different individuals. Option 1 of the Commission Policy Statement on Engineering Expertise on Shift is satisfied by assigning an individual with specified educational qualifications to each operating crew as one of the SROs (preferably the Shift Supervisor) required by 10 CFR 50.54(m)(2)(i) to provide the technical expertise on shift. However, the ISTS 5.2.2.g wording of, "the STA shall provide ... support to the Shift Supervisor...," is considered to be easily misinterpreted to require separate individuals. Therefore, the wording is revised so that the STA function may be provided by either a separate individual or the individual who also fulfills another role in the shift command structure. This change is consistent with TSTF-258. Q-5.2-1
5.2-7	This change revises 5.2.2c to add a note that a single Radiation Protection Technician and a single Chemistry Technician may fulfill the requirements for both units. This statement was added to clarify operational practices consistent with current licensing basis.
5.2-8	A generic title has replaced the plant specific utility title for the corporate officer having responsibility for overall plant safety. A reviewer's note was added describing when the use of generic titles is allowed. A statement was also added indicating that the plant specific titles for any generic titles used will be provided in the FSAR. This change is consistent with TSTF-65. TR-5.0-005
5.3-1	This change revises Section 5.3.1 to be consistent with the current TS regarding plant staff qualifications and training.
5.3-2	New paragraph 5.3.2 is added to ensure that there is not misunderstanding when complying with 10 CFR 55.4 requirements. The Definitions in 10 CFR 55.4 state: "Actively performing the functions of an operator or senior operator means that an individual has a position on the shift crew that requires the individual to be licensed as defined in the facility's technical specifications, and that ...." Placing this paragraph in the ITS meets the 10 CFR 55.4 requirement for defining in the facility's technical specifications the function performed by licensed individuals per 10 CFR 50.54(m). Adding this paragraph is consistent with the recommendations in the April 9, 1997 letter from C. Grimes to J. Davis and TSTF-258. Q-5.2-1

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-15 A	This change revises the CTS to eliminate the title of "Shift Technical Advisor (STA)."	Yes	Yes	Yes	Yes Q-5.2-1
01-16 LG	A generic title has replaced the plant specific utility title for the corporate officer having responsibility for overall plant safety and the plant specific title is moved to the FSAR.	No- Retained CTS	Yes	No- Retained CTS	No- Retained CTS TR-5.0-005

Industry Travelers Applicable to CTS Section 6.0/ITS 5.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS	
TSTF-9, Rev. 1	Incorporated	B-PS	NRC Approved.	
TSTF-37, Rev. 1	Incorporated	5.6-2	DCPP only	
TSTF-52, Rev. 1	Incorporated	5.5-4	Incorporated draft Rev. 1 per Q3.6.1-6z	Q-3.6.1-6
TSTF-65, Rev. 1	~~Not~~ Incorporated	~~NA~~5.2-9	~~Not~~ NRC approved ~~as of traveler cut-off date.~~	TR-5.0-005
TSTF-106, Rev. 1	Not Incorporated	NA	Retain CTS	
TSTF-115	Not Incorporated	NA	Not NRC approved as of traveler cut-off date.	
TSTF-118	Incorporated	5.5-8	NRC Approved	TR-5.0-006
~~TSTF-119~~	~~Not Incorporated~~	~~NA~~	~~Retain CTS~~	TR-5.0-006
TSTF-120, Rev. 1	Not Incorporated	NA	Retain CTS	TR-5.0-006
~~TSTF-121~~	~~Incorporated~~	~~5.2-2~~		Q-5.2-1
TSTF-152	Incorporated	5.6-4	NRC Approved	TR-5.0-006
~~TSTF-167~~	~~Incorporated~~	~~5.7-2~~		Q-5.2-1
~~WOG-67, Rev.1~~ TSTF -233	Incorporated	5.6-5		TR-5.0-003
WOG-72	Incorporated	5.5-13		
~~WOG-85~~TSTF-237	Incorporated	5.5-14		Q-5.5-2

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

~~[Reviewer's Note: Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second methods is adaptable to those unit staffs requiring special titles because of unique organizational structures.~~

5.2-9
TR-5.0-005

~~The ANSI Standard shall be the same ANSI Standard referenced in Section 5.3, Unit Staff Qualifications. If alternative titles are used, all requirements of these Technical Specifications apply to the position with the alternative title as apply with the specified title. Unit staff titles shall be specified in the Final Safety Analysis Report or Quality Assurance Plan. Unit staff titles shall be maintained and revised using those procedures approved for modifying/revising the Final Safety Analysis Report or Quality Assurance Plan.]~~

5.1.1 The ~~[Plant Superintendent]~~ Plant Manager* shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. B-PS

The ~~[Plant Superintendent]~~ Plant Manager* or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety. B-PS

5.1.2 The ~~[Shift Supervisor (SS)]~~ Shift Manager shall be responsible for the control room command function. During any absence of the ~~[SS]~~ Shift Manager from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the ~~[SS]~~ Shift Manager from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function. B-PS

* Duties may be performed by the Vice President of Nuclear Operations if that organizational position is assigned. B-PS

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications. ~~shall be documented in the FSAR~~; 5.2-9 TR-5.0-005 B

b. The ~~[Plant Superintendent]~~ Plant Manager* shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant; B-PS

c. ~~The [a specified corporate executive position] Group Vice President, Nuclear Production~~ A specified corporate officer ~~shall have~~ corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and ~~B-PS~~ 5.2-9 TR-5.0-005

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

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* Duties may be performed by the Vice President of Nuclear Operations if that organizational position is assigned. B-PS

(continued)

CHANGE NUMBER	JUSTIFICATION
5.2-4	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.2-5	Not used.
5.2-6	~~This change revises Section 5.2.2f to describe the current [licensing basis] for the Shift Technical Advisor (STA).~~ This change revises ITS Section 5.2.2.f (ISTS Section 5.2.2.g) to describe the current [TS] and to eliminate the title of "Shift Technical Advisor (STA)." STAs are not used at all plants (the function may be fulfilled by one of the other on-shift individuals). This Section is revised so that it does not imply that the STA and the Shift Supervisor must be different individuals. Option 1 of the Commission Policy Statement on Engineering Expertise on Shift is satisfied by assigning an individual with specified educational qualifications to each operating crew as one of the SROs (preferably the Shift Supervisor) required by 10 CFR 50.54(m)(2)(i) to provide the technical expertise on shift. However, the ISTS 5.2.2.g wording of, "the STA shall provide ... support to the Shift Supervisor...," is considered to be easily misinterpreted to require separate individuals. Therefore, the wording is revised so that the STA function may be provided by either a separate individual or the individual who also fulfills another role in the shift command structure. This change is consistent with TSTF-258. Q-5.2-1
5.2-7	This change revises 5.2.2c to add a note that a single Radiation Protection Technician and a single Chemistry Technician may fulfill the requirements for both units. This statement was added to clarify operational practices consistent with current licensing basis.
5.2-8	Not applicable to CPSES. See conversion comparison table (enclosure 6B).
5.2-9	A generic title has replaced the plant specific utility title for the corporate officer having responsibility for overall plant safety. A reviewer's note was added describing when the use of generic titles is allowed. A statement was also added indicating that the plant specific titles for any generic titles used will be provided in the FSAR. This change is consistent with TSTF-65. DC-5.0-002 TR-5.0-005
5.3-1	This change revises Section 5.3.1 to be consistent with the current TS regarding plant staff qualifications and training.
5.3-2	New paragraph 5.3.2 is added to ensure that there is not misunderstanding when complying with 10 CFR 55.4 requirements. The Definitions in 10 CFR 55.4 state: "Actively performing the functions of an operator or senior operator means that an individual has a position on the shift crew that requires the individual to be licensed as defined in the facility's technical specifications, and that ...." Placing this paragraph in the ITS meets the 10 CFR 55.4 requirement for defining in the facility's technical specifications the function performed by licensed individuals per 10 CFR 50.54(m). Adding this paragraph is consistent with the recommendations in the April 9, 1997 letter from C. Grimes to J. Davis and TSTF-258. Q-5.2-1

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 5.0

TECH SPEC CHANGE		APPLICABILITY				
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY	
5.2-6	Revises Section 5.2.2f to describe the current [licensing basis] ~~for STA.~~ and eliminate the title of Shift Technical advisor (STA).	Yes	Yes LA 50/36 moved text to FSAR Section 13.1 which permits on-shift SRO tc fill STA position	Yes	Yes	Q-5.2-1
5.2-7	Revises 5.2.2c to add note that a single Radiation Protection Technician and a single Chemistry Technician may fulfill the requirments for both units.	No Not current procedure or operational requirement	Yes	No Wolf Creek is a single unit site	No Callaway is a single unit site	
5.2-8	Revises Sections 5.2.2 and 5.3.1 to reflect License Amendment 128/126 dated 6/11/98 which changed requirements for the DCPP Operations Director.	Yes LA128/126	No	No	No	DC-5.0-002
5.2-9	A generic title has replaced the plant specific utility title for the corporate officer having responsibility for overall plant safety.	No	Yes	No	No	TR-5.0-005
5.3-1	Revises Section 5.3.1 to be consistent with current TS regarding plant staff qualifications and training.	Yes LA 43/42	Yes	Yes	Yes	
5.3-2	New paragraph 5.3.2 is added to ensure that there is not misunderstanding when complying with 10 CFR 55.4 requirements.	Yes	Yes	Yes	Yes	Q-5.2-1
5.5-1	Revises Section 5.5.4, "Radioactive Effluent Controls Program," to reflect new 10 CFR Part 20 requirements. ~~and NRC letter dated 7/28/95 consistent with proposed traveler.~~	Yes	Yes	Yes	Yes	Q-5.2-1
5.5-2	Revises Section 5.5.3, "Post Accident Sampling," to ensure the capability to obtain and analyze radioactive "iodines" in lieu of "gases." This change is consistent with the current TS and plant practices.	Yes	Yes	Yes	Yes	
5.5-3	Revises Section 5.5.8, "Inservice Testing Program," to delete "including applicable supports." This change is consistent with the current TS.	Yes	Yes	Yes	Yes	

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: TR 5.0-006 **APPLICABILITY:** CA, CP, DC, WC

REQUEST:

Revise the Traveler Status Sheet to reflect the latest status and revisions of the following travelers:

- TSTF-118 - NRC Approved
- TSTF-119 - NRC Rejected
- TSTF-120, Rev. 1
- TSTF-152 - NRC Approved

ATTACHED PAGES:

Encl. 5A Traveler Status Page

Industry Travelers Applicable to CTS Section 6.0/ITS 5.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS	
TSTF-9, Rev. 1	Incorporated	B-PS	NRC Approved.	
TSTF-37, Rev. 1	Incorporated	5.6-2	DCPP only	
TSTF-52, Rev. 1	Incorporated	5.5-4	Incorporated draft Rev. 1 per Q3.6.1-6z	Q-3.6.1-6
TSTF-65, Rev. 1	~~Not~~ Incorporated	~~NA~~5.2-9	~~Not~~ NRC approved ~~as of traveler cut-off date~~.	TR-5.0-005
TSTF-106, Rev. 1	Not Incorporated	NA	Retain CTS	
TSTF-115	Not Incorporated	NA	Not NRC approved as of traveler cut-off date.	
TSTF-118	Incorporated	5.5-8	NRC Approved	TR-5.0-006
~~TSTF-119~~	~~Not Incorporated~~	~~NA~~	~~Retain CTS~~	TR-5.0-006
TSTF-120, Rev. 1	Not Incorporated	NA	Retain CTS	TR-5.0-006
~~TSTF-121~~	~~Incorporated~~	~~5.2-2~~		Q-5.2-1
TSTF-152	Incorporated	5.6-4	NRC Approved	TR-5.0-006
~~TSTF-167~~	~~Incorporated~~	~~5.7-2~~		Q-5.2-1
~~WOG-67, Rev.1~~ TSTF -233	Incorporated	5.6-5		TR-5.0-003
WOG-72	Incorporated	5.5-13		
~~WOG-85~~TSTF-237	Incorporated	5.5-14		Q-5.5-2