JLS CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS

CTS 3.4 - REACTOR COOLANT SYSTEM

ITS 3.4 - REACTOR COOLANT SYSTEM

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION AND LICENSEE INITIATED ADDITIONAL CHANGES

INDEX OF ADDITIONAL INFORMATION

ADDITIONAL INFORMATION NUMBER	APPLICABILITY	ENCLOSED
3.4.Gen-1	CA, CP, DC, WC	YES
3.4.1-1	CA, CP, DC, WC	YES
3.4.1-2	CA, CP, DC, WC	YES
3.4.1-3	DC	NA
3.4.2-1	CA, CP, DC, WC	YES
3.4.3-1	CA, CP, DC, WC	YES
3.4.4-1	CA, CP, DC, WC	YES
3.4.5-1	CA, WC	YES
3.4.5-2	CA, WC	YES
3.4.5-3	CA, WC	YES
3.4.5-4	CP	NA
3.4.6-1	CA, CP, DC, WC	YES
3.4.6-2	DC	NA
3.4.7-1	WC	YES
3.4.7-2	WC	YES
3.4.7-3	CA	NA
3.4.8-1	CA, CP, DC, WC	YES
3.4.8-2	DC	NA
3.4.9-1	CA, CP, DC, WC	YES
3.4.9-2	CA	NA
3.4.9-3	CP, DC, WC	YES
3.4.9-4	DC	NA
3.4.10-1	CA, CP, DC, WC	YES
3.4.11-1	CA, CP, DC, WC	YES
3.4.11-2	CA, CP, DC, WC	YES
3.4.11-3	CA, CP, DC, WC	YES
3.4.11-4	CA, CP, DC, WC	YES
3.4.11-5	WC	YES
3.4.11-6	CA, CP, WC	YES
3.4.12-1	CA, CP, DC, WC	YES
3.4.12-2	CA, CP, DC, WC	YES
3.4.12-3	CA, CP, DC, WC	YES
3.4.12-4	CA, CP, WC	YES
3.4.12-5	CA, WC	YES
3.4.12-6	DC	NA
3.4.12-7	CP	NA
3.4.12-8	CP	NA
3.4.13-1	DC, WC	YES
3.4.13-2	CA, DC, WC, CP	YES
3.4.13-3	CA, CP, WC	YES
3.4.13-4	CP, DC	NA
3.4.13-5	DC	NA
3.4.13-6	CA	NA

INDEX OF ADDITIONAL INFORMATION
(cont.)

ADDITIONAL INFORMATION NUMBER	APPLICABILITY	ENCLOSED
3.4.14-1	CA, CP, WC	YES
3.4.14-2	CP, DC, WC	YES
3.4.14-3	CA, CP, DC, WC	YES
3.4.14-4	CA, WC	YES
3.4.14-5	DC, WC	YES
3.4.15-1	CA, DC, WC	YES
3.4.15-2	CA, WC	YES
3.4.15-3	WC	YES
3.4.15-4	CP, DC	NA
3.4.15-5	DC	NA
3.4.16-1	CA, CP, DC, WC	YES
3.4.16-2	WC	YES
3.4.16-3	WC	YES
3.4.G-1	CP	YES
CA 3.4-002	CA	NA
CA 3.4-003	CA	NA
CA 3.4-004	CA, CP, DC, WC	YES
CP 3.4-004	CP	NA
DC 3.4-ED	DC	NA
DC ALL-001 (3.4 changes only)	DC	NA
DC ALL-002 (3.4 changes only)	DC	NA
DC ALL-005 (3.4 changes only)	DC	NA
DC 3.4-003	DC	NA
TR 3.4-004	CA, CP, DC, WC	YES
TR 3.4-005	CA, CP, DC, WC	YES
TR 3.4-006	CA, CP, DC, WC	YES
TR 3.4-009	CA, CP, DC, WC	YES
WC 3.4-001	WC, CP	YES
WC 3.4-002	CA, CP, DC, WC	YES
WC 3.4-004	WC	YES
WC 3.4-006	WC	YES
WC 3.4-007	WC, CA, DC	YES
WC 3.4-008	WC, CA	YES
WC 3.4-009	WC	YES
WC 3.4-0010	WC, CA	YES

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 3.4.Gen-1 **APPLICABILITY:** DC, CP, WC, CA

REQUEST: ITS 3.4.x Bases

General

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

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

FLOG RESPONSE: The submitted ITS Bases markups for Section 3.4 have been compared to the STS Bases. Some differences that were identified were in accordance with the markup methodologies (e.g., deletion of brackets and reviewer's notes). Most of the differences were editorial in nature and would not have affected the review. Examples of editorial changes are:

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

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

ATTACHED PAGES:

Encl. 5B B 3.4-15, B 3.4-17, B 3.4-19, B 3.4-21, B 3.4-22, B 3.4-25, B 3.4-27, B 3.4-28, B 3.4-34, B 3.4-38, B 3.4-39, B 3.4-42, B 3.4-47, B 3.4-48, B 3.4-49, B 3.4-52, B 3.4-55, B 3.4-64, B 3.4-68, B 3.4-69, B 3.4-71, B 3.4-72, B 3.4-74, B 3.4-75, B 3.4-80, B 3.4-87, B 3.4-88, B 3.4-98

ACTION

B.1 and B.2 (continued)

increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psig within 36 hours.

redline Q 3.4.GEN-1

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.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. ~~The RHR loops provide this circulation.~~

The operation of one RCP in MODES 3, 4, and 5 provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during RCS boron concentration reductions. With no reactor coolant loop in operation in either MODES 3,4, or 5, boron dilutions must be terminated and dilution sources isolated. The boron dilution analysis in thes MODES take credit for the mixing volume associated with having at least one reactor coolant loop in operation. (Ref.1).

RCS Loops – MODE 5 (Loops Filled) ~~have been identified in the NRC Policy Statement as important contributors to risk reduction~~ satisfies criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

66%, or equivalent

wide range

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level ≥ ~~10%~~. Q3.4.5-2 Q3.4.5-3

One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side ~~narrow~~ range water levels ≥ ~~10%~~. Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation. Q3.4.5-2 Q3.4.5-3

wide

66%

The narrow range level instrumentation may be used in MODE 5 if the one wide range level indicator per SG were unavailable.

Note 1 permits all RHR pumps to be removed from operation for ~~de-energized~~ ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are required to be performed without flow or pump noise. ~~designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again.~~ The 1 hour time period is adequate to perform the necessary testing, and operating experience has shown that boron

(continued)

BASES

APPLICABILITY

Q3.4.7-2

In MODE 5 with RCS loops filled ~~(as defined in plant procedures)~~, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side ~~narrow~~ wide range water level of at least two SGs is required to be ≥ ~~10%~~ 66%.

Q3.4.5-2
Q3.4.5-3

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
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).

ACTIONS

Q3.4.5-2
Q3.4.5-3

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side wide range water levels < ~~10%~~ 66%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Notes 1 and 4, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. Addition of borated water with a concentration greater than or equal to the minimum required RWST concentration but less than the actual RCS boron concentration shall not be considered a reduction in boron concentration. (Ref. 2). To prevent inadvertent criticality during a boron dilution, forced circulation from at least one RCP is required to provide proper mixing. ~~and preserve the margin to criticality in this type of operation.~~ The immediate Completion Times reflect the importance of maintaining operation for heat removal.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

66%, or equivalent

Q3.4.5-2
Q3.4.5-3

Verifying that at least two SGs are OPERABLE by ensuring their secondary side narrow range water levels are ≥ 10% ensures an alternate decay heat removal method is available via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

The narrow range level instrumentation may be used in MODE 5 if the one wide range level indicator per SG were unavailable.

SR 3.4.7.3

Q3.4.5-2
Q3.4.5-3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. If secondary side ~~narrow~~ wide range water level is ≥ 10% 66% in at least two SGs, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

~~None.~~ 1. USAR, Section 15.4.6

2. NRC letter (W. Reckley to N. Carns) dated November 22, 1993: "Wolf Creek Generating Station - Positive Reactivity Addition; Technical Specification Bases Change."

3. NRC Information Notice 95-35, "Degraded Ability of SGs to Remove Decay Heat by Natural Circulation."

CHANGE NUMBER	JUSTIFICATION
	plant conditions suitable for the precision heat balance. Since this parameter does not normally change significantly and the flow meters can be used in the interim, there is no need to perform this SR within the 24 hour period specified in NUREG-1431 Rev. 1. The 7 day period provides sufficient time to establish steady state plant thermohydraulic conditions and obtain equilibrium xenon. In addition, the THERMAL POWER specified in the Note would be changed from the generic value in brackets (90 % RTP) to 95 % RTP. This change is acceptable because it specifies a power level in better agreement with current operating procedures for performing a precision heat balance. Current TS do not specify a power level for this measurement.
3.4-41	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
3.4-42	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
3.4-43	A new Condition is added to LCO 3.4.1 to reflect the current licensing basis of Wolf Creek for RCS flow rate. License Amendment 61 approved revisions to incorporate the provisions of the RCS flow TS entitled "RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR" into the "DNB PARAMETERS" specification. These changes were made to support the use of VANTAGE 5H fuel with the Intermediate Flow Mixer grid feature. This amendment also approved operation at an increased power level. Q3.4.5-2 Q3.4.5-3
3.4-44	~~Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).~~ INSERT 6A-8b Q3.4.12-2
3.4-45	~~ITS 3.4.12 has been revised to move the Note for Required Action B.1 regarding CCP pump swap operations and the Applicability Note for accumulator isolation to the LCO, as discussed in traveler WOG-51, Rev. 1. Plant-specific time allowances for exceeding the LCO's number of [ECCS] pumps capable of injecting into the RCS are incorporated[, as discussed in CN 3.4-18]. These Notes detail situations where exceptions to the LCO are permitted and are more appropriately annotated under the LCO.~~ INSERT 6A-8a
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

INSERT 6A-8a Q 3.4.12-2

Consistent with traveler TSTF-285, ITS 3.4.12 has been revised to move the Note for Required Action B.1 regarding CCP pump swap operations and the Note under Applicability regarding accumulator isolation to the LCO. These Notes have been reworded for clarity and detail situations where exceptions to the LCO are permitted. Also, plant-specific time allowances for exceeding the LCO number of [ECCS] pumps capable of injecting into the RCS are incorporated [, as discussed in CN 3.4-18].

INSERT 6A-8b Q 3.4.5-2
Q 3.4.5-3

Steam generator levels for MODE 3, 4, and 5 are specified to ensure SG tubes are covered. The current TS did not ensure tube coverage.

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
3.4-38	Consistent with the 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. ~~Consistent with TSTF-108, the details on the method by which the RCS flow rate are verified are moved from the SR 3.4.1.4 to the Bases.~~	Yes	~~Yes~~ No	~~Yes~~ No	~~Yes~~ No Q3.4.1-1
3.4-39	The shutdown requirements of ITS 3.4.11 would require the plant 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 all 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 reactor power and to provide additional time to perform an RCS precision [illegible] 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 - Specific to DCPP	No	No	No
3.4-43	A new Condition 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 Q3.4.5-2 Q3.4.5-3

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.5-3 **APPLICABILITY:** CA, WC

REQUEST: CTS 4.4.1.2.2, 4.4.1.3.2 and 3.4.1.4.1.b and ITS 3.4.5, 6 and 7 (Callaway and Wolf Creek)

Comment: Ten percent wide range level was specified as the necessary heat sink level. Now in the ITS the level is narrow range. Was this is a known error in the TS that is now being corrected or was this just discovered as part of the conversion effort? Please provide the technical basis for concluding that 10% (4% for Callaway) narrow range is adequate. Additionally explain why different narrow range level values are used at each plant and why wide range level is used in Mode 5 at one and not the other.

FLOG RESPONSE: The CTS Bases for Reactor Coolant Loops and Coolant Circulation is silent on the background behind the 10% wide range value specified. During the ITS conversion process, industry traveler TSTF-114 was developed (and subsequently approved by NRC), to recognize the importance of keeping the SG tubes covered as discussed in NRC IN 95-35, "Degraded Ability of SGs to Remove Decay Heat by Natural Circulation." Since 10% on the wide range level instrumentation does not ensure the SG tubes are fully covered, the values were changed in the conversion amendment. This was not an error known prior to starting the amendment development for ITS.

Plant Specific Discussion

The basis for using 10% narrow range for Wolf Creek was that NUREG-1431 provided a level Wolf Creek believed was conservative in relation to the CTS. After further review of this comment and Comment Number 3.4.5-2, Wolf Creek believes that it is appropriate to change the value to 6% narrow range since it is used throughout the Emergency Operating Procedures (EMGs), it has operator awareness because of the EMG familiarity, and ensures an SG water level approximately 100 inches above the top of the highest SG tube. Additionally, Wolf Creek believes that the wide range level instrumentation should be used for MODE 5 since it is calibrated for cold conditions and provides a larger span. Therefore, the ITS and Bases are revised to reflect the use on 6% narrow range and 66% wide range in MODE 5.

ATTACHED PAGES:

See attached pages in the response to Comment Number Q 3.4.5-2.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORM.. TION NO: Q 3.4.6-1 **APPLICABILITY:** DC, CP, WC, CA

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: Q 3.4.7-1 **APPLICABILITY:** WC

REQUEST: ITS 3.4.7.2 (Wolf Creek)

Comment: It should read "required SGs" rather than "required Sgs".

FLOG RESPONSE: The smooth copy of the ITS has been marked to read "required SGs." A final review of the smooth ITS and ITS Bases is planned prior to resubmitting to the NRC the smooth copy of the ITS and Bases

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.7-2 **APPLICABILITY:** WC

REQUEST: ITS LCO Bases 3.4.7 and 3.4.8 (Wolf Creek)

Comment: The TS condition "Loops Not Filled" should be defined in the TS Bases subject to the Bases Control Program and not in an unnamed plant procedure for which the control mechanism is not specified.

FLOG RESPONSE: During the development of the conversion application, the current Technical Specification Clarifications were reviewed to determine if the clarification should be incorporated into the ITS or ITS Bases. Wolf Creek had developed a Technical Specification Clarification to define "loops filled" and "loops not filled" and has subsequently incorporated these definitions into general operating procedures. The added text to the ITS Bases was to indicate that these definitions are contained in plant procedures. Wolf Creek is revising the ITS Bases to delete the reference to plant procedures. ITS Section 5.4.1 requires written procedures be established, implemented and maintained. Control mechanisms are in place to ensure procedure changes are reviewed and approved. Therefore, the Bases text is unnecessary and the WCGS ITS is now consistent with NUREG-1431, Rev. 1.

ATTACHED PAGES:

Encl. 5B B 3.4-35, B 3.4-38

BASES

Q3.4.7-2

APPLICABILITY

In MODE 5 with RCS loops filled ~~(as defined in plant procedures)~~, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side narrow range water level of at least two SGs is required to be ≥ 10%.

wide

66%

Q3.4.5-2
Q3.4.5-3

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
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).

ACTIONS

A.1 and A.2

66%

Q3.4.5-2
Q3.4.5-3

wide range

If one RHR loop is inoperable and the required SGs have secondary side water levels < 10%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Notes 1 and 4, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. Addition of borated water with a concentration greater than or equal to the minimum required RWST concentration but less than the actual RCS boron concentration shall not be considered a reduction in boron concentration. (Ref. 2). To prevent inadvertent criticality during a boron dilution, forced circulation from at least one RCP is required to provide proper mixing. ~~and preserve the margin to criticality in this type of operation.~~ The immediate Completion Times reflect the importance of maintaining operation for heat removal.

(continued)

BASES

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

Note 1 permits all RHR pumps to be ~~de-energized~~ removed from operation for ≤ 1 hour ~~15 minutes when switching from one loop to another~~. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet temperature is maintained at least 10°F below saturation temperature. ~~The Note prohibits boron dilution or draining operations when RHR forced flow is stopped.~~ The Note requires reactor vessel water level be above the vessel flange to ensure the operating RHR pump will not be intentionally deenergized during mid-loop operations.

redline Q 3.4.Gen-1

WC 3.4-001

Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

Q 3.4.7-2

APPLICABILITY

In MODE 5 with loops not filled ~~(as defined in plant procedures)~~, this LCO requires core heat removal and coolant circulation by the RHR System. One RHR loop provides sufficient capability for this purpose. However, one additional RHR loop is required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
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 3.9.5, "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)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.8-1 **APPLICABILITY:** DC, CP, WC, CA

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 9

CHANGE
NUMBER — JUSTIFICATION

CHANGE NUMBER	JUSTIFICATION
	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 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 ~~Condition A.~~ the Required Actions of this LCO. ~~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.~~ INSERT 6A-9b This change is consistent with traveler WOG-87. Q 3.4.11-4 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). INSERT 6A-9a Q 3.4.8-1
3.4-49	LCO 3.4.12, "[LTOP] System," provides four differenct methods for pressure relief. Any of the four 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 word "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 ~~WOG-100~~ TSTF-280. Q 3.4.12-1

INSERT 6A-9a Q 3.4.8-1

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.

INSERT 6A-9b Q 3.4.11-4

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.3 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 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.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.9-1 **APPLICABILITY:** DC, CP, WC, CA

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 SR 3.4.9.1 (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: Q 3.4.9-3 **APPLICABILITY:** DC, CP, 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.

For DCPP, the NRC recently approved (6/5/98) changing the CTS SR 4.4.3.2 from 92 day to "Refueling Interval" in L:A 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

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev ~~2~~ 3	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		Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. ~~1~~ 2	Incorporated	3.4-31	Approved by NRC. TR 3.4-004
TSTF-93 Rev. 3	Incorporated	3.4-17	Approved by NRC Q3.4.9-3
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. ~~3~~ 4	Incorporated	3.4-39	Q 3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev. ~~1~~ 2	Incorporated	3.4-36	Q3.4.13-2
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..
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
TSTF-285 ~~WOG-51, Rev. 1~~	Incorporated	~~3.4-23~~, 3.4-45, 3.4-52	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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	3.4-10	DCPP only Approved by NRC. 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	Q3.4.12-1

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.10-1 **APPLICABILITY:** DC, CP, WC, CA

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 USAR Chapter 15.

Plant Specific Discussion

In the Wolf Creek Applicable Safety Analysis Bases for ITS 3.4.10, changes were made to items b and e. The change to item b deleted the Loss of reactor coolant flow and added Feedwater line break as new item b since the analysis of this event shows the primary and secondary side safeties lift, as discussed in USAR Section 15.2.8 (see USAR Figures 15.2-14 and 15.2-19 for the pressurizer pressure and volume transients). The change to item e denotes that this event is not a station blackout; standby power is available from the diesel generators consistent with the discussion in USAR Section 15.2.6.

Further review of this section of the ITS Bases determined that additional changes are warranted. The Loss of reactor coolant flow accident is being retained in the Bases since analysis shows that primary and secondary side safeties will lift. Item c is being revised consistent with the discussion in USAR Section 15.2.2 and 15.2.3, i.e., no loss of external load analysis is presented in USAR Section 15.2 since the turbine trip is more limiting. RCCA ejection should be added as new item h since there is pressure surge analysis, discussed on page 15.4-35 of Section 15.4.8.2.2, that is incorporated by reference to WCAP-7588, Rev. 1-A, January 1975, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetic Methods," Risher, D. H., Jr. This pressure surge analysis concludes that with an ejection worth of one dollar at BOL and HFP conditions, the resulting stress levels do not exceed faulted stress limits.

All of the changes in this Bases section transform a generic discussion to one that applies to this plant specifically.

ATTACHED PAGES:

Encl. 5B B 3.4-47

BASES (continued)

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 UFSAR (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:

redline Q3.4.Gen-1

undelete

a. Uncontrolled rod withdrawal from full power;

Q3.4.10-1

b. ~~Loss of reactor coolant flow~~ Feedwater line break;

c. Loss of external electrical load/turbine trip;

Q3.4.10-1

d. Loss of normal feedwater;

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

f. Locked rotor.

g. Loss of reactor coolant flow

h. Rod cluster control assembly ejection

Q3.4.10-1

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in ~~events c, d, and e (above)~~ the above events 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.~~ 10 CFR 50.36(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.

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.11-1 **APPLICABILITY:** DC, CP, WC, CA

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 currently credited 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: Q 3.4.11-2 **APPLICABILITY:** DC, CP, WC, CA

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 SR Bases Note change). 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 3.4-58
Encl. 6A	7

CHANGE NUMBER	NSHC	DESCRIPTION
4-08	LS-34	TSTF-288 -- Q 3.4.11-2 Consistent with traveler WOG-60 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.
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. This change is consistent with traveler WOG-87. INSERT 3A-8a — Q 3.4.11-4
5-01	A	This change moves the steam generator tube surveillances to ITS SR 3.4.13.2 and Administrative Controls Sections 5.5.9, "Steam Generator (SG) Tube Surveillance Program" and 5.6.10, "Steam Generator Tube Inspection Report."
5-02	A	LCO 3.4.5 is deleted for consistency with NUREG-1431 Rev. 1. Steam generator operability requirements in MODES 1-4 are specified in the RCS loop and operational leakage specifications.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.4

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 FSAR.	Yes- Moved to TRM.	Yes- Moved to USAR.	Yes-Moved to FSAR.
4-05 LS-31	The shutdown requirement of CTS 3.4.4 would require the plant 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 [3.4.8] 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 WOG-60, TSTF-288 the requirement to perform the PORV and block valve cycling surveillances is revised such that the surveillance is only required to be performed in MODES 1 and 2.	Yes Q 3.4.11-2	Yes	Yes	Yes
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
5-01 A	This change moves the Steam Generator Tube Surveillances to ITS 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

Q 3.4.11-4

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev ~~1~~ 3	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		Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. ~~1~~ 2	Incorporated	3.4-31	Approved by NRC. TR 3.4-004
TSTF-93 Rev. 3	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. ~~3~~ 4	Incorporated	3.4-39	Q 3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev. ~~1~~ 2	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..
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
TSTF-285 ~~WOG-51, Rev. 1~~	Incorporated	~~3.4-23~~, 3.4-45, 3.4-52	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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	3.4-10	DCPP only Approved by NRC. 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

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1 (continued)

Q3.4.11-4

~~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)~~.

This SR is modified by two Notes. Q3.4.11-2

~~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 Actions of 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. Q3.4.11-4 Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2.

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.~~ WC3.4-007 Operating experience has shown that these valves usually pass the Surveillance when performed at the required Inservice Testing Program frequency. The Frequency is acceptable from a reliability standpoint. The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. In accordance with Reference 5, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.

~~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.~~

(continued)

CHANGE
NUMBER

JUSTIFICATION

purposes (per Bases). This allowance is properly presented as an SR Note. A properly placed exception (i.e., an SR Noted 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 ~~WOG-50~~ TSTF-288 Q3.4.11-2

3.4-36 SR 3.4.13.1 and LCO 3.4.15 are revised per traveler TSTF-116. The note addresses the concern that an RCS water inventory balance connot 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 WCGS. See Conversion Comparison Table (Enclosure 3B).

Q 3.4.1-1

3.4-38 ~~Consistent with TSTF-105, the details on the method by which the RCS flow rate are verified are moved from 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 WCGS. See Conversion Comparison Table (Enclosure 3B)

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. INSERT 6A-7a Q3.4.11-6 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.

3.4-40 Consistent with traveler ~~WOG-99~~ TSTF-282, the Note to SR 3.4.1.4 Q3.4.1-2 would be modified to provide additional time to perform an RCS precision flow rate measurement. The time allowed would be changed from 24 hours to 7 days. This change is acceptable because other indication of RCS flow is available (SR 3.4.1.3, RCS total flow meters) and additional time normally would be required to establish

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.11-3 **APPLICABILITY:** DC, CP, WC, CA

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 <500°F 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 Caliaway 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 page

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev ~~2~~ 3	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		Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. ~~1~~ 2	Incorporated	3.4-31	Approved by NRC. TR 3.4-004
TSTF-93 Rev. 3	Incorporated	3.4-17	Approved by NRC Q3.4.9-3
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. ~~3~~ 4	Incorporated	3.4-39	Q 3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev. ~~1~~ 2	Incorporated	3.4-36	Q3.4.13-2
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..
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
TSTF-285 ~~WOG-51, Rev. 1~~	Incorporated	~~3.4-23~~, 3.4-45, 3.4-52	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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	3.4-10	DCPP only Approved by NRC. 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	Q3.4.12-1

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.11-4 **APPLICABILITY:** DC, CP, WC, CA

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	4-10
Encl. 3A	8
Encl. 3B	5
Encl. 4	68, 69
Encl. 5A	Traveler Status page, 3.4-25, 3.4-26, 3.4-28
Encl. 5B	B 3.4-55, B 3.4-57, B 3.4-58
Encl. 6A	9
Encl. 6B	7

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-5

ACTION:

a. With one or both PORVs 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, **continue restoration activities and** 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-6 4-05-LS-31

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 **continue restoration activities 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-6 4-05-LS-31 4-03-M

c. With both PORVs 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 **continue restoration activities 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-6 4-05-LS-31 4-03-M

Q3.4-11-4

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, **continue restoration activities and** 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-31

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 ~~In addition to the requirements of~~ **No additional requirements other than those required by** Specification 4.0.5.#, ~~each PORV shall be demonstrated OPERABLE at least once per 18 months by performing a CHANNEL CALIBRATION of the actuation instrumentation.~~ 4-04-LG 4-08-LS-34

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days# by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION **a.,** b. or c. in Specification 3.4.4. 4-08-LS-34 4-09-LS-36

---

*With all RCS cold leg temperatures above 368°F.

 # Only required to be performed in MODES 1 and 2. 4-08-LS-34

** Action d. does not apply when the block valve(s) is inoperable solely as a result of complying with Actions b or c. 4-09-LS-36 Q3.4.11-4

CHANGE NUMBER	NSHC	DESCRIPTION
4-08	LS-34	TSTF-288 — Q 3.4.11-2 Consistent with traveler ~~WOG-60~~, 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.
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. This change is consistent with traveler WOG-87. INSERT 3A-8a — Q 3.4.11-4
5-01	A	This change moves the steam generator tube surveillances to ITS SR 3.4.13.2 and Administrative Controls Sections 5.5.9, "Steam Generator (SG) Tube Surveillance Program" and 5.6.10, "Steam Generator Tube Inspection Report."
5-02	A	LCO 3.4.5 is deleted for consistency with NUREG-1431 Rev. 1. Steam generator operability requirements in MODES 1-4 are specified in the RCS loop and operational leakage specifications.

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 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).

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.4

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 FSAR.	Yes- Moved to TRM.	Yes- Moved to USAR.	Yes-Moved to FSAR.
4-05 LS-31	The shutdown requirement of CTS 3.4.4 would require the plant 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 [3.4.8] 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 WOG-60, the requirement to perform the PORV and block valve cycling surveillances is revised such that the surveillance is only required to be performed in MODES 1 and 2.	Yes	Yes	Yes	Yes
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.	Yes	Yes	Yes	Yes
5-01 A	This change moves the Steam Generator Tube Surveillances to ITS 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

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

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] Q3.4.11-4
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

V. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS-36
(continued)

block valves are capable of opening against 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. INSERT LS-36 Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. Q3.4.11-4

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

The only acciidents 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.

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.

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev ~~2~~ 3	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.
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~~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. ~~3~~ 4	Incorporated	3.4-39	Q 3.4.11-3
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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..
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TSTF-153	Incorporated	3.4-01	Approved by NRC. TR 3.4-009
TSTF-162	Incorporated	NA	Approved by NRC. TR 3.4-006
TSTF-285 ~~WOG-51, Rev. 1~~	Incorporated	~~3.4-23~~, 3.4-45, 3.4-52	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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	3.4-10	DCPP only Approved by NRC. 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

5/15/97

- - - - - 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.
- - - - - - - - - - - - -

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
C. One block valve inoperable.	C.1	Place associated PORV in manual control.	1 hour	3.4-47 Q3.4.11-4
	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	3.4-39
	AND			
	D.~~1~~2	Be in MODE 3.	6 hours	
	AND			
	D.~~1~~3	~~Be in MODE 4~~. Reduce T_{avg} to <500°F.	12 hours	

(continued)

ACTIONS (continued)

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
	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
F. More than one block valve inoperable.	F.1	Place associated PORVs in manual control.	1 hour
	AND		
			(continued)

3.4-39
B-PS
3.4-39

------ 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

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 Actions of ~~Condition A B or E~~ this LCO. 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 Condition B or E.~~	92 days	3.4-47 Q3.4.11-4 3.4-35 3.4-47 Q3.4.11-4
SR 3.4.11.2	------------------NOTE------------------ Only required to be performed in MODES 1 and 2. -- Perform a complete cycle of each PORV.	~~[18 months]~~ In accordance with the Inservice Testing Program	3.4-53 3.4-35 B WG 3.4-007
~~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
~~SR 3.4.11.4~~	~~Verify PORVs and block valves are capable of being powered from emergency power sources.~~	~~[18] months~~	B-PS

BASES

ACTIONS (continued)

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

~~one~~ s Q 3.4.Gen-1

If one PORV 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 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 not~~ may not be capable of mitigating an ~~overpressure~~ event ~~when placed in manual control~~ if the inoperable block valve is not full open. If the block valve is restored within the Completion Time of 72 hours, the power will be restored ~~and~~ to 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.

INSERT B 3.4-5 Q 3.4.11-4

(continued)

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.

ACTIONS (continued)

5 and 6 (with the reactor vessel head on) ~~maintaining~~ automatic PORV OPERABILITY may be required. See LCO 3.4.12.

F.1. and F.2~~, and F.3~~

If more than one block valve is inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control and restore at least one block valve within 2 hours ~~and restore the remaining block valve within 72 hours~~. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

INSERT B 3.4-57 Q3.4.11-4

G.1, ~~and~~ G.2 and G.3

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 within 6 hours and ~~to MODE 4~~ T_{avg} must be reduced to <500°F within 12 hours. Additional action is required to be initiated immediately to continue efforts to restore the inoperable block valve(s) to OPERABLE status. This will ensure expedient measures are taken to re-establish OPERABLE 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~~, 3 (with any RCS cold leg temperature ≤ 368°F), 4, 5 and 6 (with the reactor vessel head on) ~~maintaining~~ automatic 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 ~~if needed~~ ~~except when the block valve(s) is closed in accordance with the Required Actions of Condition B or E~~. The basis for the Frequency of 92 days is the ASME Code, Section XI (Ref. ~~4~~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; however, demonstrating block valve OPERABILITY on a quarterly basis via valve cycling is not required. If the block valve is closed to isolate an~~ ~~otherwise~~ ~~inoperable PORV~~ Q3.4.11-4

(continued)

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.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1 (continued)

Q3.4.11-4

~~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)~~ This SR is modified by two Notes. Q3.4.11-2

~~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 Actions of 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. Q3.4.11-4 Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2.

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.~~ WC3.4-007 Operating experience has shown that these valves usually pass the Surveillance when performed at the required Inservice Testing Program frequency. The Frequency is acceptable from a reliability standpoint. The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. In accordance with Reference 5, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.

~~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.~~

(continued)

CHANGE NUMBER	JUSTIFICATION
	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 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 ~~Condition A. 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.~~ the Required Actions of this LCO. Q 3.4.11-4 This change is consistent with traveler WOG-87. INSERT 6A-9b 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). INSERT 6A-9a Q 3.4.8-1
3.4-49	LCO 3.4.12, "[LTOP] System," provides four differenct methods for pressure relief. Any of the four 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 word "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 ~~WOG-100~~ TSTF-280. Q 3.4.12-1

INSERT 6A-9a Q 3.4.8-1

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.

INSERT 6A-9b Q 3.4.11-4

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 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.

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-45	~~ITS 3.4.12 has been revised to move the Note for Required Action B.1 regarding CCP pump swap operations and the Applicability Note for accumulator isolation to the LCO, as discussed in traveler WOG-51, Rev. 1. Plant-specific time allowances for exceeding the LCO's number of [ECCS] pumps capable of injecting into the RCS are incorporated, [as discussed in CN 3.4-18].~~ ~~TSTF-285~~ INSERT 6B-7a	Yes	No - Operation of 2 CCPs are allowed per CTS.	Yes	Yes Q3.4.12-2
3.4-46	ITS LCO 3.4.2 and its Condition A and SR 3.4.2.1 are modified to refer to "operating" RCS loops.	Yes	Yes	Yes	Yes
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 an F to prevent entry solely due to inoperable PORV(s) under Condition B or E.	Yes	Yes	Yes	Yes Q3.4.11-4
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

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.11-5 **APPLICABILITY:** WC

REQUEST: ITS Bases 3.4.11 Background (Wolf Creek)

Comment: On the top of smooth Bases Page 3.4-55 the sentence beginning "The functional design..." should not end with "... Pressurizer." It should include the phrase that comprises the next paragraph.

FLOG RESPONSE: The smooth copy of the ITS has been marked to indicate the referenced text as one paragraph. A final review of the smooth ITS and ITS Bases is planned prior to resubmitting to the NRC the smooth copy of the ITS and Bases.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.11-6 **APPLICABILITY:** CP, WC, CA

REQUEST: Difference 3.4-49 (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. JFD 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

purposes (per Bases). This allowance is properly presented as an SR Note. A properly placed exception (i.e., an SR Noted 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 ~~WOG-50~~ TSTF-288 Q3.4.11-2

3.4-36 SR 3.4.13.1 and LCO 3.4.15 are revised per traveler 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 WCGS See Conversion Comparison Table (Enclosure 3B).

3.4-38 Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B) Q3.4.1-1

~~Consistent with TSTF-105, the details on the method by which the RCS flow rate are verified are moved from 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.~~

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. INSERT 6A-7a Q3.4.11-6 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.

3.4-40 Consistent with traveler ~~WOG-99~~ TSTF-282, the Note to SR 3.4.1.4 Q3.4.1-2 would be modified to provide additional time to perform an RCS precision flow rate measurement. The time allowed would be changed from 24 hours to 7 days. This change is acceptable because other indication of RCS flow is available (SR 3.4.1.3, RCS total flow meters) and additional time normally would be required to establish

New initial Required Actions are added to Conditions D, E and G to immediately initiate 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.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.12-1 **APPLICABILITY:** DC, CP, WC, CA

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, 3.4-29, 3.4-33
Encl. 6A 9

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev ~~2~~ 3	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		Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. ~~1~~ 2	Incorporated	3.4-31	Approved by NRC. TR 3.4-004
TSTF-93 Rev. 3	Incorporated	3.4-17	Approved by NRC Q3.4.9-3
TSTF-94 Rev. 1	Not Incorporated	NA	Retained current TS. TR 3.4-005
~~TSTF-105~~	~~Incorporated~~	~~3.4-28~~	Q 3.4.1-1
TSTF-108, Rev. 1	Not Incorporated	NA	LCO 3.4.19 does not apply.
TSTF-113, Rev. ~~3~~ 4	Incorporated	3.4-39	Q 3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev. ~~1~~ 2	Incorporated	3.4-36	Q3.4.13-2
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..
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
TSTF-285 ~~WOG-51, Rev. 1~~	Incorporated	~~3.4-23~~, 3.4-45, 3.4-52	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
TSTF-233 ~~WOG-67, Rev 1~~	Incorporated	3.4-10	DCPP only Approved by NRC. TR 3.4-009
WOG-87, Rev. 2	Incorporated	3.4-47	Q 3.4.11-4
~~WOG-89~~ 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 centrifugal charging pump 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 Q3.4.12-1 3.4-49 B

~~b~~d. The RCS depressurized and an RCS vent of ≥ ~~2.07~~ 2.0 square inches. B-PS

made Q3.4.12-2

NOTES

1. Two centrifugal charging pumps may be made capable of ~~injection into the RCS for up to~~ injecting for ≤ 4 hours for pump swap operation. Q3.4.12-2 3.4-45 3.4-18

2. Two safety injection pumps and two centrifugal charging pumps may be capable of injecting into the RCS: (a) in MODE 3 with any RCS cold leg temperature ≤368°F and ECCS pumps OPERABLE pursuant to LCO 3.5.2, "ECCS - Operating," and (b) for up to 4 hours after entering MODE 4 from MODE 3 or until the temperature of one or more RCS cold leg decreases below 325°F, whichever comes first. 3.4-18

3. One or more safety injection pumps may be capable of injecting into the RCS in MODES 5 and 6 when the RCS water level is below the top of the reactor vessel flange for the purpose of protecting the decay heat removal function. 3.4-20

~~1~~4. Accumulator ~~isolation is only required~~ may be unisolated when accumulator pressure is ~~greater than or equal to~~ less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. 3.4-45 Q3.4.12-2

SURVEILLANCE REQUIREMENTS (continued)

	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 Q 3.4.12-3	3.4-08 ~~3.4-09~~ B-PS
SR 3.4.12.5	~~-----NOTE-----~~ ~~Only required to be performed when complying with LCO 3.4.12.b d.~~ ~~----------~~ Verify required RCS vent ≥2.0 ~~2.07~~ square inches open.	Q 3.4.12-1 12 hours for ~~unlocked open~~ vent ~~valve(s)~~ pathway(s) not locked, sealed or otherwise secured in the open position AND 31 days for ~~locked open~~ vent valve(s) locked, sealed or otherwise secured in the open position	3.4-49 B-PS 3.4-50 3.4-50
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~~	~~3.4-08~~ B-PS Q 3.4.12-3

CHANGE
NUMBER | JUSTIFICATION

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 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 ~~Condition A.~~ the Required Actions of this LCO. ~~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.~~ This change is consistent with traveler WOG-87. INSERT 6A-4b Q 3.4.11-4

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). INSERT 6A-9a Q 3.4.8-1

3.4-49 LCO 3.4.12, "[LTOP] System," provides four differenct methods for pressure relief. Any of the four 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 word "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 ~~WOG-100~~ TSTF-280. Q 3.4.12-1

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.12-2 **APPLICABILITY:** DC, CP, WC, CA

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 has been approved by the TSTF and is designated as TSTF-285. This traveler has been submitted to the NRC and is 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. 2	4-34, 4-34 Insert page
Encl. 5A	Traveler Status page, 3.4-29
Encl. 5B	B 3.4-67, B 3.4-68
Encl. 6A	8
Encl. 6B	7, 8

3.4.9.3 At least one of the following groups of two overpressure protection devices shall be OPERABLE **with a maximum of zero safety injection pumps and one centrifugal charging pump capable of injecting into the RCS and the accumulators isolated or depressurized below allowed RCS pressure per the PTLR** when the Reactor Coolant System (RCS) is not depressurized through a 2 square inch or larger vent: **9-06-M** **9-15-M**

a. Two residual heat removal (RHR) suction relief valves with Setpoints of 450 ± 3%, or

b. Two power-operated relief valves (PORV) with Setpoints which do not exceed the limits established in **the PTLR,** ~~Figure 3.4-4,~~ or **9-01-LG**

c. One RHR suction relief valve and one PORV with Setpoints as prescribed above

Q 3.4.12-2

APPLICABILITY: MODE 3 when the temperature of any RCS cold leg is less than or equal to 368°F, MODE 4, MODE 5, and MODE 6 when the head is on the Reactor Vessel.

INSERT 3.4.9.1-1 **9-17-LS-24**

ACTION:

a. With one of the two required overpressure protection devices inoperable in MODE 3 or 4, restore two overpressure protection devices to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.

b. With one of the two required overpressure protection devices inoperable in MODES 5 or 6, restore two overpressure protection devices to OPERABLE status within 24 hours, or complete depressurization and venting of the RCS through at least a 2 square inch vent within the next 8 hours.

c. With both of the two required overpressure protection devices inoperable, complete depressurization and venting of the RCS through at least a 2 square inch vent within 8 hours.

~~d. In the event either the PORVs, or the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.~~ **9-07-TR-2**

e. The provisions of 3.0.4 are not applicable.

(NEW) With one or more safety injection pumps or more than one centrifugal charging pump capable of injecting into the RCS, immediately initiate action to verify a maximum of zero safety injection pumps and a maximum of one centrifugal charging pump is capable of injecting into the RCS or depressurize and vent the RCS with an RCS vent of ≥2.0 square inches within the next 8 hours. **9-15-M**

(NEW) With an accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing cold leg temperature allowed in the PTLR, isolate the affected accumulator within 1 hour or, within the next 12 hours, either increase all RCS cold leg temperature to >368°F or depressurize the affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allwed in the PTLR. **9-10-M**

(NEW) With the LTOP inoperable for any other reason, within 8 hours depressurize the RCS and establish RCS vent of ≥2.0 square inches. **9-11-M**

INSERT 3.4.9.3-1

Q 3.4.12-2

9-17-LS-24

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

1. Two centrifugal charging pumps may be made capable of injecting ~~into the RCS up to~~ for ≤ 4 hours for pump swap operations.

2. Two safety injection pumps and two centrifugal charging pumps may be capable of injecting into the RCS:

   (a) In MODE 3 with any RCS cold leg temperature < 368° F and ECCS pumps OPERABLE pursuant to LCO 3.5.2, "ECCS-Operating", and

   (b) For up to 4 hours after entering MODE 4 from MODE 3 or until the temperature of one or more RCS cold legs decreases below 325° F, whichever comes first.

3. One or more safety injection pumps may be capable of injecting into the RCS in MODES 5 and 6 when the RCS water level is below the top of the reactor vessel flange for the purpose of protecting the decay heat removal function.

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

Q 3.4.12-2

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev ~~1~~ 3	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		Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. ~~1~~ 2	Incorporated	3.4-31	Approved by NRC. TR 3.4-004
TSTF-93 Rev. 3	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. ~~3~~ 4	Incorporated	3.4-39	Q 3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev. ~~1~~ 2	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..
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
TSTF-285 ~~WOG-51, Rev. 1~~	Incorporated	~~3.4-23~~, 3.4-45, 3.4-52	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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	3.4-10	DCPP only Approved by NRC. 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 centrifugal charging pump 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 Q3.4.12-1 3.4-49 B

~~b~~d. The RCS depressurized and an RCS vent of ≥ ~~2.07~~ 2.0 square inches. B-PS

NOTES

1. Two centrifugal charging pumps may be made capable of ~~injection into the RCS for up to~~ injecting for ≤ 4 hours for pump swap operation. Q3.4.12-2 3.4-45 3.4-18

2. Two safety injection pumps and two centrifugal charging pumps may be capable of injecting into the RCS: (a) in MODE 3 with any RCS cold leg temperature ≤368°F and ECCS pumps OPERABLE pursuant to LCO 3.5.2, "ECCS - Operating," and (b) for up to 4 hours after entering MODE 4 from MODE 3 or until the temperature of one or more RCS cold leg decreases below 325°F, whichever comes first. 3.4-18

3. One or more safety injection pumps may be capable of injecting into the RCS in MODES 5 and 6 when the RCS water level is below the top of the reactor vessel flange for the purpose of protecting the decay heat removal function. 3.4-20

~~1~~4. Accumulator ~~isolation is only required~~ may be unisolated when accumulator pressure is ~~greater than or equal to~~ less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. 3.4-45 Q3.4.12-2

BASES

APPLICABLE SAFETY ANALYSIS (continued)

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of ~~2.07~~ 2.0 square inches is capable of mitigating the ~~allowed LTOP~~ limiting LTOP ~~overpressure~~ transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, ~~two~~ one centrifugal charging pump~~s~~ OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the ~~minimum~~ maximum coolant input or heat input bounded by that assumed in the analyses and required 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 that a maximum of zero safety injection pumps and ~~two~~ one centrifugal charging pump~~s~~ 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. Q3.4.12-2

The LCO is modified by four Notes. Note 1 allows two centrifugal charging pumps to be capable of injecting into the RCS for < 4 hours for pump swap operations. This provides the necessary allowance to perform the pump swap activities in a controlled manner and provides sufficient time to complete the activities necessary to restore a maximum of one centrifugal charging pump to a status capable of injecting into the RCS. This is accomplished by racking out the breaker for one pump or employing Q3.4.12-2

(continued)

BASES

LCO (continued)

two independent means to prevent a pump start in accordance with SR 3.4.12.2.

Note 2 recognizes the Applicability overlap between LCO's 3.4.12 and 3.5.2 and states that two safety injection pumps and two centrifugal charging pumps may be capable of injecting into the RCS:

redline Q3.4.Gen-1

(a) In MODE 3 with any RCS cold leg temperature < 368° F and ECCS pumps OPERABLE pursuant to LCO 3.5.2, "ECCS-Operating", and

(b) For up to 4 hours after entering MODE 4 from MODE 3 or the temperature of one or more RCS cold legs decreases below 325°F, whichever comes first.

Note 3 states that one or more safety injection pumps may becapable of injecting into the RCS in MODES 5 and 6 when the RCS water level is below the top of the reactor vessel flange for the purpose of protecting the decay heat removal function.

the — may be unisolated — Q3.4.12-2

Note 4 states that accumulator ~~isolation is only required~~ when the accumulator pressure is ~~greater than or equal to~~ the maximum RCS pressure for the existing RCS cold leg temperature as allowed by the P/T limit curves provided in the PTLR. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

less than

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

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

~~1~~a. 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.

(bracketed text)

redline Q3.4.Gen-1

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

(continued)

CHANGE
NUMBER — JUSTIFICATION

plant conditions suitable for the precision heat balance. Since this parameter does not normally change significantly and the flow meters can be used in the interim, there is no need to perform this SR within the 24 hour period specified in NUREG-1431 Rev. 1. The 7 day period provides sufficient time to establish steady state plant thermohydraulic conditions and obtain equilibrium xenon. In addition, the THERMAL POWER specified in the Note would be changed from the generic value in brackets (90 % RTP) to 95 % RTP. This change is acceptable because it specifies a power level in better agreement with current operating procedures for performing a precision heat balance. Current TS do not specify a power level for this measurement.

3.4-41 — Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).

3.4-42 — Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).

3.4-43 — A new Condition is added to LCO 3.4.1 to reflect the current licensing basis of Wolf Creek for RCS flow rate. License Amendment 61 approved revisions to incorporate the provisions of the RCS flow TS entitled "RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR" into the "DNB PARAMETERS" specification. These changes were made to support the use of VANTAGE 5H fuel with the Intermediate Flow Mixer grid feature. This amendment also approved operation at an increased power level.

Q3.4.5-2
Q3.4.5-3

3.4-44 — ~~Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).~~ INSERT 6A-8b

Q3.4.12-2

3.4-45 — ~~ITS 3.4.12 has been revised to move the Note for Required Action B.1 regarding CCP pump swap operations and the Applicability Note for accumulator isolation to the LCO, as discussed in traveler WOG-51, Rev. 1. Plant-specific time allowances for exceeding the LCO's number of [ECCS] pumps capable of injecting into the RCS are incorporated[, as discussed in CN 3.4-18]. These Notes detail situations where exceptions to the LCO are permitted and are more appropriately annotated under the LCO.~~ INSERT 6A-8a

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

INSERT 6A-8a

Q 3.4.12-2

Consistent with traveler TSTF-285, ITS 3.4.12 has been revised to move the Note for Required Action B.1 regarding CCP pump swap operations and the Note under Applicability regarding accumulator isolation to the LCO. These Notes have been reworded for clarity and detail situations where exceptions to the LCO are permitted. Also, plant-specific time allowances for exceeding the LCO number of [ECCS] pumps capable of injecting into the RCS are incorporated [, as discussed in CN 3.4-18].

INSERT 6A-8b

Q 3.4.5-2
Q 3.4.5-3

Steam generator levels for MODE 3, 4, and 5 are specified to ensure SG tubes are covered. The current TS did not ensure tube coverage.

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
3.4-45	~~ITS 3.4.12 has been revised to move the Note for Required Action B.1 regarding CCP pump swap operations and the Applicability Note for accumulator isolation to the LCO, as discussed in traveler WOG-51, Rev. 1. Plant-specific time allowances for exceeding the LCO's number of [ECCS] pumps capable of injecting into the RCS are incorporated. [as discussed in CN 3.4-18]~~. ~~TSTF-285~~ INSERT 6B-7a	Yes	No - Operation of 2 CCPs are allowed per CTS.	Yes	Yes Q3.4.12-2
3.4-46	ITS LCO 3.4.2 and its Condition A and SR 3.4.2.1 are modified to refer to "operating" RCS loops.	Yes	Yes	Yes	Yes
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 an F to prevent entry solely due to inoperable PORV(s) under Condition B or E.	Yes	Yes	Yes	Yes Q3.4.11-4
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

INSERT 6B-7a

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-45	Consistent with traveler TSTF-285, ITS 3.4.12 has been revised to move the Note for Required Action B.1 regarding CCP pump swap operations and the Note under Applicability regarding accumulator isolation to the LCO. Also, plant-specific time allowances for exceeding the LCO number of [ECCS] pumps capable of injecting into the RCS are incorporated [, as discussed in CN 3.4-18].				

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-50	This change is consistent with current TS SR 4.4.9.3.3. The 12 hour frequency applies to vent pathways 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 flow rate verification on the 18 month frequency. DC ALL-005	Yes	No	No	No
3.4-52	Consistent with traveler ~~OOG-51~~ TSTF-285, 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. Q3.4.12-2

3.4-53 INSERT 6B-8a WC 3.4-007

3.4-54 INSERT 6B-8b DC 3.4-003

INSERT 6B-8a

WC 3.4-007

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-53	This change revises the ITS SR 3.4.11.2 Frequency from "18 months" to "In accordance with the Inservice Testing Program." The CTS for this surveillance establishes the frequency as being per the IST Program [CTS SR 4.4.4.1].	No	No	Yes	Yes

INSERT 6B-8b

DC 3.4-003

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-54	Consistent with the current license bases as approved LA 124/122, ITS LCO 3.4.13 is revised to reflect reduced steam generator primary-to-secondary leakage limits of 150 gallons per day from any one steam generator and an additional surveillance Requirement to determine primary-to-secondary leakage every 72 hours.	Yes	No	No	No

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.12-3 **APPLICABILITY:** DC, CP, WC, CA

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 locked open every 31 days (when the RHR relief valves are being used for 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 Wolf Creek in Amendment No. 49 dated September 12, 1991.

ATTACHED PAGES:

Encl. 5A	3.4-33
Encl. 6A	2
Encl. 6B	2

SURVEILLANCE REQUIREMENTS (continued)

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 Q 3.4.12-3	3.4-08 ~~3.4-09~~ B-PS
SR 3.4.12.5	~~NOTE~~ ~~Only required to be performed when complying with LCO 3.4.12.b d.~~ Verify required RCS vent ≥2.0 ~~2.07~~ square inches open.	Q 3.4.12-1 12 hours for ~~unlocked open~~ vent ~~valve(s)~~ pathway(s) not locked, sealed or otherwise secured in the open position <u>AND</u> 31 days for ~~locked open~~ vent valve(s) locked, sealed or otherwise secured in the open position	3.4-49 B-PS 3.4-50 3.4-50
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~~	~~3.4-08~~ B-PS Q 3.4.12-3

CHANGE NUMBER	JUSTIFICATION
3.4-05	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
3.4-06	These changes are consistent with the current plant specific analysis and are reflective of the current Technical Specifications. The plant specific analysis does not allow injection from any safety injection pumps and does allow injection from [one] centrifugal charging pump. The changes are made consistently in the LCO, ACTIONS and SURVEILLANCE REQUIREMENTS.
3.4-07	Not applicable to WCGS. 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.
	Q 3.4.12-3
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.
3.4-10	Not applicable to WCGS. 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 to 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 consistent with LCO 3.4.12.. SR 3.4.14.3 is not used.

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
3.4-08	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
3.4-10	The DCPP plant specific limiting temperature specified in degrees below which the RCS must not be subject to low temperature overpressure is replaced by the generic statement "the temperature below which LTOP is required as specified in the PTLR."	Yes	No	No	No
3.4-11	The plant does not have the RHR autoclosure portion of the RHR System interlock as the system was deleted from the 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.	No - The valve interlock is not in the current TS	Yes	Yes - See Amendment No. 49.	Yes - See OL Amendment No. 42.
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.	No - RHR valve testing after MODE 5 operation is not in CTS.	Yes	No - WCNOC does not have this exclusion.	No - Callaway does not have this exclusion.
3.4-13	The RHR Isolation Valves which are RCS PIVs are excluded from being retested following flow through the valves.	No - This change is out of scope for DCPP.	Yes	Yes	Yes

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.12-4 **APPLICABILITY:** WC, CA, CP

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:

Encl. 5B B 3.4-69

BASES

LCO (continued)

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.

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

Q3.4.Gen-1

redline (bracketed text)

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

An RCS vent is OPERABLE when open with an area of ≥ ~~2.07~~ 2.0 square inches.

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

APPLICABILITY

This LCO is applicable in MODE 3 when ~~the temperature of~~ any RCS cold leg temperature is ≤ 368°F, 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 in MODES 1, 2, and 3 ~~above 275°F~~. When the reactor vessel head is off, overpressurization cannot occur. ~~With fuel off loaded, the reactor vessel head may be placed on the vessel for radiological conditions but not bolted. Overpressure protection is maintained because sources for thermally induced overpressure are not available and the reactor vessel head will lift and relieve at low pressure if hydraulically induced pressure is present.~~

Q3.4.12-4

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 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.

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.12-5 **APPLICABILITY:** WC, CA

REQUEST: Differences 3.4-18 and 3.4-45 (Wolf Creek and Callaway)

Comment: The justification for the 4-hour pump swap is inadequate. The STS allows 15 minutes. The CTS is used as justification however, finding a pump inoperable and then restoring it (which is the case covered by the CTS) is very different than simply switching from one operable pump to another.

FLOG RESPONSE: Four hours is a reasonable time restriction for swapping centrifugal charging pumps (CCP) during the low temperature overpressure protection (LTOP)/cold overpressure mitigation system (COMS) Applicability.

Current Technical Specification (CTS) 3/4.4.9 Bases state "Operation below 350F but greater than 325F with all centrifugal charging and Safety Injection pumps OPERABLE is allowed for up to 4 hours. ..... Given the short time duration that this condition is allowed and the low probability of a single failure causing an overpressure event during this time, the single failure of a PORV is not assumed. Initiation of both trains of Safety Injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents." Additionally, CTS 3.5.4 requires all Safety Injection pumps and one CCP to be inoperable. If this requirement is not met, then four hours is allowed to return the pump(s) to an inoperable status.

Performing CCP swap operations for maintenance activities requires both pumps to be capable of injecting for a limited period of time. During the time allowed for pump swap operation, the inoperable CCP must first be restored to OPERABLE status to meet ITS LCO 3.5.3 (MODE 4) and USAR/FSAR Section 16.1.2.3 (one OPERABLE CCP in boration flow path, MODES 4-6). Then the other CCP must be rendered capable of injecting. In order to render the other CCP incapable of injecting into the RCS, the requirements of ITS SR 3.4.12.2 must be met. SR 3.4.12.2 Bases states that a pump is rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative controls. The Bases also state that an alternate method of cold overpressure protection control may be employed using at least two independent means to render a pump incapable of injecting. Each method includes local actions (e.g., breaker racked out and tagged, valve closed and tagged). These actions for restoring the one CCP and then rendering the other CCP incapable of injecting into the RCS cannot be performed from the control room. Swapping of CCP trains is a short duration evolution but must be performed in a controlled manner especially when coordinating activities outside the control room. The 4 hour time allowance provides for normal operation of the plant and allows plant manipulations/evolutions to be performed in a time frame in which they can be safely performed.

Amendment No. 103 (Callaway) and Amendment No. 89 (Wolf Creek) revised current TS 3.5.4 to provide a 4 hour AOT to restore one CCP to an inoperable status in MODES 5 and 6. This 4 hour AOT was specifically reviewed and approved by the NRC as noted in their safety evaluations for those license amendments. This portion of the COMS/LTOP Applicability is the most limiting, as it may involve water solid operation. Current TS 3.5.3 (SR 4.5.3.2) allows 4 hours to secure one CCP after entering MODE 4 from MODE 3. Current TS 3.5.2 requires both CCPs to be operable in MODE 3. Therefore, all of the ITS 3.4.12 Applicability is based on the current TS except for MODE 4 beyond 4 hours after entry from MODE 3. NSHC LS-24 justifies 4 hours for all of MODE 4.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.13-1 **APPLICABILITY:** DC, WC

REQUEST: Change 6-25 LS-26 (Diablo Canyon and Wolf Creek)

Comment: The change discussion is not adequate. The NSHC contains the necessary justification.

FLOG RESPONSE: DOC 6-15 LS-26 has been revised to incorporate additional justification from NSHC LS-26 from Enclosure 4. DOC 6-25 LS-26 has been revised to include: "The RCS is isolated from other systems by valves. During plant life these interfaces can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. Increasing allowed leakage limits from 1 gpm up to 5 gpm for the pressure isolation valves will not challenge the pressure relief capacity of interfacing systems. This amount of leakage is considered negligible when compared with the capacity of the pressure relief valves. Pressure isolation valve leakage limits apply to leakage rates for individual valves.

The basis for this LCO is the 1975 Reactor Safety Study (NUREG-75/014)) which identified potential intersystem Loss of Coolant Accidents (LOCAs) as a significant contributor to the risk of core melt. A subsequent study (NUREG-0677) evaluated various pressure isolation valve configurations to determine the probability of intersystem LOCAs. This study concluded that periodic leak testing of the pressure isolation valves can substantially reduce intersystem LOCA probability.

The previous criteria of 1 gpm for all valve sizes is considered arbitrary and is not an indicator of imminent accelerated deterioration or potential valve failure. A study (EG&G Report, EGG-NTAP-6175) concluded allowable leak rates based on valve size was superior to a single allowable value. The single value imposes an unjustified penalty on the larger valves without providing information on potential valve degradation. In addition, enforcing the single value criteria resulted in higher personnel radiation exposures because larger valves must be repaired in place."

ATTACHED PAGES:

Encl. 3A 14

CHANGE NUMBER	NSHC	DESCRIPTION
6-25	LS-26	The Operational Leakage LCO has been modified to change the allowed leakage limit for reactor coolant system pressure isolation valves for consistency with NUREG-1431 Rev. 1. The RCS pressure isolation valve LCO permits system operation in the presence of leakage through valves in amounts which do not compromise safety. INSERT 3A-14b Q3.4.13-1
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 in within 12 hours after achieving steady state conditions. The RCS water inventory balance must be performed with the reactor at steady state conditions as discussed in the ITS Bases. This change is in conformance with traveler TSTF-116.
6-27	A	RCS leakage detection system descriptions are revised for consistency with current TS LCO 3.3.3.1 and USAR Sections 5.2.5.2.2 and 11.5.2.3.2.2.
6-28	LG	~~The current TS definition of CONTROLLED LEAKAGE is deleted to be consistent with NUREG-1431 Rev.1. The RCP seal water return flow limit is moved to a licensee controlled document. Seal injection limitations are established by the ECCS flow balance test procedures moved from CTS 4.5.2.h to a licensee controlled document (reference CN 2-15-LG of Enclosure 3A in the ECCS conversion package) and by the throttle valve position surveillance in ITS SR 3.5.2.7.~~ INSERT 3A-14 a Q3.5.5-2
7-01	R	Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
8-01	LS-16	This change in conformance with NUREG-1431 Rev. 1, revises the applicability of the specification to MODES 1, 2, or 3 with (T_{avg}) ≥ 500°F. The change deletes the requirement to perform an isotopic analysis for Iodine every 4 hours in Modes 4 and 5 and in Mode 3 below 500°F, whenever the reactor coolant exceeds its Dose Equivalent I-131 limit. In addition, this change deletes the requirement to perform the once per 4 hour surveillance for Dose Equivalent I-131 in the event the gross specific activity limit is exceeded, in accordance with industry traveler TSTF-28. The latter is an unnecessary requirement since

6-29	LS-38	INSERT 3A-14 c	Q3.4.14-3
6-30	A	INSERT 3A-14d	

INSERT 3A-14a Q 3.5.5-2

The current TS definition of CONTROLLED LEAKAGE is deleted as discussed in DOC 1-28-LG in Section 1.0. The RCP seal water return flow limit is moved to a licensee controlled document. Seal injection limitations are established by the throttle valve position surveillance in CTS SR 4.5.2.g.2) which is moved to ITS SR 3.5.2.7. This surveillance ensures that the ECCS analyses remain valid. Since facility performance and operational details of the type embodied by the RCP seal water return flow limit are required to be described in the USAR per 10CFR50.34, it is acceptable to move the requirements of CTS LCO 3.4.6.2.e and CTS SR 4.4.6.2.1.c to the USAR.

INSERT 3A-14b Q 3.4.13-1

The RCS is isolated from other systems by valves. During plant life these interfaces can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. Increasing allowed leakage limits from 1 gpm up to 5 gpm for the pressure isolation valves will not challenge the pressure relief capacity of interfacing systems. This amount of leakage is considered negligible when compared with the capacity of the pressure relief valves. Pressure isolation valve leakage limits apply to leakage rates for individual valves.

The basis for this LCO is the 1975 Reactor Safety Study (NUREG-75/014)) which identified potential intersystem Loss of Coolant Accidents (LOCAs) as a significant contributor to the risk of core melt. A subsequent study (NUREG-0677) evaluated various pressure isolation valve configurations to determine the probability of intersystem LOCAs. This study concluded that periodic leak testing of the pressure isolation valves can substantially reduce intersystem LOCA probability.

The previous criteria of 1 gpm for all valve sizes is considered arbitrary and is not an indicator of imminent accelerated deterioration or potential valve failure. A study (EG&G Report, EGG-NTAP-6175) concluded allowable leak rates based on valve size was superior to a single allowable value. The single value imposes an unjustified penalty on the larger valves without providing information on potential valve degradation. In addition, enforcing the single value criteria resulted in higher personnel radiation exposures because larger valves must be repaired in place."

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.13-2 **APPLICABILITY:** DC, WC, CA, CP

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 3.4-81

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev ~~2~~ 3	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		Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. ~~1~~ 2	Incorporated	3.4-31	Approved by NRC. TR 3.4-004
TSTF-93 Rev. 3	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. ~~3~~ 4	Incorporated	3.4-39	Q 3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev. ~~1~~ 2	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..
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
TSTF-285 ~~WOG-51, Rev. 1~~	Incorporated	~~3.4-23~~, 3.4-45, ~~3.4-52~~	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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	3.4-10	DCPP only Approved by NRC. 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

BASES

ACTIONS

A.1 (continued)

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 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 ~~and near operating pressure~~. Q3.4.13-2
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

(stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows)

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.13-3 **APPLICABILITY:** CP, WC, CA

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 3.4-79

BASES

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB.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 sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of 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 Systems 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.~~ Q3.4.13-3 Violation of this LCO could result in continued degradation of a component or system.

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

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.14-1 **APPLICABILITY:** CP, WC, CA

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 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 check 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: Q 3.4.14-2 **APPLICABILITY:** DC, CP, 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: DOC 6-11-LS-11 for Comanche Peak and Diablo Canyon is modified to include the bracketed information form 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 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:

Encl. 2	4-19
Encl. 5A	3.4-37, 3.4-38
Encl. 5B	B 3.4-86, B 3.4-87, B 3.4-88

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to.

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators ~~not isolated from the Reactor Coolant System~~ and 500 gallons per day through any one steam generator, 6-05-A

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. ~~8 gpm per RC pump CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and~~ 6-28-LG

f. ~~1 gpm leakage~~ **0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm** at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve. ~~specified in Table 3.4-1.*~~ 6-25-LS 6-07-LG 6-10-LG

APPLICABILITY: MODES 1, 2, 3, and 4# 6-08-LS-9

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

reduce the leakage rate to within limits, otherwise

Q3.4.14-2

WC 3.4-009 — remote manual

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, ~~reduce the leakage rate to within limits within 4 hours,~~ **isolate the high pressure portion of the affected system from the low pressure portion within four hours by use of at least one ~~closed manual~~, deactivated ~~automatic~~, 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 ~~HOT~~ **COLD** SHUTDOWN within the following ~~12~~ **30** hours. ~~with an RCS pressure of less than 600 psig.~~ **, *** 6-11-LS 6-12-M 6-24-M 6-29-LS-38 6-30-A Q3.4.14-3

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

remote manual WC 3.4-009

~~*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the natural test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.~~ 6-10-LG

#BB-PV-8702A/B and EJ-HV-8701 A/B are excluded in MODE 4 when in, or during transition to or from, the RHR mode of operation. 6-08-LS-9

##Each valve used to satisfy this action must have been verified to meet surveillance requirement 4.4.6.2.2. 6-12-M

** INSERT 4-19a
*** INSERT 4-19b
Q3.4.14-3

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the residual heat removal (RHR) flow path when in, or during the transition to or from, the RHR mode of operation.

ACTIONS

-----NOTES-----
1. Separate Condition entry is allowed for each flow path.

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	-----NOTE----- Each valve used to satisfy Required Action A.1 ~~and Required Action A.2~~ must have been verified to meet SR 3.4.14.1 and be in the reactor coolant pressure boundary ~~or the high pressure portion of the system~~.	ED Q3.4.14-2 B-PS (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. (continued)	A.1	Isolate the high pressure portion of the affected system from the low pressure portion by use of one ~~closed manual~~, deactivated ~~automatic~~ remote manual, or check valve.	4 hours	PS WC 3.4-009
	AND			
	~~A.2.1~~	~~Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.~~	~~72 hours~~	Q 3.4.14-2 B-PS
	~~OR~~			
	A.2.2	Restore RCS PIV to within limits.	72 hours	
B. Required Action and associated Completion Time for Condition A not met.	B.1	Be in MODE 3.	6 hours	
	AND			
	B.2	Be in MODE 5.	36 hours	
C. RHR suction isolation valve ~~System autoclosure~~ interlock function inoperable.	C.1	Isolate the affected penetration by use of one ~~closed manual or~~ deactivated ~~automatic~~ remote manual valve.	4 hours	3.4-11 B-PS WC 3.4-009

BASES

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

WC 3.4-010

Reference ~~7~~ 6 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability ~~or isolation of a leaking flow path~~

Q 3.4.14-2

(continued)

BASES

ACTIONS (continued)

~~with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.~~

Q3.4.14-2

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~~.

Q3.4.14-2

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 timeframe considers the time required to complete the Action and the low probability of a second valve failing during this time period.

Q3.4.14-2

block of text missing but is lined through (see next page)

Q3.4 Gen-1

B.1 and B.2

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 status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. 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.

(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].

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 PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

~~or~~

Q3.4.Gen-1

~~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

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 status, the plant must be brought to MODE 3

(continued)

BASES

ACTIONS (continued)

C.1

The inoperability of the RHR ~~System autoclosure~~ suction isolation valve interlock ~~renders the RHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing~~ could allow inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure. If the ~~RHR System autoclosure~~ RHR suction isolation valve interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated ~~automatic~~ remote manual valve within 4 hours. This Action accomplishes the purpose of the ~~autoclosure function~~ interlock.

WC3.4-009 undelete

WC 3.4-009

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Q3.4.14-2

Performance of leakage testing on each RCS PIV ~~or isolation valve~~ used to satisfy Required Action A.1 ~~and Required Action A.2~~ is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

redline Q3.4.Gen-1 redline Q3.4.Gen-1

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. ~~8~~ 7) as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. ~~7~~ 6), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

WC 3.4-010

WC3.4-010

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.14-3 **APPLICABILITY:** DC, CP, WC, CA

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	4-19
Encl. 3A	14
Encl. 3B	9
Encl. 4	2, new LS-38

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to.

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators ~~not isolated from the Reactor Coolant System~~ and 500 gallons per day through any one steam generator, **6-05-A**

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. ~~8 gpm per RC pump CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and~~ **6-28-LG**

f. ~~1 gpm leakage~~ **0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm** at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve. ~~specified in Table 3.4-1~~ ~~*~~ **6-25-LS** **6-07-LG** **6-10-LG**

APPLICABILITY: MODES 1, 2, 3, and 4# **6-08-LS-9**

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

reduce the leakage rate to within limits, otherwise

Q3.4.14-2

WC 3.4-009 — remote manual

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, ~~reduce the leakage rate to within limits within 4 hours,~~ **isolate the high pressure portion of the affected system from the low pressure portion within four hours by use of at least one ~~closed manual,~~ deactivated ~~automatic~~, 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 ~~HOT~~ **COLD** SHUTDOWN within the following ~~12~~ **30** hours. ~~with an RCS pressure of less than 600 psig.~~ ***, **** **6-11-LS** **6-12-M** **6-24-M** 6-29-LS-38 6-30-A Q3.4.14-3

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

remote manual WC 3.4-009

~~*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the natural test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.~~ **6-10-LG**

#BB-PV-8702A/B and EJ-HV-8701 A/B are excluded in MODE 4 when in, or during transition to or from, the RHR mode of operation. **6-08-LS-9**

##Each valve used to satisfy this action must have been verified to meet surveillance requirement 4.4.6.2.2. **6-12-M**

*** INSERT 4-19a
**** INSERT 4-19b
Q3.4.14-3

INSERT 4-19a Q 3.4.14-3

** Separate Condition entry is allowed for each PIV flow path. 6-29-LS-38

INSERT 4-19b Q 3.4.14-3

*** Enter applicable Required Actions for systems made inoperable by an inoperable PIV. 6-30 A

CHANGE NUMBER	NSHC	DESCRIPTION
6-25	LS-26	The Operational Leakage LCO has been modified to change the allowed leakage limit for reactor coolant system pressure isolation valves for consistency with NUREG-1431 Rev. 1. The RCS pressure isolation valve LCO permits system operation in the presence of leakage through valves in amounts which do not compromise safety. INSERT 3A-14b Q3.4.13-1
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 in within 12 hours after achieving steady state conditions. The RCS water inventory balance must be performed with the reactor at steady state conditions as discussed in the ITS Bases. This change is in conformance with traveler TSTF-116.
6-27	A	RCS leakage detection system descriptions are revised for consistency with current TS LCO 3.3.3.1 and USAR Sections 5.2.5.2.2 and 11.5.2.3.2.2.
6-28	LG	~~The current TS definition of CONTROLLED LEAKAGE is deleted to be consistent with NUREG-1431 Rev.1. The RCP seal water return flow limit is moved to a licensee controlled document. Seal injection limitations are established by the ECCS flow balance test procedures moved from CTS 4.5.2.h to a licensee controlled document (reference CN 2-15-LG of Enclosure 3A in the ECCS conversion package) and by the throttle valve position surveillance in ITS SR 3.5.2.7.~~ INSERT 3A-14a Q3.5.5-2
7-01	R	Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
8-01	LS-16	This change in conformance with NUREG-1431 Rev. 1, revises the applicability of the specification to MODES 1, 2, or 3 with (T_{avg}) ≥ 500°F. The change deletes the requirement to perform an isotopic analysis for Iodine every 4 hours in Modes 4 and 5 and in Mode 3 below 500ºF, whenever the reactor coolant exceeds its Dose Equivalent I-131 limit. In addition, this change deletes the requirement to perform the once per 4 hour surveillance for Dose Equivalent I-131 in the event the gross specific activity limit is exceeded, in accordance with industry traveler TSTF-28. The latter is an unnecessary requirement since

6-29	LS-38	INSERT 3A-14 c	Q3.4.14-3
6-30	A	INSERT 3A-14d	

INSERT 3A-14c Q 3.4.14-3

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 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.

INSERT 3A-14d 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.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.4

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
6-24 M	Revises ACTION to require going to COLD SHUTDOWN rather than HOT SHUTDOWN with an RCS pressure less than 600 psig.	No - Not part of current DCPP TS.	No - The 600 psig action is not part of the current TS.	Yes	Yes
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 ≤ .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 USAR 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 - Not in CTS.	No - Not in CTS.	Yes. Moved to USAR.	Yes. Moved to FSAR.
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 to TRM.	No - Amendment 89 relocated to USAR Chapter 16.	No - Amendment 103 relocated to FSAR Chapter 16.
8-01 LS-16	This change revises the applicability of the specification to MODES 1, 2, or 3 with $(T_{avg}) \geq 500°F$. The change deletes the requirement to perform an isotopic analysis for Iodine every 4 hours in MODES 4 and 5 and in MODE 3 below 500°F, whenever the reactor coolant exceeds its Dose Equivalent Iodine or at any time the reactor coolant exceeds Gross Specific Activity limits.	Yes	Yes	Yes	Yes

6-29
LS-38
INSERT 3B - 9a
6-30
A

Q3.4.14-3

INSERT 3B-9a

Q 3.4.14-3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
6-29 LS-38	Separate Condition entry is allowed for each flow path with excessive leakage from RCS PIVs.	Yes	Yes	Yes	Yes
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

NO SIGNIFICANT HAZARDS CONSIDERATIONS (NSHC)
CONTENTS (continued)

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, the LCO is revised to allow separate Condition entry for each flow path with excessive leakage from RCS PIVs.

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 flow path. 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 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.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS-38
(continued)

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 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, within 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 flow paths 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: Q 3.4.14-4 **APPLICABILITY:** WC, CA

REQUEST: Change 6-24 M (Callaway and Wolf Creek)

Comment: Cold shutdown rather than hot shutdown is more restrictive however, the discussion does not address the extension of the time from 12 to 30 hours.

FLOG RESPONSE: DOC 6-24-M is revised to add the following:

"CTS LCO 3.0.3 specifies the standard shutdown track Completion Times when Required Actions and Completion Times aren't met as MODE 3 within 6 hours, MODE 4 within 12 hours, and MODE 5 within 36 hours. This DOC changes the termination point of the shutdown track in CTS 3.4.6.2 ACTION c from the non-standard MODE 4 with RCS pressure less than 600 psig in 18 hours to the standard MODE 5 within 36 hours. The cumulative effect is a more restrictive change."

ATTACHED PAGES:

Encl. 3A 13

CHANGE NUMBER	NSHC	DESCRIPTION
6-19	TR-3	This change in conformance with NUREG-1431 Rev. 1, removes the specific requirement for performing the PIV surveillance prior to returning a valve to service following maintenance, repair or replacement. Explicit post maintenance TS surveillance requirements have been deleted because these requirements are adequately addressed by administrative post-maintenance programs.
6-20	A	Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
6-21	LS-35	Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
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 inadvertantly 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	Specification 3.4.6.1 (Leakage Detection Systems) is revised such that the provisions of Specification 3.0.4 are not applicable. This will allow entry into the applicable MODES with only one of the Leakage Detection Systems OPERABLE, subject to the requirements of the ACTION statements. This change is consistent with NUREG-1431 Rev. 1 and traveler TSTF-60 and is acceptable because of the diverse means available to detect RCS leakage. INSERT 3A-13a — Q3.4.15-1
6-24	M	Action c of Specification 3.4.6.2 (Operational Leakage) is revised for consistency with NUREG-1431 Rev. 1 to require going to COLD SHUTDOWN rather than going to HOT SHUTDOWN with an RCS pressure less than 600 psig. ~~This is a more restrictive shutdown requirement.~~ INSERT 3A-13b Q3.4.14-4

INSERT 3A-13a Q 3.4.15-1

In addition, given the leakage detection diversity, the ACTION for CTS 3.4.6.1 is revised to allow continued operation for up to 30 days under compensatory actions as long as one of the detection systems is OPERABLE. This condition is superior from a plant safety perspective than imposing a plant shutdown transient under LCO 3.0.3 which could give rise to an initiating event when the plant's leakage monitoring capability is degraded."

INSERT 3A-13b Q 3.4.14-4

CTS LCO 3.0.3 specifies the standard shutdown track Completion Times when Required Actions and Completion Times aren't met as MODE 3 within 6 hours, MODE 4 within 12 hours, and MODE 5 within 36 hours. This DOC changes the termination point of the shutdown track in CTS 3.4.6.2 ACTION c from the non-standard MODE 4 with RCS pressure less than 600 psig in 18 hours to the standard MODE 5 within 36 hours. The cumulative effect is a more restrictive change.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.14-5 **APPLICABILITY:** DC, WC

REQUEST: Change 6-25 LS-26 (Diablo Canyon and Wolf Creek)

Comment: The justification of the change is inadequate. The NSHC contains the proper justification.

FLOG RESPONSE: See the response to Comment Number 3.4.13-1.

ATTACHED PAGES:

See attached pages for response to Comment Number 3.4.13-1.

ADDITIONAL INFORMAT ON COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.15-1 **APPLICABILITY:** DC, WC, CA

REQUEST: ITS 3.4.15 and Bases ITS 3.4.15 Required Action E.1 (Callaway, Diablo Canyon and Wolf Creek)

Comment: Callaway and Wolf Creek: As written ITS 3.4.15 does not implement CTS 3.4.6.1 as marked up (allowing up to two methods to be inoperable). Specifically, in the ITS as written, with two monitoring methods inoperable TS 3.0.3 would have to be entered as there is no Condition for two methods inoperable. Diablo Canyon: ITS 3.4.15 and Bases ITS 3.4.15 Required Action E.1. E.1 Bases state that "With two of the three groups of leak detection monitoring not operable, the two groups will enter their respective ACTION and Completion statements." What in the construction of the ITS supports that statement and more importantly what is the justification for this as the CTS requires 2 of 3 groups of equipment to be operable?

FLOG RESPONSE: Given the independence of the three monitoring systems, the plant can simultaneously be in Conditions A and B, A and C, or B and C, but not in all three given Condition E invoking ITS LCO 3.0.3 if all three monitoring systems are inoperable. If ITS LCO 3.0.3 were intended for the simultaneous inoperability of two systems, Condition E (Condition F in NUREG-1431) would be so worded. This is also supported by ITS page 1.3-1 (second paragraph of the Description section) and Example 1.3-3 which clearly state the plant may be in multiple Conditions at the same time. DOC 6-23-LS-25 is revised to add:

"In addition, given the leakage detection diversity, the ACTION for CTS 3.4.6.1 is revised to allow continued operation for up to 30 days under compensatory actions as long as one of the detection systems is OPERABLE. This condition is superior from a plant safety perspective than imposing a plant shutdown transient under LCO 3.0.3 which could give rise to an initiating event when the plant's leakage monitoring capability is degraded."

For Diablo Canyon, the Bases statement "With two of the three groups of leak detection monitoring not operable, the two groups will enter their respective ACTION and Completion statements" is deleted.

ATTACHED PAGES:

Encl. 3A	13
Encl. 3B	8
Encl. 4	52, 53

CHANGE NUMBER	NSHC	DESCRIPTION
6-19	TR-3	This change in conformance with NUREG-1431 Rev. 1, removes the specific requirement for performing the PIV surveillance prior to returning a valve to service following maintenance, repair or replacement. Explicit post-maintenance TS surveillance requirements have been deleted because these requirements are adequately addressed by administrative post-maintenance programs.
6-20	A	Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
6-21	LS-35	Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
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 inadvertantly 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	Specification 3.4.6.1 (Leakage Detection Systems) is revised such that the provisions of Specification 3.0.4 are not applicable. This will allow entry into the applicable MODES with only one of the Leakage Detection Systems OPERABLE, subject to the requirements of the ACTION statements. This change is consistent with NUREG-1431 Rev. 1 and traveler TSTF-60 and is acceptable because ~~of the diverse means available to detect RCS leakage.~~ INSERT 3A-13a — Q3.4.15-1
6-24	M	Action c of Specification 3.4.6.2 (Operational Leakage) is revised for consistency with NUREG-1431 Rev. 1 to require going to COLD SHUTDOWN rather than going to HOT SHUTDOWN with an RCS pressure less than 600 psig. ~~This is a more restrictive shutdown requirement.~~ INSERT 3A-13b Q3.4.14-4

INSERT 3A-13a Q 3.4.15-1

In addition, given the leakage detection diversity, the ACTION for CTS 3.4.6.1 is revised to allow continued operation for up to 30 days under compensatory actions as long as one of the detection systems is OPERABLE. This condition is superior from a plant safety perspective than imposing a plant shutdown transient under LCO 3.0.3 which could give rise to an initiating event when the plant's leakage monitoring capability is degraded."

INSERT 3A-13b Q 3.4.14-4

CTS LCO 3.0.3 specifies the standard shutdown track Completion Times when Required Actions and Completion Times aren't met as MODE 3 within 6 hours, MODE 4 within 12 hours, and MODE 5 within 36 hours. This DOC changes the termination point of the shutdown track in CTS 3.4.6.2 ACTION c from the non-standard MODE 4 with RCS pressure less than 600 psig in 18 hours to the standard MODE 5 within 36 hours. The cumulative effect is a more restrictive change.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
6-16 LS-14	This change removes the requirement for monitoring the reactor head flange leakoff system.	Yes	Yes	Yes	Yes
6-17 LG	The definition of steady state is moved to Bases.	Yes -	Yes	No - WCGS does not have this definition.	No - Callaway does not have this definition.
6-18 LS-15	This change relaxes the requirement for PIV testing following operation in MODE 5. The previous requirement was testing following 72 hours in MODE 5 which is revised to 7 days in MODE 5.	No - Not part of current DCPP TS.	Yes	Yes	No - already in current TS per Amendment 105.
6-19 TR-3	This change removes the specific requirement for performing the PIV surveillance prior to returning a valve to service following maintenance, repair or replacement.	Yes	Yes	Yes	Yes
6-20 A	IST requirements are moved to Section 5 of the improved TS.	Yes	Yes	No - WCGS does not have this requirement.	No - Callaway does not have this requirement.
6-21 LS-35	This change increases the RCP seal injection flow Completion Time from 4 to 72 hours, with a new added verification that at least 100% of the assumed charging flow remains available.	Yes	Yes	No - See CN 6-28-LG.	No - See CN a6-28-LG.
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.	No - not part of current DCPP TS.	Yes	Yes	Yes
6-23 LS-25	The leakage detection system specification is revised such that the provisions of 3.0.4 are not applicable, and two monitoring systems can be inoperable without invoking LCO 3.0.3. Q3.4.15-1	Yes	No - The non-applicabilty of 3.0.4 is already part of the current TS.	Yes	Yes

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS-25
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Specification 3.4.6.1 is revised such that the provisions of Specification 3.0.4 are not applicable. This will allow entry into the applicable MODES with only one of the Leakage Detection Systems OPERABLE, subject to the requirements of the ACTION statements. This change is consistent with NUREG-1431 Rev. 1 and traveler TSTF-60 and is acceptable because of the diverse means available to detect RCS leakage. Q3.4.15-1

INSERT LS-25

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

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*

2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*

3. *Involve a significant reduction in a margin of safety."*

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

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The primary function of the Leakage Detection Systems is to detect significant reactor coolant pressure boundary (RCPB) degradation as soon as practical to minimize the potential for propagation to a gross failure. The TS requires multiple diverse systems to ensure that leakage from a variety of locations and leakage rates can be detected in sufficient time to take measures to place the plant in a safe condition. These system are passive and can not initiate or increase the consequences of an accident. No credit is explicitly taken for these systems in the accident analyses. Entry into the applicable MODES while subject to the compensatory measures called

or plant operation with two monitoring systems inoperable for up to 30 days Q3.4.15-1

In addition, given the leakage detection diversity, the ACTION for CTS 3.4.6.1 is revised to allow continued operation for up to 30 days under compensatory actions as long as one of the detection systems is OPERABLE. Allowing continued plant operation for 30 days, with alternate methods invoked to further monitor RCPB integrity and at least one monitoring system available to detect abnormal leakage, is superior from a plant safety perspective than imposing a plant shutdown transient under LCO 3.0.3 which could give rise to an initiating event when the plant's leakage monitoring capability is degraded.

NSHC LS-25
(continued)

for in the ACTION statements will not have any effect on the status of the RCPB. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The method of plant operation is unaffected since the change is based on the acceptability of the revised ~~current~~ ACTION statement~~s~~ in providing compensatory RCPB leakage detection capability. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated. Q3.4.15-1

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

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

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS-25" 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: Q 3.4.15-2 **APPLICABILITY:** WC, CA

REQUEST: CTS 3.4.6.1 b&c and CTS 4.4.6.1 b&c markups (Callaway and Wolf Creek)

Comment: Have the systems been renamed, were the names in the CTS incorrect, or are different systems being relied on in the ITS?

FLOG RESPONSE: The same systems are being used. The system names in the CTS have been revised to be consistent with the names used in USAR Section 5.2.5.2.2. DOC 6-27-A already provides this explanation.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.15-3 **APPLICABILITY:** WC

REQUEST: ITS Bases Page B 3.4-97 (Wolf Creek)

Comment: In the smooth Bases discussion of A.1 and A.2 it should be "and makeup" not "andmakeup"

FLOG RESPONSE: The smooth copy of the ITS has been marked to read "and makeup". A final review of the smooth ITS and ITS Bases is planned prior to resubmitting to the NRC the smooth copy of the ITS and Bases.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.16-1 **APPLICABILITY:** DC, CP, WC, CA

REQUEST: Difference 3.4-39

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

FLOG RESPONSE: See the response to Comment Number 3.4.11-3.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.16-2 **APPLICABILITY:** WC

REQUEST: ITS Figure 3.4.16.1 (Wolf Creek)

Comment: In order to be consistent with the ITS LCO and CTS Figure 3.4-1 the units should be micro (μ) Curies/gm and not milli (m) Curies/gm as indicated.

FLOG RESPONSE: ITS Figure 3.4.16-1 units have been revised to "(μCi/gm)".

ATTACHED PAGES:

Encl. 5A 3.4-49

3.4.16-2

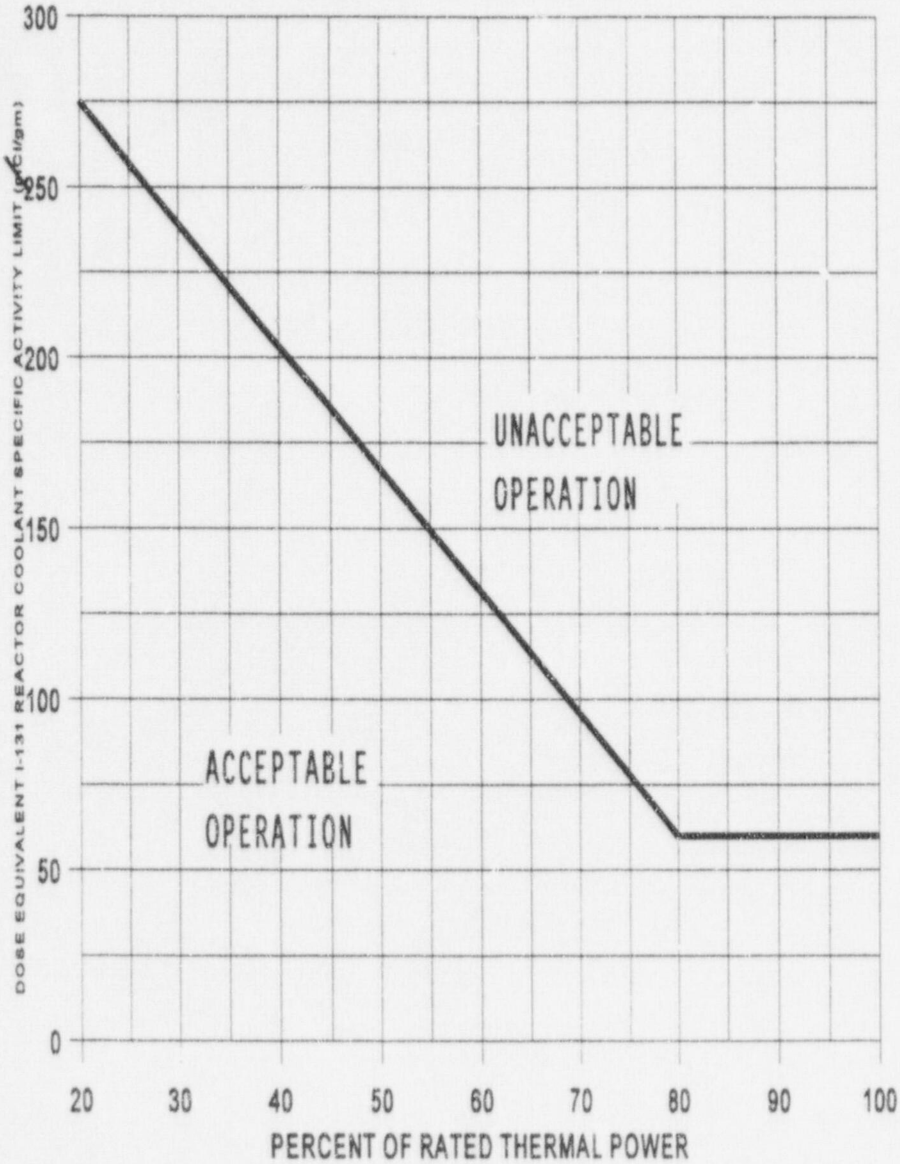

FIGURE 3.4.16-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity
Limit Versus Percent of RATED THERMAL POWER

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.16-3 **APPLICABIL!TY:** WC

REQUEST: ITS Bases 3.4.16 Applicability (Wolf Creek)

Comment: Page B 3.4-103 of the smooth Bases should read "the reactor" not "thereactor"

FLOG RESPONSE: The smooth copy of the ITS has been marked to read "the reactor". A final review of the smooth ITS and ITS Bases is planned prior to resubmitting to the NRC the smooth copy of the ITS and Bases.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

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

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 DCPP, 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. (Not applicable to DCPP.)

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 DCPP, WCGS and CPSES.)

ATTACHED PAGES:

Encl. 5B B 3.4-19, B 3.4-43

BASES (continued)

SL 3.4.Gen-1

APPLICABLE SAFETY ANALYSES (continued)

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the limit values, 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.~~ 10 CFR 50.36(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 at rated power.

redline Q 3.4 Gen-1 CA 3.4-004

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.

APPLICABILITY

In MODES 1 and 2, the reactor ~~is critical and thus~~ when critical 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)

BASES

LCO (continued)

insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. ~~The exact design value of [125 kW is derived from the use of seven heaters rated at 17.9 kW each]. The amount needed to maintain pressure is dependent on the heat losses.~~

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

either the offsite power source or the

In MODES 1, 2, and 3, there is a need to maintain the availability of pressurizer heaters, capable of being powered from 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. WC3.4-002

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 bring the plant to a MODE in which the LCO does not apply. ~~restore the plant to 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. Additionally, the unit must be brought ~~with 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.~~ CA3.4-004

(e.g., by de-energizing all CRDMs, by opening the RTBs, or de-energizing the motor generator sets.)

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: 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 are approved by NRC. Add Rev. 1 to TSTF-94 and TSTF-151.

ATTACHED PAGES:

Encl. 5A	Traveler Status page
Encl. 6B	5

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev ~~2~~ 3	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		Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. ~~1~~ 2	Incorporated	3.4-31	Approved by NRC. TR 3.4-004
TSTF-93 Rev. 3	Incorporated	3.4-17	Approved by NRC Q3.4.9-3
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. ~~3~~ 4	Incorporated	3.4-39	Q 3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev. ~~1~~ 2	Incorporated	3.4-36	Q3.4.13-2
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..
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
TSTF-285 ~~WOG-51, Rev. 1~~	Incorporated	3.4-22, 3.4-45, 3.4-52	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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	3.4-10	DCPP only Approved by NRC. TR 3.4-009
WOG-87, Rev. 2	Incorporated	3.4-47	Q 3.4.11-4
~~WOG-90~~ 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	Q3.4.12-1

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
3.4-29	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 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 TR 3.4-004	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

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	3.4-18
Encl. 5B	B 3.4-38

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-P5 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.~~ Reactor vessel water level is above the vessel flange. 3.4-03

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

BASES

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

Note 1 permits all RHR pumps to be ~~de-energized~~ **removed from operation** for ≤ **1 hour** ~~15 minutes when switching from one loop to another~~. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet temperature is maintained > 10°F below saturation temperature. ~~The Note prohibits boron dilution or draining operations when RHR forced flow is stopped.~~ **The Note requires reactor vessel water level be above the vessel flange to ensure the operating RHR pump will not be intentionally deenergized during mid-loop operations.**

at least

WC 3.4-001

redline

Q 3.4.Gen-1

Note 2 allows one RHR loop to be inoperable for a period of ≤ **2 hours**, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

Q 3.4.7-2

APPLICABILITY

In MODE 5 with loops not filled ~~(as defined in plant procedures)~~, this LCO requires core heat removal and coolant circulation by the RHR System. One RHR loop provides sufficient capability for this purpose. However, one additional RHR loop is required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
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 3.9.5, "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)

ADDITIONAL INFORMATION COVER SHEET

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

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 3.4-43

BASES

LCO (continued)

insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. ~~The exact design value of [125 kW is derived from the use of seven heaters rated at 17.9 kW each]. The amount needed to maintain pressure is dependent on the heat losses.~~

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

either the offsite powersource or the

WC3.4-002

In MODES 1, 2, and 3, there is a need to maintain the availability of pressurizer heaters, capable of being powered from 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.

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 bring the plant to a MODE in which the LCO does not apply. ~~restore the plant to 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. Additionally, the unit must be brought ~~with 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.~~

CA3.4-004

(e.g., by de-energizing all CRDMs, by opening the RTBs, or de-energizing the motor generator sets.)

(continued)

ADDITIONAL INFORMATION COVER SHEET

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

REQUEST: Editorial change in ITS Bases 3.4.1 Actions from "C.3" to "C.2". The Required Actions were revised and renumbered and the ITS Bases were not revised accordingly.

ATTACHED PAGES:

Encl. 5B B 3.4-5, B 3.4-6

BASES

ACTIONS

C.1.1 (continued)

With RCS flow rate not within limits, the unit is allowed 2 hours to restore flow rate to within limits in accordance with Required Action C.1.1.

C.1.2.1, C.1.2.2 and C.1.2.3

If RCS flow rate is not restored to within limit, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action C.1.2.1 and reduce the Power Range Neutron Flux - High trip setpoints to ≤ 55% RTP in accordance with Required Action C.1.2.2. Reducing power to < 50% RTP increases the DNB margin. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 2 hours for Required Action C.1.2.1 is consistent with that allowed for Required Action C.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 2 hours for Required Actions C.1.1 and C.1.2.1 are not additive.

The allowed Completion Time of 6 hours to reset the trip setpoints per Required Action C.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied; and additional time becomes available to reduce the trip setpoints.

If RCS flow rate cannot be restored to within limit, the plant must be brought to a MODE in which the LCO does not apply. This requires the plant to be placed in at least MODE 2 (RTP ≤ 5%). Once power is below 5%, the potential for violating accident analysis assumptions is eliminated. The Completion Time of 74 hours for C.1.2.3 is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this time period.

~~C.3~~ C.2 $\boxed{\text{WC 3.4-004}}$

Subsequent return to power operation is performed in stages to assure that RCS flow rate is within limits prior to exceeding 50% RTP, 75% RTP and within 24 hours of achieving ≥ 95% RTP. Action C.3 assures that the condition leading to reduced RCS flow rate has been identified,

(continued)

BASES

C.2

ACTIONS

~~C.3~~ (continued) WC 3.4-004

to the extent necessary, and corrected prior to unrestricted power operation.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action. For example, this means that, during performance of Required Action C.1.1, if the flow rate is restored to within limit at 80% RTP, power does not need to be reduced below 50% RTP or 75% RTP to comply with Required Action C.2.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 3.4-006 **APPLICABILITY:** WC

REQUEST: CTS 3.4.2.2 establishes the pressurizer safety valve setpoint at 2485 ±1% psig. This corresponds to a setpoint pressure range of 2460.15 psig to 2509.85 psig. In NUREG-1431, Rev. 1, the setpoint pressure range is identified as ≥[2460] psig to ≤[2510] psig. A portion of the range specified in the ISTS and ITS would be outside the CTS range of 2485 ±1% psig. Wolf Creek surveillance procedure, STS MT-005, currently specifies this range as 2461 to 2509 psig. Therefore, the ITS 3.4.10 LCO is modified to reflect the conservative lift setting currently specified in the CTS and plant procedures.

ATTACHED PAGES:

Encl. 5A 3.4-22

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ ~~2460~~ 2461 psig and ≤ ~~2510~~ 2509 psig. B B-PS WC 3.4-006

APPLICABILITY: MODES 1, 2, and 3
~~MODE 4 with all RCS cold leg temperatures > 275°F.~~ 3.4-18

------------------------------NOTE------------------------------
The lift settings are not required to be within the LCO limits during MODES 3 ~~and 4~~ for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup. 3.4-18 B

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1	Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 AND B.2	Be in MODE 3. Be in MODE 4. ~~with any RCS cold leg temperatures ≤ 275°F.~~	6 hours 12 hours

3.4-18

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 3.4-007 **APPLICABILITY:** DC, WC, CA

REQUEST: Revise the Frequency of ITS SR 3.4.11.2 to read: "In accordance with the Inservice Testing Program" consistent with the CTS.

ATTACHED PAGES:

Encl. 5A	3.4-28
Encl. 5B	B 3.4-58
Encl. 6A	10
Encl. 6B	8

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*s* of ~~Condition A B or E.~~ this LCO. ~~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 Condition B or E.~~	92 days	3.4-47 Q3.4.11-4 3.4-35 3.4-47 Q3.4.11-4
SR 3.4.11.2	------------------NOTE------------------ Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each PORV.	In accordance with the Inservice Testing Program ~~18 months~~	3.4-53 3.4-35 B WC 3.4-007
~~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
~~SR 3.4.11.4~~	~~Verify PORVs and block valves are capable of being powered from emergency power sources.~~	~~[18] months~~	B-PS

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1 (continued)

Q3.4.11-4

~~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).~~

This SR is modified by two Notes. Q3.4.11-2

~~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 Actions of 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. Q3.4.11-4 Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2.

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.~~ WC3.4-007 Operating experience has shown that these valves usually pass the Surveillance when performed at the required Inservice Testing Program frequency. The Frequency is acceptable from a reliability standpoint. The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. In accordance with Reference 5, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.

~~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.~~

(continued)

CHANGE NUMBER	JUSTIFICATION
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.
3.4-51	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
3.4-52	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
3.4-53	INSERT 6A-10a — WC 3.4-007

3.4-53 This change revises the ITS SR 3.4.11.2 Frequency from "18 months" to "In accordance with the Inservice Testing Program." The CTS for this surveillance establishes the frequency as being per the IST Program [CTS SR 4.4.4.1].

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	
3.4-50	This change is consistent with current TS SR 4.4.9.3.3. The 12 hour frequency applies to vent pathways 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 flow rate verification on the 18 month frequency. DC ALL-005	Yes	No	No	No	
3.4-52	Consistent with traveler ~~OG-81~~ TSTF-285, 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.	Q3.4.12-2

3.4-53 INSERT 6B-8a WC 3.4-007

3.4-54 INSERT 6B-8b DC 3.4-003

INSERT 6B-8a

WC 3.4-007

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-53	This change revises the ITS SR 3.4.11.2 Frequency from "18 months" to "In accordance with the Inservice Testing Program." The CTS for this surveillance establishes the frequency as being per the IST Program [CTS SR 4.4.4.1].	No	No	Yes	Yes

INSERT 6B-8b

DC 3.4-003

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-54	Consistent with the current license bases as approved LA 124/122, ITS LCO 3.4.13 is revised to reflect reduced steam generator primary-to-secondary leakage limits of 150 gallons per day from any one steam generator and an additional surveillance Requirement to determine primary-to-secondary leakage every 72 hours.	Yes	No	No	No

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 3.4-008 **APPLICABILITY:** WC, CA

REQUEST: Revise ITS SR 3.4.12.8 Bases to clarify when the SR must be performed.

ATTACHED PAGES:

Encl. 5B B 3.4-75

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.6 (continued)

develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.7 Not Used.

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valve are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.4 for the RHR suction valve Surveillance and for a description of the requirements of the Inservice Testing Program.) This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.~~

block of text missing that would be lined/struck through — Q 3.4 Gen-1

SR 3.4.12.8

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ ~~275~~ 368°F and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

The 12 hour allowance ~~Frequency~~ considers the unlikelihood of a low temperature overpressure event during this time.

A Note has been added indicating that this SR is not required to be ~~met~~ performed until 12 hours after decreasing RCS cold leg temperature to ≤~~275~~ 368°F. ~~The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting.~~ ~~The test must be performed within 12 hours after entering the LTOP MODES.~~ WC 3.4-008

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 3.4-009 **APPLICABILITY:** WC

REQUEST: Revise ITS 3.4.14, Required Action A.1 and C.1 from "deactivated automatic" to "deactivated remote manual" since the RHR suction valves are remote manual valves and all other pressure isolation valves are check valves.

ATTACHED PAGES:

Encl. 2	4-19
Enc. 5A	3.4-38
Encl. 5B	B 3.4-88

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to.

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators ~~not isolated from the Reactor Coolant System~~ and 500 gallons per day through any one steam generator, **6-05-A**

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. ~~8 gpm per RC pump CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and~~ **6-28-LG**

f. ~~1 gpm leakage~~ **0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm** at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve. ~~specified in Table 3.4-1.*~~ **6-25-LS** **6-07-LG** **6-10-LG**

APPLICABILITY: MODES 1, 2, 3, and 4#. **6-08-LS-9**

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

WC 3.4-009 — remote manual

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, ~~reduce the leakage rate to within limits within 4 hours,~~ **isolate the high pressure portion of the affected system from the low pressure portion within four hours by use of at least one closed manual, deactivated automatic, 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 ~~HOT~~ **COLD** SHUTDOWN within the following ~~12~~ **30** hours. ~~with an RCS pressure of less than 600 psig.~~ **, *** **6-11-LS** **6-12-M** **6-24-M**

reduce the leakage rate to within limits, otherwise

Q 3.4.14-2

6-29-LS-38
6-30-A
Q 3.4.14-3

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

remote manual WC 3.4-009

~~*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the natural test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.~~ **6-10-LG**

#BB-PV-8702A/B and EJ-HV-8701 A/B are excluded in MODE 4 when in, or during transition to or from, the RHR mode of operation. **6-08-LS-9**

##Each valve used to satisfy this action must have been verified to meet surveillance requirement 4.4.6.2.2. **6-12-M**

** INSERT 4-19a
*** INSERT 4-19b
— Q 3.4.14-3

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. (continued)	A.1	Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve. remote manual	4 hours	PS WC 3.4-009
	AND			
	~~A.2.1~~	~~Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.~~	~~72 hours~~	Q 3.4.14-2 B-PS
	~~OR~~			
	A.2.2	Restore RCS PIV to within limits.	72 hours	
B. Required Action and associated Completion Time for Condition A not met.	B.1	Be in MODE 3.	6 hours	
	AND			
	B.2	Be in MODE 5.	36 hours	
C. RHR suction isolation valve ~~System autoclosure~~ interlock function inoperable.	C.1	Isolate the affected penetration by use of one ~~closed manual or~~ deactivated automatic valve. remote manual	4 hours	3.4-11 B-PS WC 3.4-009

BASES

ACTIONS (continued)

C.1

The inoperability of the RHR ~~System autoclosure~~ suction isolation valve interlock ~~renders the RHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing~~ could allow inadvertent opening ~~of the valves~~ at RCS pressures in excess of the RHR systems design pressure. If the ~~RHR System autoclosure~~ RHR suction isolation valve interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated ~~automatic~~ valve within 4 hours. This Action accomplishes the purpose of the ~~autoclosure function~~ interlock.

WC3.4-009

undelete

remote manual

WC 3.4-009

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Q3.4.14-2

Performance of leakage testing on each RCS PIV ~~or isolation valve~~ used to satisfy Required Action A.1 ~~and Required Action A.2~~ is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

redline Q3.4.Gen-1

redline Q3.4.Gen-1

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. ~~8~~ 7) as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. ~~7~~ 6), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

WC 3.4-010

WC 3.4-010

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 3.4-010 **APPLICABILITY:** WC, CA

REQUEST: Move the CTS list of Pressure Isolation Valves to the Background Bases for ITS 3.4.14.

ATTACHED PAGES:

Encl. 3B	6
Encl. 5B	B 3.4-85, B 3.4-86, B 3.4-88, B 3.4-90

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5-02 A	CTS 3.4.5 is deleted. Steam Generator operability requirements in MODES 1-4 are specified in the RCS loop and leakage specifications.	Yes	No - CPSES does not have this specification.	Yes	Yes
5-03 A	Clarification to remove potential interpretation problems related to probe orientation versus entry point.	Yes	No - Same as CPSES change 1-15 for CTS Section 3/4.0	Yes	Yes
6-01 M	This change adds the performance of an RCS water inventory balance every 24 hours as a new requirement when the [containment sump level and flow monitoring system] is inoperable.	Yes	Yes	Yes	Yes
6-02 LS-8	This change allows the performance of an RCS water inventory balance every 24 hours as an alternative to the requirement to perform 24 hour samples of the containment atmosphere when a required radioactivity monitor is inoperable.	Yes	Yes	Yes	Yes
6-03 A	This change adds the word "required" to clarify that only those detectors which are being used to satisfy the LCO must be demonstrated to be OPERABLE.	Yes	Yes	Yes	Yes
6-04 A	The word "DIGITAL" has been deleted to be consistent with the terminology used in NUREG-1431 as it relates to Channel Operational Tests.	No - "Digital" not included in CTS.	Yes	No - "Digital" not included in CTS.	No - "Digital" not included in CTS.
6-05 A	This change deletes the phrase "not isolated from the Reactor Coolant System" when referring to leakage through the SGs.	No - The phrase is not part of the current DCPP TS.	Yes	Yes	Yes
6-06 A	This change moves the LCO for CONTROLLED LEAKAGE (seal injection flow) from "Operational Leakage" to LCO 3.5.5 "ECCS"	Yes	Yes	No - See CN 6-28-LG.	No - See CN 6-28-LG.
6-07 LG	This change moves the listing of RCS Pressure Isolation Valves.	Yes - Moved to the FSAR.	Yes - Moved to TRM	Yes - Moved to ~~USAR Chapter 16.~~ ITS 3.4.14 Bases	Yes -Moved to ~~FSAR Table 16.4-1~~ ITS 3.4.14 Bases

WC 3.4-010

BASES

BACKGROUND (continued)

study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System;

b. Safety Injection System; and

c. Chemical and Volume Control System.

The PIVs are listed below: ~~in the FSAR USAR, Chapter 16 Section [ ] (Ref. 6).~~ INSERT B 3.4-85 WC 3.4-010

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (c)(2)(ii).

(continued)

VALVE NUMBER	FUNCTION
BBV8948 A, B, C, D	SI/RHR/Accumulator Cold Leg Injection
BBV8949 A, B, C, D	SI/RHR Hot Leg Injection
BBV001, 022, 040, 059	BIT Cold Leg Injection
BBPV8702 A, B	RHR Normal Suction
EJV8841 A, B	RHR Hot Leg Recirc Ctmt Isolation
EJHV8701 A, B	RHR Normal Suction
EMV001, 002, 003, 004	SI Hot Leg Injection Ctmt Isolation
EM 8815	BIT Injection Ctmt Isolation
EPV010, 020, 030, 040	SI Cold Leg Injection Ctmt Isolation
EPV8818 A, B, C, D	RHR Cold Leg Injection Ctmt Isolation
EPV8956 A, B, C, D	Accumulator Injection Isolation

BASES

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

WC 3.4-010

Reference ~~7~~ 6 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, ~~or isolation of a leaking flow path~~

Q3.4.14-2

(continued)

BASES

ACTIONS (continued)

C.1

The inoperability of the RHR ~~System autoclosure~~ suction isolation valve interlock ~~renders the RHR suction isolation valves~~ ~~incapable of isolating in response to a high pressure condition and preventing~~ could allow inadvertent opening ~~of the valves~~ at RCS pressures in excess of the RHR systems design pressure. If the ~~RHR System autoclosure~~ RHR suction isolation valve interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated ~~automatic~~ remote manual valve within 4 hours. This Action accomplishes the purpose of the ~~autoclosure function~~ interlock.

WC3.4-009 undelete

WC 3.4-009

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Q3.4.14-2

Performance of leakage testing on each RCS PIV ~~or isolation valve~~ used to satisfy Required Action A.1 ~~and Required Action A.2~~ is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

redline Q3.4.Gen-1 redline Q3.4.Gen-1

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. ~~8~~ 7) as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. ~~7~~ 6), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

WC 3.4-010

WC 3.4-010

(continued)

BASES

SURVEILLANCE RREQUIREMENTS

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

~~design pressure of 600 psig. The interlock setpoint that~~ prevents the valves from being opened is set so the actual RCS pressure must be < ~~442~~ 425 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. This SR is not required to be performed when the RHR suction isolation valves are open to satisfy LCO 3.4.12.

~~These SRs are modified by Notes allowing the RHR autoclosure function to be disabled 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. ~~USAR, Chapter 16.~~ WC 3.4-010
6 ~~7.~~ ASME, Boiler and Pressure Vessel Code, Section XI.
7 ~~8.~~ 10 CFR 50.55a(g).

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	YES
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	YES
5.5-8	CA, CP, DC, WC	YES
5.5-9	CA, DC, WC	YES
5.5-10	CA, WC	YES
5.5-11	DC	NA
5.5-12	WC	YES
5.5-13	DC	NA
5.5-14	CP	NA
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	YES
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	NA
TR 5.0-006	CA, CP, DC, WC	YES
WC 5.0-ED	WC	YES
WC 5.0-001	WC	YES
WC 5.0-002	WC	YES
WC 5.0-003	WC	YES
WC 5.0-004	WC	YES
WC 5.0-005	WC	YES
WC 5.0-006	WC	YES

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:** DC, CP, WC, CA

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, 6-6, 6-17, 6-18, 6-20, 6-23, 6-24
Encl. 3A	2, 3, 5, 6, 7, 9, 10
Encl. 3B	2, 5, 8
Encl. 4	1, New LS-5
Encl. 5A	Traveler Status page, 5.0-3, 5.0-4, 5.0-5, 5.0-6, 5.0-10, 5.0-11, 5.0-31, 5.0-37, 5.0-38
Encl. 6A	1, 2, 4
Encl. 6B	1, 2, 4, 5

Unit Staff (Continued)

b. ~~At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the Unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Operator shall be in the control room;~~ 01-05-A

c. An individual from the Health Physics Groups*, qualified in radiation protection procedures, shall be on site when fuel is in the reactor;

d. ~~ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;~~ 01-03-A

01-08-LG

e. ~~A site Fire Brigade of at least 5 members* shall be maintained onsite 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; and~~ ~~Manager~~ WC 5.0-003

f. Administrative procedures shall be developed and implemented to limit the working hours of Unit Staff who perform safety related functions; e.g., Senior Operators, Operators, Health Physicists, 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).~~ INSERT 2-2a 1-09-A Q 5.2-1

g. The Superintendent Operations or Manager Operations shall hold a senior reactor operator license.

*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.

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.

~~6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)~~

01-04-LG

~~FUNCTION~~

~~6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, REPORTABLE EVENTS and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Chairman Nuclear Safety Review Committee.~~

~~COMPOSITION~~

~~6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field.~~

~~RESPONSIBILITIES~~

~~6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.~~

~~RECORDS~~

~~6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to Chairman Nuclear Safety Review Committee.~~

6.2.4 SHIFT TECHNICAL ADVISOR

An individual**

~~The Shift Technical Advisor (STA)**~~ shall provide technical support to the ~~Shift Supervisor~~ unit operations shift crews in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the Unit.

1-15-A

Q 5.2-1

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 with the following exceptions:

a. Licensed Operators and Senior Operators shall meet or exceed the qualifications of ANSI/ANS 3.1-1981 as endorsed by Regulatory Guide 1.8, Revision 2.

~~*Not responsible for sign-off function.~~

01-04-LG

This

nd elete

~~**The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Supervisor or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.~~

Manager — WC 5.0-003

~~01-46-LG~~

1-15-A

Q 5.2-1

PROCEDURES AND PROGRAMS (Continued)

e. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS Of THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the **functional capability** ~~operability~~ of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with methodology in the ODCM, **02-05-A**

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to **10 times the concentration values in** 10 CFR Part 20, Appendix B, Table **2** ~~II~~, Column 2, **to 10 CFR 20.1001-20.2402 .** **02-14-M**

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative ~~and projected~~ dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM. ~~at least every 31 days.~~ **Determination of projected dose contributions for radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.** **02-07-A**

6) Limitations on the **functional capability** ~~operability~~ and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50, **02-05-A**

from the site / at or

7) Limitations of the dose rate resulting ~~from~~ radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY ~~conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,~~ ~~to the following:~~ shall be in accordance with the following: Q 5.2-1 **02-18-A**

a. For noble gases: Less than or equal to a dose rate of 500 mrem/year to the whole body and less than or equal to a dose rate of 3000 mrem/year to the skin, and

b. For Iodine-131, Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/year to any organ.

PROCEDURES AND PROGRAMS (Continued)

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

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 40 CFR Part 190. 2-1B-A Q 5.2-1 02-06-A

11) The provisions of Technical Specification 4.0.2 and 4.0.3 are applicable to the Radiological Effluent Controls Program. 02-22-A Q 5.2-1

~~f. Radiological Environmental Monitoring Program~~

~~A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:~~

~~1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.~~

~~2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and the modifications to the monitoring program are made if required by the results of this census, and~~

~~3) Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

g. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria, in accordance with the applicable ASTM Standard. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. an API gravity or an absolute specific gravity within limits,

2. a flash point within limits for ASTM 2D fuel oil,

ANNUAL REPORTS (Continued)

~~b. Documentation of all challenges to the PORVs or safety valves.~~ 03-18-A LS Q 5.2-1

~~c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~ 03-04-A

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

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 a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. 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. 03-07-A

03-06-A

~~ANNUAL~~ RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.7 The ~~Annual~~ Radioactive Effluent Release Report covering the operation of the unit during the previous calander year of operation shall be submitted ~~before~~ **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 (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.I of Appendix I to 10 CFR Part 50.

MONTHLY OPERATING REPORT

6.9.1.8 Routine reports of operating statistics and shutdown experience ~~**including documentation of all challenges and failures to the PORVs of pressurizer safety valves,**~~ shall be submitted on a monthly basis ~~to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the NRC Regional Office~~, no later than the 15 th of each month following the calendar month covered by the report. ~~03-18-A~~ Q 5.2-1 03-08-A

~~6.11 RADIATION PROTECTION PROGRAM~~ 03-10-LG

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

6.12 HIGH RADIATION AREA 03-11-A

6.12.1 Pursuant to ~~Paragraph 20.203(c)(5) of~~ 10 CFR Part 20, in lieu of the ~~"control device" or "alarm signal" required by Paragraph 20.203(c)(2)~~ requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is greater than 100 mrem/hr **but** equal to or less than 1000 mR/h at ~~45 cm (18 in.)~~ 30 cm (12 in.) ~~from the radiation source or from any surface which the radiation penetrates~~ 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 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 equal to or less than 1000 mR/h at 30 cm (12 in.), 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: Q 5.2-1 Q 5.2-1

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 personnel 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 Superintendent Radiation Protection~~ ~~health physics supervision~~ in the RWP. Manager Chemistry/Radiation Protection Q 5.2-1 03-11-A WC 5.0-003

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than or equal to 1000 mR/h at ~~45 cm (18 in.)~~ 30 cm (12 in.) ~~from the radiation source or from any surface which the radiation penetrates~~ shall be provided with locked doors or continuously guarded to prevent ~~unauthorized~~ ~~inadvertant~~ entry, and the keys shall be maintained under the administrative control of the ~~Shift Supervisor/Supervising Operator~~ on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel 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 personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area. ← undelete 03-11-A 03-20-LS-3 ~~03-19-A~~ Q 5.2-1

Shift Manager/Control Room Supervisor — WC 5.0-003

HIGH RADIATION AREA (Continued)

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h at 30 cm (12 in.) that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking or that is not continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device. Q 5.2-1 03-11-A 03-20-LS-3

~~6.13 PROCESS CONTROL PROGRAM (PCP)~~ 03-12-LG

~~Changes to the PCP:~~

~~a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2.o. This documentation shall contain:~~

~~1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and~~

~~2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.~~

~~b. Shall become effective after review and acceptance by the PSRC and the approval of the Plant Manager~~

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

a. The ODCM shall contain the methodology and parmeters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and 02-09-A

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Specification 6.9.1.6 and Specification 6.9.1.7. 02-09-A

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained ~~as required by Specification 6.10.2.o~~. This documentation shall contain: 03-09-LG

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.~~106~~1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent dose, or setpoint calculations. 02-09-A

b. Shall become effective after ~~review and acceptance by the PSRC and~~ the approval of the Plant Manager. 02-13-LG

CHANGE NUMBER	NSHC	DESCRIPTION
		performed in accordance with guidance in an NRC letter dated October 25, 1993, from William T. Russell to the chairpersons of industry owners groups. Furthermore, sections which contain details of procedure review and approval, including temporary changes, are contained in regulations and standards (10 CFR 50.36; 10CFR 50, Appendix B, Criterion II and V; ANSI N18.7-1976; and N45.2-1971). Therefore, duplication of these requirements in the CTS is not required.
01-05	A	The requirement for the presence of a Reactor Operator (RO) or an SRO in the control room is deleted from the TS since the requirement is consistent with and duplicative of the manning requirement in 10 CFR 50.54(m)(2)(iii). Deletion of the CTS requirements does not change the manning requirements and is therefore considered an administrative change.
01-06	LG	The details regarding the minimum shift crew requirements have been removed from the CTS because they are redundant to 10 CFR 50.54(k), (l), and (m) with the exception of the requirement for non-licensed operators. The corresponding ITS section 5.2.2b requires meeting the requirements of these regulations which specify the shift complement regarding licensed operators for all modes of operation. The minimum shift crew requirements will be moved to a licensee controlled document.
01-07	LG	Revises Section 6.2.2a, Unit Staff Organization, to reflect the non-licensed operator staffing requirements for a single unit site consistent with the NUREG-1431, Rev.1 requirements. The minimum shift crew composition as described in Table 6.2-1 has been moved to a licensee controlled document and provides requirements for the minimum number of non-licensed operators necessary for plant operations. This proposed change is consistent with NUREG-1431, Rev. 1.
01-08	LG	Move the fire brigade requirements to a licensee controlled document. Moving these requirements is consistent with NUREG-1431, Rev. 1. These requirements can be found in BTP ASB 9.5-1 and their duplication in the ITS is not required.
01-09	A	~~Not used~~. INSERT 3A-2a — Q 5.2-1

1-09-A 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.

01-16 LG Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B.

TR 5.0-005

CHANGE NUMBER	NSHC	DESCRIPTION
01-10	M	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
01-11	A	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
01-12	A	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
01-13	A	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
01-14	A	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
01-15	A	INSERT 3A-3a Q 5.2-1
02-01	A	CTS Section [6.6.1a] for Reportable Event actions has been deleted from the CTS. This section only repeats the regulatory reporting requirements defined in 10 CFR 50.72 and 10 CFR 50.73, and is unnecessary in the TS. Deletion of this section from CTS does not impact safety because it is redundant to the regualtions cited and is therefore acceptable.
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. [The 14 day Safety limit Violation report in the CTS is not required since 50.73 would require 30 day Licensee Event Report.] Since the plant must meet the applicable requirements contained in the regulations, sufficent regulatory controls are maintained to allow removing these duplicate regulatory requirements from the current TS. The notification requirement to executive 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 the TS. These changes are consistent with NUREG-1431 and traveler TSTF-5.
02-03	A	The implementation procedure requirements related to [the security plan, the emergency plan,] process control programs and radiological environmental and offsite dose calculation programs are deleted from the CTS consistent with NUREG-1431. These types of procedures are either required by Regulatory Guide 1.33, Rev. 2, Feb.1978

1-15-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.

CHANGE NUMBER	NSHC	DESCRIPTION
02-09	A	The description of the [ODCM] (or equivalent programs and procedures) was revised to be consistent with NUREG-1431. The [ODCM] description is also revised to reflect new 10 CFR Part 20 requirements.
02-10	M	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
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	This change moves the Emergency Diesel Generator Reliability Program requirement to a licensee controlled document. Moving this program is consistent with NUREG-1431.
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	Per GL 89-01, concentrations of radioactive material releases in liquid effluents to unrestricted area shall conform to 10 times the concentration values in Appendix B, Table 2, Column 2 of 10 CFR 20.1001-20.2401. ~~Proposed traveler pending.~~ INSERT 3A-5a Q 5.2-1
02-15	LG	CTS Section [6.6.1.b] contains requirements for the plant review and submittal of a reportable event. This information is to be moved to a licensee controlled document. Given that these reviews and submittal of results are required following the event without a specified completion time, the requirements are not necessary. The moving of this information maintains consistency with NUREG-1431, Rev. 1.
02-16	A	To maintain consistency with the Bases for TS 3/4.8.1, 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

INSERT 3A-5a Q 5.2-1

This limitation provides reasonable assurance that the levels of radioactive materials in bodies of water in unrestricted areas will result in exposures within (1) the Section II.A design objectives of Appendix I to 10 CFR Part 50 and (2) restrictions authorized by 10 CFR 20.1301(e). These changes are consistent with NUREG-1431 as modified by TSTF-258.

CHANGE NUMBER	NSHC	DESCRIPTION
		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 of NUREG-1431.
02-17	LS-1	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
02-18	A	Revises the Radioactive Effluent Controls Program dose rate limits released to areas beyond the site boundary to reflect new 10 CFR Part 20 requirements. ~~Consistent with NRC letter dated 7/28/95 (Christopher I. Grimes to Owners Groups). Additionally, the NRC issued a draft Generic Letter in 1993 which proposed changes to the Standard Technical Specifications. This change is consistent with the draft Generic Letter and NUREG-1431, Rev. 1 as amended by a proposed traveler to reflect changes consistent with 10 CFR Part 20.~~ INSERT 3A-6a
02-19	LS-2	Consistent with NUREG-1431, 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. The fuel properties that can have an immediate detrimental impact on diesel combustion, (i.e., API gravity, kinematic viscosity, flash point and appearance) are verified prior to addition to the storage tank. The "other properties" may be analyzed after addition to the tank. The 31 day verification interval for these properties is acceptable because the fuel properties of interest, even if they are not within their stated limits, would not have an immediate affect on diesel generator operation. The CTS 30 day verification interval was probably chosen because it was a convenient time interval for sending the sample and receiving the results from the laboratory selected for testing. NUREG-1431 has selected a 31 day testing interval. The 1 day increase in the interval would not have a significant affect on the acceptability of the diesel fuel oil.
02-20	A	Consistent with NUREG-1431, Rev. 1 and traveler TSTF-118, add the statement that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program frequencies. These sentences provide consistency with the current application of these requirements as provided in ISTS 5.5.6 and ISTS 5.5.11. Amendment [101] moved the Diesel Fuel Oil Testing Program to Section 6.0 of the current TS but did not include a statement that the provisions of SR 3.0.2 and SR 3.0.3 are applicable. SR 3.0.2 and SR 3.0.3 are applicable to the surveillances

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.

02-23 LG Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B) DC 5.0-004

CHANGE NUMBER	NSHC	DESCRIPTION
02-21	A	INSERT 3A-7b WC 5.0-004 which reference these programs, and therefore, the lack of an applicability statement in the Programs introduces confusion.
02-22	A	INSERT 3A-7a Q5.2-1
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	The requirement to submit a Startup Report is deleted from the CTS to be consistent with NUREG-1431. This report required no staff approval and was submitted after the fact and is therefore not required to ensure safe plant operation. The approved 10 CFR 50, Appendix B, QA Plan, and USAR startup testing program provides assurance that the affected activities are adequately performed and that appropriate corrective actions, if required, are taken.
03-03	A	The Annual Reports section is revised to be consistent with NUREG-1431 and traveler TSTF-152. Names and formats are revised consistent with NUREG-1431. Also, revises the annual report section to reflect the new 10 CFR Part 20 requirements and associated recommended changes noted in NRC letter dated July 28, 1995, "Changes to Technical Specifications Resulting from 10CFR20 and 50.36a Changes." (From Christopher I. Grimes to Owners Groups Chairs).
03-04	A	The requirement to report specific activity limit violations is deleted consistent with NUREG-1431. This report is a history of Reactor Coolant System (RCS) specific activity Limiting Conditions for Operation (LCO) entries. GL 83-43 and revised reporting requirements in the regulations intended that LCO entry reports no longer be required. The reporting requirements in regulations cover situations such as seriously degraded barriers (fuel failure). Therefore, every violation of the RCS specific activity LCO need not be reported. Serious degradation of a fission product barrier, among other more serious events are required to be reported by 10 CFR 50.73. This change is administrative in that it only affects reports and do not affect plant operations.
03-05	A	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
03-06	A	CTS [6.9.1.7], "Annual Radioactive Effluent Release Report" and CTS [6.14.c] is revised consistent with NUREG-

INSERT 3A-7a Q 5.2-1

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 surveillances. 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.

INSERT 3A-7b WC 5.0-004

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 are completed during the ninth refueling outage. Since the footnote is a one-time exception and has been satisfied, the footrote is no longer applicable and can be deleted.

CHANGE NUMBER	NSHC	DESCRIPTION
		required by 10 CFR 20.1101(c). The CTS is redundant to requirements in the regulations and thus is deleted.
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 and add clarification and conform with NUREG-1431 and RG 8.3.8.~~ INSERT 3A-9b Q5.2-1
03-12	LG	The Process Control Program (PCP) section is proposed to be moved outside the CTS consistent with NUREG-1431. The PCP implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71. Therefore, relocation of the description of the PCP from the CTS does not affect the safe operation of the facility. The PCP will be adequately described in licensee controlled documents.
03-13	M	The following report[s] will be added to the ITS Administrative Controls section: "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" [and "Post Accident Monitoring (PAM) Report."] The PTLR is more restrictive because it is not currently required. This change is also described in the description of changes for CTS Section 3/4.4 (PTLR). [The PAM Report is already required per TS 3/4.3.].
03-14	M	Shutdown Margin values would be moved to the Core Operating Limits Report (COLR) traveler TSTF-9. In addition, moderator temperature coefficient limits would also be moved to the COLR. The addition of these specifications to the COLR is considered to be more restrictive.
03-15	M	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
03-16	A	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
03-17	A	Deletes the methodology section references in the COLR. These references are adequately defined by the analytical methods themselves as approved by the NRC and it is redundant to repeat the information in the ITS. This change is consistent with NUREG-1431.
03-18	~~A~~ LS	Q5.2-1 ~~CTS 6.9.1.8, "Monthly Operating Report" is revised to incorporate the reporting of all challenges and failures to the PORVs or pressurizer safety valves. The requirement to report these challenges and failures was~~ INSERT 3A-9a

INSERT 3A-9a Q 5.2-1

03-18-LS-5 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.

INSERT 3A-9b Q 5.2-1

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.

CHANGE NUMBER	NSHC	DESCRIPTION
		Q5.2-1
		~~required on an annual basis. This change is consistent with NUREG-1431, Rev. 1.~~
		Q5.2-1
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 8.3.8. 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.~~ Not Used.
03-20	LS-3	Consistent with NUREG-1431, 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. This option is acceptable because it provides access control to these areas commensurate with the other methods already permitted by the CTS. (i.e, locked doors, or barricades with posted signs and flashing lights for isolated areas without enclosures.)

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. USAR Chapter 13.	Yes. 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. INSERT 3B-2a — Q 5.2-1	NA	NA	NA	NA
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. Callaway 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. Callaway is a single unit site.
01-12 A	Deletes the Commanche 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

01-15 A INSERT 3B-2b — Q 5.2-1

01-16 LG INSERT 3B-2c — TR 50-005

INSERT 3B-2a

Q 5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
01-09 A	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.	Yes	Yes	Yes	Yes

INSERT 3B-2b

Q 5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE 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

INSERT 3B-2c

TR 5.0-005

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
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

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 ITS SR 3.8.3.3. TSTF-237 Q5.5-1	No. Not in CTS.	No. Not in CTS.	Yes	Yes
02-17 LS-1	"Reactor Coolant Pump Flywheel" is being revised consistent with WOG-85. The proposed changes provide an exception to the examination requirements in Regulatory Guide 1.14, Revison 1, "Reactor Coolant Pump Flywheel Integrity."	Yes	No. See Section 3/4.4, CN 10-03-LS	No. LAR submitted 12/3/96.	Yes
02-18 A	Revise the Radioactive Effluent Controls Program dose rate limits to reflect changes to 10 CFR Part 20. ~~a draft Generic Letter and a proposed traveler.~~	~~No. Already in CTS.~~ Yes Q5.2-1	~~No. Already in CTS.~~ Yes	Yes	Yes Q5.2-1
02-19 LS-2	The surveillance interval for verifying that other properties are within 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 (See CN 01-60-LS24).	No. Addressed in 3/4.8 (See CN 01-60-LS24).	Yes	Yes
02-20 A	Add the provisions of Specification SR 3.0.2 and SR 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
03-01 A	The method for submitting all reports is revised to be in accordance with 10 CFR 50.4.	Yes	Yes	Yes	Yes

02-21 A INSERT 3B-5b WC 5.0-004

02-22 A INSERT 3B-5a Q5.2-1

02-23 LG INSERT 3B-5C DC 5.0-004

INSERT 3B-5a

Q 5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE 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

INSERT 3B-5b

WC 5.0-004

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
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 are 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

INSERT 3B-5c

DC 5.0-004

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
02-23 LG	DCPP Administrative Programs, CTS 6.8.4.d, "Backup Method for Determining Subcooling Margin," and 6.8.4.f, "Containment Poar and Turbine Building Cranes," were evaluated for reloaction outside the TS to a licensee-controlled document consiste with 10 CFR 50.36 screening criteria.	Yes	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-15 M	Adds refueling boron concentration limits to COLR.	Yes	Yes	No. Already in CTS.	Yes
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 Wolf Creek Monthly Operating Report.~~ INSERT 3B-8a	~~No~~	~~No~~	~~Yes~~	~~No~~ Q5.2-1
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 Q5.2-1
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.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
03-18 LS-5	The CTS requirement to provide documentation of all challenges to the PORV's or safety valves is deleted.	Yes	Yes	Yes	Yes

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)
CONTENTS

INSERT 4-a

Q 5.2-1

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

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

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATION

NSHC LS-5
(continued)

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 SECTION 5.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-9, Rev. 1	Incorporated		NRC approved.
TSTF-37, Rev. 1	Incorporated	5.6-2	DCPP only.
TSTF-52	Incorporated	5.5-4	Incorporated draft Rev. 1 per Q3.6.1-6 Q3.6.1-6
TSTF-65 Rev. 1	~~Not~~ Incorporated	~~NA~~ 5.2-9	~~Not~~ NRC approved ~~as of traveler cutoff date.~~ CPSES only. TR 5.0-005
TSTF-106, Rev. 1	Not Incorporated	NA	Retain CTS.
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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	5.6-5	NRC approved. TR 5.0-003
WOG-72	Incorporated	5.5-13	
TSTF-237 ~~WOG-85~~	Incorporated	5.5-14	Q 5.5-2
~~Proposed Traveler~~ TSTF-258	Incorporated	5.2-2, 5.5-1, 5.2-3, 5.2-6, 5.3-2, 5.5-16, 5.6-6, 5.7-1,	~~WOG mini-group Action Item 147.~~ Q 5.2-1

5/15/97

5.2 Organization

5.2.2 Unit Staff (continued)

assigned ~~for each control room from which a reactor is operating~~ when the unit is in MODES 1, 2, 3, or 4. 5.2-4

~~[Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.]~~ B-PS

~~b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.~~ 5.2-2

~~e~~b. Shift crew composition may be one less than the minimum requirement of 10 CFR 50.54(m)(2)(I) and 5.2.2.a and 5.2.2.~~g~~f 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. Ed

~~d~~c. An individual from the Health Physics Group qualified in radiation protection procedures ~~Health Physics Technician~~ shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position. B-PS

~~e~~d. Administrative procedures shall be developed and implemented to limit the working hours of ~~unit staff~~ personnel who perform safety related functions (e.g., licensed Senior Reactor Operator (SROs), licensed Reactor Operator (ROs), health physic~~ists~~s technicians, ~~auxiliary operators~~ nuclear station operators, and key maintenance personnel). B-PS ~~Ed~~ 5.2-3 Q5.2-1

E 3

~~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~~

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

~~maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:~~

~~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.~~

~~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.~~

INSERT 5.0-4

5.2-3
Q5.2-1

~~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).~~

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

(continued)

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.

An individual

unit operations shift crew

5.2 Organization

Q 5.2-1

gf. ~~The Shift Technical Advisor (STA)~~ shall provide advisory technical support to the ~~Shift Supervisor (SS)~~ in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. ~~In addition, the STA shall~~ ~~The STA~~ position shall be manned in MODES 1, 2, 3 or 4, unless the ~~SS~~ or the individual with a Senior Operator License meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. 5.2-6

~~Shift Manager~~

WC 5.0-003

This

Shift Manager

WC 5.0-003

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.~~

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 with the following exceptions: ~~[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 B-PS

5.3.1.1 Licensed Operators and Senior Operators shall meet or exceed the qualifications of ANSI/ANS 3.1-1981 as endorsed by Regulatory Guide 1.8, Revision 2 and 10 CFR Part 55. 5.3-1

Manager Chemistry / WC 5.0-003

5.3.1.2 The position of ~~Superintendent~~ Radiation Protection shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 for a Radiation Protection Manager.

5.3.1.2 The position of Manager Operations shall hold or have previously held a senior reactor operator license for a similar unit (PWR).

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m). 5.3-2 Q 5.2-1 and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 times the concentration values in ~~10 CFR 20~~, Appendix B, Table 2, Column 2, to 10 CFR 20.1001-20.2402; 5.5-1

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative ~~and projected~~ dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM. ~~at least every 31 days~~. Determination of projected dose contributions for radioactive effluents in accordance with the methodology in the ODCM at least every 31 days; 5.5-13

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;

from the site
at or

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming ~~to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;~~ to the following: Q5.2-1 5.5-1

shall be in accordance with the following:

1. For noble gases: ~~Less than or equal to~~ a dose rate ~~of~~ 500 mrem/yr to the whole body and ~~less than or equal to~~ a dose rate ~~of~~ 3000 mrem/yr to the skin, and

≤
≤

2. For Iodine-131, for Iodine-133, for tritium, and for all 5.5-1

(continued)

5.5 Programs and Manuals

radionuclides in particulate form with half-lives greater than 8 days: ~~Less than or equal to~~ a dose rate ~~of~~ ≤ 1500 mrem/yr to any organ. 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 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

j. 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 40 CFR 190. 5.5-1 Q5.2-1

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 Q5.2-1

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR, Section 3.9(N), cyclic and transient occurrences to ensure that components are maintained within the design limits. 8-PS

(continued)

5.6 Reporting Requirements

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 15th of each month following the calendar month covered by the report. B 5.6-6 Q5.2-1

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

~~The individual specifications that address core operating limits must be referenced here.~~ B-PS

1. Specification 3.1.3: Moderator Temperature Coefficient (MTC),

2. Specification 3.1.5: Shutdown Bank Insertion Limits,

3. Specification 3.1.6: Control Bank Inserti[illegible] Limits

4. Specification 3.2.3: Axial Flux Differe[illegible],

5. Specification 3.2.1: Heat Flux Hot Channel Factor, $F_Q(Z)$,

6. Specification 3.2.2: Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$),

7. Specification 3.9.1: Boron Concentration, and

8. SHUTDOWN MARGIN for Specification 3.1.1 and 3.1.4, 3.1.5, 3.1.6, and 3.1.8.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

~~Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.~~ B-PS

(continued)

INSERT 5.0-37 — Q5.2-1

5.7-1

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area B-PS

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 (12 in.), shall be barricaded and conspicuously posted as a high radiation area and entrance there to shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technicians) 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

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~~ health physics supervision in the RWP. 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 (12 in.) shall be provided with locked or continuously guarded doors to prevent ~~unauthorized~~ inadvertent entry and the keys shall be maintained under the administrative control of the Shift Supervisor/Supervising Operator 5.7-1 5.7-2 PS

(continued)

Q 5.2-1

5.7 High Radiation Area

5.7.2 (continued)

~~Shift Foreman~~ on duty or health physics supervision. Doors shall remain locked except during periods of 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 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.

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr at 30 cm (12 in.), 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

INSERT 5.0-37

Q5.2-1

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 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 and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

(continued)

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)

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 1 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: Shift Manager/Control Room Supervisor WCSD-003

   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.

or health physics supervision

B-PS

(continued)

5.7 High Area Radiation Area

5.7.2 <u>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:</u> (continued)

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 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

(continued)

INSERT 5.0-37 continued

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)

(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 "As 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 Q5.2-1

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 Q5.2-1

(continued)

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	Revises Section 5.1.1 to maintain current TS. The Plant Manager does not currently approve prior to implementation each proposed test, experiment or modification to systems or equipment that affect nuclear safety. The design process includes design verification by qualified persons to assure that the design is adequate and meets specified design input. Design/configuration changes and test procedures that require a written evaluation under 10 CFR 50.59 are reviewed by the Plant Safety Review Committee prior to implementation. There are adequate administrative controls for review of proposed tests, experiments or modifications so that review by the Plant Manager is not necessary.
5.1-2	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
5.2-1	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
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 change is consistent with traveler TSTF-~~121~~ 258 Q5.2-1
5.2-3	~~Not used~~. INSERT 6A-1a Q5.2-1
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.
5.2-5	Not used.

5.2-3 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.

5.2-8 Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B) DC 5.0-002

5.2-9 Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B) TR 5.0-005

CHANGE NUMBER	JUSTIFICATION
5.2-6	This ~~change~~ ~~revises~~ Section ~~5.2.2f~~ ~~to~~ ~~describe~~ ~~the~~ ~~current~~ [TS] ~~for~~ the Shift ~~Technical~~ ~~Advisor~~ ~~(STA)~~. INSERT 6A-2a Q5.2-1
5.2-7	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
5.3-1	This change revises Section 5.3.1 to be consistent with current TS regarding plant staff qualifications and training.
5.3-2	INSERT 6A-2b Q5.2-1
5.5-1	These changes revise Section 5.5.4, "Radioactive Effluent Controls Program," to reflect new 10 CFR Part 20 requirements. ~~A proposed traveler is being generated to reflect a revision to NUREG-1431 consistent with 10 CFR Part 20.~~ INSERT 6A-2c Q5.2-1
5.5-2	This change 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.
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. [To maintain consistency with the Bases for 3.8.3.3, specific changes to the program description are for sampled properties of new fuel oil from "within limits" to "analyzed' within 31 days following sampling and addition of the fuel oil to storage tanks. This wording more clearly defines that within 31 days following initial new fuel oil sample, the fuel oil is analyzed to establish that the other priorities are met.]
5.5-6	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
5.5-7	Not applicable to WCGS. 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

INSERT 6A-2a Q 5.2-1

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.

INSERT 6A-2b Q 5.2-1

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.

INSERT 6A-2c Q 5.2-1

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.

5.5-18 Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B) DC 5.0-003

CHANGE NUMBER	JUSTIFICATION
5.5-19	INSERT 6A-4g — Q 5.5-7
5.5-15	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
5.5-16	INSERT 6A-4a — Q 5.2-1
5.6-1	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
5.5-17	INSERT 6A-4f — CA 5.0-003
5.6-2	This change deletes the Emergency Diesel Generator 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.
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 10 CFR 20 and 50.36a Changes" (From Christopher I. Grimes to Owners Groups Chairs). This change is consistent with traveler TSTF-152.
5.6-5	[ ] PORV lift settings are referenced in the PTLR section per ~~WOG-67, Rev. 1.~~ TSTF-233 — TR 5.0-003
5.6-6	INSERT 6A-4b — Q 5.2-1
5.7-1	~~This change revises High Radiation Area to incorporate changes consistent with [10 CFR 20.1601]. Specifically, distances from the radiation source are noted.~~ INSERT 6A-4c — Q 5.2-1
5.7-2	~~This change revises "unauthorized" to "inadvertent" in the High Radiation Area section to reflect the NRC's position as stated in Regulatory Guide 8.38, Section 1.5 regarding physical barriers for High Radiation Areas. This is consistent with traveler TSTF-167.~~ INSERT 6A-4d — Q 5.2-1
5.7-3	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
5.7-4	INSERT 6A-4e — Q 5.2-1

INSERT 6A-4a Q 5.2-1

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 surveillances. 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.

INSERT 6A-4b Q 5.2-1

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.

INSERT 6A-4c Q 5.2-1

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.

INSERT 6A-4d Q 5.2-1

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 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 maintaining appropriate radiation worker practices.

INSERT 6A-4e Q 5.2-1

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.

INSERT 6A-4f CA 5.0-003

ITS Section 5.5.10 is being revised consistent with CTS 8.8.4.c. The proposed change deletes the phrase "and low pressure turbine disc stress corrosion cracking" from the ITS to be consistent with the practices of the CTS which do not have this requirement for the Secondary Water Chemistry Program.

INSERT 6A-4g Q 5.5-7

5.5-19 CTS surveillance requirements 4.7.6.3c.3) and 4.9.13b.3) for safety-related ventilation system filter adsorber units include the requirement to measure flow rates within specified values, while imposing an artificial differential pressure, during system operation, when tested in accordance with ANSI N510-1980. This flow rate testing is to be performed at least once per 18 months, after any structural maintenance on the HEPA filter or charcoal adsorber housings, or following painting, fire, or chemical release in any ventilation zone communicating with the system. Therefore, the CTS surveillance requirements are incorporated into the ITS.

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

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.1-1	Revises Section 5.1.1 to maintain WCNOC current technical specifications (TS). 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).	Yes	Yes	Yes	Yes
5.2-3	~~Not used~~. INSERT 6B-1a	~~NA~~	~~NA~~	~~NA~~	~~NA~~ Q5.2-1
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 site.	Yes	Yes
5.2-5	Not used.	NA	NA	NA	NA
5.2-6	Revises section 5.2.2f to describe the current TS ~~for the STA~~ and to eliminate the title of Shift Technical Advisor (STA). Q5.2-1	Yes	Yes. LA 50/36 moved text to FSAR Section 13.1 which permits on-shift SRO to fill STA position.	Yes	Yes

INSERT 6B-1a

Q 5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.2-3	ITS Section 5.2.2d (ISTS 5.2.2e) is revised from specific working hour limits to administrative procedures to control working hours.	Yes	Yes	Yes	Yes

5.2-8 Insert 6B-2b - DC 5.0-002
5.2-9 Insert 6B-2c - TR 5.0-005

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
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.
5.3-1	Revises Section 5.3.1 to be consistent with current TS regarding plant staff qualifications and training.	Yes. LA 43/42.	Yes	Yes	Yes
5.5-1	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.~~ Q5.2-1	Yes	Yes	Yes	Yes
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
5.5-4	The Containment Leakage Rate Testing Program is added to the ISTS consistent with the current TS. The Containment Leakage Rate Testing Program is consistent with traveler TSTF-52.	Yes. LA 110/109	Yes. LAR 96-002.	Yes	Yes
5.5-5	Revises Section 5.5.13, "Diesel Fuel Oil Testing Program," to be consistent with current TS. The details of the method applied to this test are discussed in the associated SR 3.8.3.3 Bases.	Yes	Yes	Yes	Yes
5.5-6	Additional programs are added to the ITS (other than Containment Leakage Rate Testing Porgram discussed in CN 5.5-4).	~~Yes~~ No, no additional programs added DC 5.0-004	Yes	No. No additional programs added.	No. No additional programs added.

5.3-2 INSERT 6B-2a Q5.2-1

INSERT 6B-2a

Q 5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
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

INSERT 6B-2b

DC 5.0-002

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.2-8	Revises Section 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	No	No	No

INSERT 6B-2c

TR 5.0-005

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
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

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

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-12	The referenced frequencis for the tests listed in the Ventilation Filter Testing Program (VFTP) were evaluated as part of the 24 month fuel cycle program for DCPP (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 revsied consistent with traveler TSTF-237 ~~WOG-85~~ [and ~~a License Amendment Request submitted December 1, 1996~~ Amendment No. 106 dated June 24, 1997]. 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 Q5.5-2
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 Callaway CTS.	No	No	No	Yes
5.6-1	Revises Seciion 5.6.4, "Monthly Operating Report," to reflect a revised submittal date.	No. DCPP CTS 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.~~ TSTF-37, Rev 1.	Yes Q5.6-2	No. Not in CTS	No. Not in CTS.	No. Not in CTS.
5.6-3	Revises report dates in ITS 5.6.2, "Annual Radiological Environmental Operating Report" to be consistent with current TS.	Yes. Consistent with CTS and LA 78/77.	Yes. See LA 42/28.	Yes	Yes

5.5-16 INSERT 6B-4a — Q5.2-1

5.5-17 INSERT 6B-4b — CA 5.0-003

5.5-18 INSERT 6B-4c — DC 5.0-003

5.5-19 INSERT 6B-4d — Q5.5-7

INSERT 6B-4a

Q 5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
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

INSERT 6B-4b

CA 5.0-003

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-17	This change deletes the phrase "and low pressure turbine disc stress corrosion cracking" from ITS 5.5.10 to make the program consistent with CTS 6.8.4.c.	Yes	No	Yes	Yes

INSERT 6B-4c

DC 5.0-003

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-18	Revises DCPP Sections 5.5.9 and 5.6.10 to reflect License Amendment 124/122, dated March 12, 1998, which allows implementation of steam generator tube voltage based on repair criteria for ODSCC indications at tube to tube support plate intersections.	Yes	No	No	No

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-19	Wolf Creek CTS surveillance requirements 4.7.6.3c.3) and 4.9.13b.3) for safety-related ventilation system filter adsorber units include the requirement to measure flow rates within specified values, while imposing an artificial differential pressure, during system operation, when tested in accordance with ANSI N510-1980. The CTS surveillance requirements are incorporated into the ITS.	No	No	Yes	No

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

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
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 traveler TSTF-152.	Yes	Yes	Yes	Yes
5.6-5	DCPP LTOP arming and PORV lift settings are referenced in PTLR section per ~~WOG-57, Rev. 1~~ TSTF-233	Yes TR 5.0-003	Yes	Yes	Yes
5.7-1	Revises ~~High~~ Radiation Area ~~to incorporate changes~~ consistent ~~with~~ [10 CFR ~~20.1601~~]. INSERT 6B-5b	Yes Q5.2-1	Yes	Yes	Yes
5.7-2	~~Changes "unauthorized" to "inadvertant" in the High Radiation Area section to reflect the NRC's position as stated in RG 8.3.8, Section 1.5 regarding physical barriers for High Radiation Areas. This change is consistent with traveler TSTF-167.~~ INSERT 6B-5c	~~Yes~~	~~Yes~~	~~Yes~~	~~Yes~~ Q5.2-1
5.7-3	This change deletes the phrase "or that cannot be continuously guarded" from the ITS for Callaway to make them consistent with the CTS.	No	No	No	Yes

5.6-6 INSERT 6B-5a Q5.2-1

5.7-4 INSERT 6B-5d Q5.2-1

INSERT 6B-5a

Q 5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
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

INSERT 6B-5b

Q 5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.7-1	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

INSERT 6B-5c

Q 5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.7-2	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

INSERT 6B-5d

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE 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

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 5.3-1 **APPLICABILITY:** DC, WC, CA

REQUEST: ITS 5.3.1 (Wolf Creek, Callaway and Diablo Canyon)

Comment: Part 55 of Title 10 of the Code of Federal Regulations was revised in March 1987 to establish upgraded requirements for licensed reactor operators. NRC Regulatory Guide (RG) 1.8, Revision 2, April 1987, describes methods acceptable to the staff for complying with the revised rule. The Statements of Consideration for the Part 55 rule change state that, "Those facility licensees that have made a commitment that is less than that required by the new rules must conform to the new rules automatically." The staff is concerned some facilities continue to have technical specifications that reference older industry standards that may not fully meet the revised requirements of 10 CFR Part 55.

The staff previously considered that the standards applied through the industry's accreditation process were equivalent to the guidance contained in RG 1.8, Revision 2. However, the staff has recently found that current INPO guidance in this area is very general; only advising licensees to follow regulatory requirements. In RG 1.8, Revision 2, the NRC staff endorses, with conditions, certain parts of industry standard ANSI/ANS-3.1-1981 as an acceptable approach for complying with the qualification and training requirements of 10 CFR Parts 50 and 55. This endorsement applies to the positions identified as shift supervisor, senior operator, licensed operator, shift technical advisor, and radiation protection manager. For positions other than those identified, the RG finds acceptable the approach provided in ANSI N18.1-1971.

For Callaway, the ITS proposes to adopt the CTS which adopts ANSI/ANS 3.1-1978 for the unit staff (besides SROs, ROs and STAs) and RG 1.8, September 1975 for the radiation protection manager. For Wolf Creek, the ITS proposes to adopt the CTS which adopts ANSI/ANS 3.1-1978 for the unit staff (besides SROs and ROs) and RG 1.8, September 1975 for the radiation protection manager. For Diablo Canyon, the ITS proposes to adopt the CTS which adopts ANSI/ANS 3.1-1978 for the unit staff (besides the radiation protection manager) though it does makes a reference to ROs and SROs having to meet the minimum qualifications of Part 55.

Please describe how your commitment to an ANSI standard other than that endorsed by NRC RG 1.8, Revision 2 currently meets the requirements of 10 CFR Part 55, as discussed in the Statements of Consideration for the rule change and would meet those requirements with the ITS as proposed.

FLOG RESPONSE: Callaway, Wolf Creek, and Diablo Canyon believe that this question is outside the scope of the ITS conversion process because it is a generic industry question. The NRC's question regarding compliance with 10 CFR Part 55 should be discussed on a generic basis. Therefore, it is proposed that this issue be resolved separate from the ITS conversion and the submitted ITS 5.3.1 remain as is which is consistent with the CTS.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

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

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:

Encl. 5A	Traveler Status page
Encl. 6A	3
Encl. 6B	4

INDUSTRY TRAVELERS APPLICABLE TO SECTION 5.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-9, Rev. 1	Incorporated		NRC approved.
TSTF-37, Rev. 1	Incorporated	5.6-2	DCPP only.
TSTF-52	Incorporated	5.5-4	Incorporated draft Rev. 1 per Q 3.6.1-6 Q 3.6.1-6
TSTF-65 Rev. 1	~~Not~~ Incorporated	~~NA~~ 5.2-9	~~Not~~ NRC approved ~~as of traveler cutoff date~~ CPSES only. TR 5.0-005
TSTF-106, Rev. 1	Not Incorporated	NA	Retain CTS.
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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	5.6-5	NRC approved. TR 5.0-003
WOG-72	Incorporated	5.5-13	
TSTF-237 ~~WOG-85~~	Incorporated	5.5-14	Q 5.5-2
~~Proposed Traveler~~ TSTF-258	Incorporated	5.2-2, 5.5-1, 5.2-3, 5.2-6, 5.3-2, 5.5-16, 5.6-6, 5.7-1,	~~WOG mini-group Action Item 147.~~ Q 5.2-1

CHANGE
NUMBER JUSTIFICATION

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.

5.5-9 Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).

5.5-10 Section 5.5.12c specifics a surveillance program to ensure that the quantity of radioactivity contained in all outside liquid radwaste tanks that are not surrounded by liners, dikes, or walls is less than the predetermined quantities. This change lists the tanks that the surveillance program is applicable to as is in the current TS. This change is a plant specific requirement consistent with the current TS.

5.5-11 The documents referenced for the testing frequency for the Ventilation Filter Testing Program (VTFP) do not provide frequencies for combined pressure drop tests or the heater power rating test. The current TS frequency is added for these two tests.

5.5-12 Not applicable to WCGS. 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 change is consistent with traveler WOG-72.

5.5-14 ITS Section 5.5.7 is being revised consistent with traveler ~~WOG-85~~ TSTF-237 [and ~~License Amendment Request submitted December 3, 1996~~ Amendment No. 106 dated June 24, 1997]. The proposed changes to ITS 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."

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-12	The referenced frequencis for the tests listed in the Ventilation Filter Testing Program (VFTP) were evaluated as part of the 24 month fuel cycle program for DCPP (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 TSTF-237	Section 5.5.7 is being revsied consistent with traveler ~~WOG-85~~ [and ~~a License Amendment Request submitted December 1, 1996~~] Amendment No. 106 dated June 24, 1997. 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 Q5.5-2
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 Callaway CTS.	No	No	No	Yes
5.6-1	Revises Section 5.6.4, "Monthly Operating Report," to reflect a revised submittal date.	No. DCPP CTS 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.~~ TSTF-37, Rev 1.	Yes Q5.6-2	No. Not in CTS	No. Not in CTS.	No. Not in CTS.
5.6-3	Revises report dates in ITS 5.6.2, "Annual Radiological Environmental Operating Report" to be consistent with current TS.	Yes. Consistent with CTS and LA 78/77.	Yes. See LA 42/28.	Yes	Yes

5.5-16 INSERT 6B-4a Q5.2-1
5.5-17 INSERT 6B-4b CA 5.0-003
5.5-18 INSERT 6B-4c DC 5.0-003

ADDITIONAL INFORMATION COVER SHEET

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

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 5.5-7 **APPLICABILITY:** WC

REQUEST: CTS 3.7.6 and Changes 10-15-LG and 10-17-A (Wolf Creek)

Comment: The CTS markup is inconsistent with the comments as nothing is lined out. Further, the deletions (at least as they are reflected in ITS 5.5.11) need a better explanation. Provide explanation.

FLOG RESPONSE: The CTS markup should have lined out references to the overall Pressurization System flow rate of 2,200 cfm ± 10% in two places, i.e., in CTS 4.7.6.c 1) and CTS 4.7.6.c 3).

As further justification for DOC 10-15-LG, the following paragraph is added: "The acceptability of deleting the overall Pressurization System flow rate is based on the system design. The design flow rate for each Pressurization System fan is 2,200 cfm as shown on FSAR/USAR Table 9.4-4. Air flow into the fan consists of the flow from the Pressurization System filter adsorber unit plus a much larger flow of recirculated air from the Control Building. Thus, the 2,200 cfm flow rate through the system has only an indirect effect on the flow rate through the filter adsorber unit. The relative flow rates from the filter adsorber unit and from the Control Building are established by performing a flow balance on the Pressurization System. The proposed surveillance testing in accordance with the Ventilation Filter Testing Program (ITS 5.5.11) will assure the required flow rate through the Pressurization System filter adsorber unit. Based on the above discussion, there is no need to specify an overall system flow rate in the ITS."

While evaluating this question, the need to further modify ITS 5.5.11 for WCGS was identified. The basis for this change is that CTS 4.7.6c.3) and 4.9.13b.3) requires measuring flow rate while imposing an artificial differential pressure (dP). This requirement is unique to WCGS among the FLOG plants. Therefore, this CTS requirement has been incorporated into the ITS as Section 5.5.11.f. New JFD 5.5-19 is initiated to address adding this requirement to the CTS.

ATTACHED PAGES:

Attachment No. 13, CTS 3/4.7 - ITS 3.7

Encl. 2	7-15
Encl. 3A	13

Attachment No. 17, CTS 6.0 - ITS 5.0

Encl. 5A	5.0-23
Encl. 6A	4
Encl. 6B	4

(New) Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP); 10-08-A

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by: 10-08-A

1) Verifying that the Control Room Emergency Ventilation System satisfies the in-place penetration and bypass leakage testing acceptance criteria; of less than 1% for HEPA filters and 0.05% for charcoal adsorbers and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm ±10% for the Filtration System and ~~2200 cfm ±10% for the Pressurization System with~~ 750 cfm ±10% ~~going through~~ for the Pressurization System filter adsorber unit; 10-08-A Q5.5-7 10-15-LG

2) Verifying, ~~within 31 days after removal~~, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and ~~[illegible]~~ 70% relative humidity, for a methyl iodide penetration of less than 2%; and 10-23-LS-13 10-08-A Q5.5-12

3) Verifying system flow rate of 2000 cfm ±10% at greater than or equal to 6.6 inches W.G. (dirty filter) for the Filtration System and ~~2200 cfm ±10%~~ 750 cfm ±10% at greater than or equal to 3.6 inches W.G. (dirty filter) for the ~~Pressurization System with 750 cfm ±10% going through the~~ Pressurization System filter adsorber unit during system operation when tested in accordance with ANSI N510-1980. 10-17-A Q5.5-7 10-15-LG

d. After every 720 hours of charcoal adsorber operation by verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and ~~[illegible]~~ 70% relative humidity, for a methyl iodide penetration of less than 2%; - Q5.5-2 10-23-LS-13 10-08-A

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.6 inches Water Gauge while operating the system at a flow rate of 2000 cfm ±10% for the Filtration System and less than 3.6 inches Water Gauge while operating the system at a flow rate of 750 cfm ±10% for the Pressurization System filter adsorber unit 10-08-A

2) Verifying that on an actual or simulated actuation ~~a Control Room Ventilation Isolation or High Gaseous Radioactivity test~~ signal, the system automatically actuates ~~switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks~~ 10-10-TR-1

CHANGE NUMBER	NSHC	DESCRIPTION
10-09	LS-27	This change deleted the ACTION for an OPERABLE ventilation train not being capable of being supplied from an emergency power source. Per the definition of OPERABLE in NUREG-1431, the ventilation system would be considered OPERABLE with either a NORMAL or EMERGENCY power source.
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 two 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.
10-12	LS-32	This change deletes the required STB for the 31 day test. Both trains will still be tested on a 31 day frequency. This change is acceptable based on the evaluation of the effectiveness of STB testing provided in NSHC LS-32.
10-13	LG	Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
10-14	A	The statement that LCO 3.0.4 is not applicable is deleted based upon the new ITS definition of LCO 3.0.4 which does not apply in MODES 5 and 6.
10-15	LG	The ventilation system flow rates would be moved to licensee controlled documents. These flow rates are established in conjunction with flow balancing of the ventilation systems. INSERT 3A-13a Q 3.5-7
10-16	LG	Not Applicable to WCGS. 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)
10-18	LS-36	Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
10-19	A	Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).

10-15-LG The acceptability of deleting the overall Pressurization System flow rate is based on the system design. The design flow rate for each Pressurization System fan is 2,200 cfm as shown on FSAR/USAR Table 9.4-4. Air flow into the fan consists of the flow from the Pressurization System filter adsorber unit plus a much larger flow of recirculated air from the Control Building. Thus, the 2,200 cfm flow rate through the system has only an indirect effect on the flow rate through the filter adsorber unit. The relative flow rates from the filter adsorber unit and from the Control Building are established by performing a flow balance on the Pressurization System. The proposed surveillance testing in accordance with the Ventilation Filter Testing Program (ITS 5.5.11) will assure the required flow rate through the Pressurization System filter adsorber unit. Based on the above discussion, there is no need to specify an overall system flow rate in the ITS.

5.5 Programs and Manuals

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 absorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, ~~and ASME N510-1989]~~ at the system flowrate specified below ~~[± 10%]~~. 5.5-11 B-PS

Emergency Ventilation System – WC 5.0-006

ESF Ventilation System	Delta P	Flowrate
Control Room Filtration System	6.6 in. W. G.	2000 cfm
Control Room Pressurization System	3.6 in. W. G.	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	4.7 in. W. G.	6500 cfm

B-PS

Emergency Ventilation System – WC 5.0-006

e. Demonstrate at least once per 18 months that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1975. 5.5-11 B-PS

Emergency Ventilation System - Pressurization

ESF Ventilation System	Wattage
Control Room	5 ± 1 kW
Auxiliary/Fuel Building	37 ± 3 kW

B-PS

Emergency Exhaust WC5.0-006

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

f. INSERT 5.0-23 Q5.5-7 5.5-19

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks ~~or fed into the offgas treatment system~~, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, Revision 0, July 1981, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Revision 2, July 1981, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures." B-PS B B-PS B-PS

(continued)

f. Demonstrate at least once per 18 months for each of the ESF systems that following the creation of an artificial Delta P across the combined HEPA filters, the prefilters, and the charcoal absorbers of not less than the value specified below (dirty filter conditions), that the flowrate through these flow paths is with ± 10% of the value specified below when tested in accordance with ANSI N510-1980.

ESF Ventilation System	Delta P	Flowrate
Control Room Filtration System	6.6 in. W.G.	2000 cfm
Control Room Pressurization System	3.6 in. W.G.	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	4.7 in. W.G.	6500 cfm

5.5-18 Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B) DC 5.0-003

5.5-19 INSERT 6A-4g Q 5.5-7

CHANGE NUMBER | JUSTIFICATION

5.5-15 Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).

5.5-16 INSERT 6A-4a Q 5.2-1

5.6-1 Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).

5.5-17 INSERT 6A-4f CA 5.0-003

5.6-2 This change deletes the Emergency Diesel Generator 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.

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 10 CFR 20 and 50.36a Changes" (From Christopher I. Grimes to Owners Groups Chairs). This change is consistent with traveler TSTF-152.

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

5.6-6 INSERT 6A-4b Q 5.2-1

5.7-1 ~~This change revises High Radiation Area to incorporate changes consistent with [10 CFR 20.1601]. Specifically, distances from the radiation source are noted.~~ INSERT 6A-4c Q 5.2-1

5.7-2 ~~This change revises "unauthorized" to "inadvertent" in the High Radiation Area section to reflect the NRC's position as stated in Regulatory Guide 8.38, Section 1.5 regarding physical barriers for High Radiation Areas. This is consistent with traveler TSTF-167.~~ INSERT 6A-4d Q 5.2-1

5.7-3 Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).

5.7-4 INSERT 6A-4e Q 5.2-1

INSERT 6A-4a Q 5.2-1

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 surveillances. 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.

INSERT 6A-4b Q 5.2-1

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.

INSERT 6A-4c Q 5.2-1

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.

INSERT 6A-4d Q 5.2-1

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 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 maintaining appropriate radiation worker practices.

INSERT 6A-4e Q 5.2-1

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.

INSERT 6A-4f CA 5.0-003

ITS Section 5.5.10 is being revised consistent with CTS 8.8.4.c. The proposed change deletes the phrase "and low pressure turbine disc stress corrosion cracking" from the ITS to be consistent with the practices of the CTS which do not have this requirement for the Secondary Water Chemistry Program.

INSERT 6A-4g Q 5.5-7

5.5-19 CTS surveillance requirements 4.7.6.3c.3) and 4.9.13b.3) for safety-related ventilation system filter adsorber units include the requirement to measure flow rates within specified values, while imposing an artificial differential pressure, during system operation, when tested in accordance with ANSI N510-1980. This flow rate testing is to be performed at least once per 18 months, after any structural maintenance on the HEPA filter or charcoal adsorber housings, or following painting, fire, or chemical release in any ventilation zone communicating with the system. Therefore, the CTS surveillance requirements are incorporated into the ITS.

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

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-12	The referenced frequencis for the tests listed in the Ventilation Filter Testing Program (VFTP) were evaluated as part of the 24 month fuel cycle program for DCPP (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 TSTF-237	Section 5.5.7 is being revsied consistent with traveler ~~WOG-85~~ [and ~~a License Amendment Request submitted December 1, 1996~~]. 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 Amendment No. 106 dated June 24, 1997	Yes	Yes	Yes Q5.5-2
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 Callaway CTS.	No	No	No	Yes
5.6-1	Revises Section 5.6.4, "Monthly Operating Report," to reflect a revised submittal date.	No. DCPP CTS 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.~~ TSTF-37, Rev 1.	Yes Q5.6-2	No. Not in CTS	No. Not in CTS.	No. Not in CTS.
5.6-3	Revises report dates in ITS 5.6.2, "Annual Radiological Environmental Operating Report" to be consistent with current TS.	Yes. Consistent with CTS and LA 78/77.	Yes. See LA 42/28.	Yes	Yes

5.5-16 INSERT 6B-4a — Q5.2-1

5.5-17 INSERT 6B-4b — CA5.0-003

5.5-18 INSERT 6B-4c — DC5.0-003

5.5-19 INSERT 6B-4d — Q5.5-7

INSERT 6B-4a

Q 5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
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

INSERT 6B-4b

CA 5.0-003

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-17	This change deletes the phrase "and low pressure turbine disc stress corrosion cracking" from ITS 5.5.10 to make the program consistent with CTS 6.8.4.c.	Yes	No	Yes	Yes

INSERT 6B-4c

DC 5.0-003

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-18	Revises DCPP Sections 5.5.9 and 5.6.10 to reflect License Amendment 124/122, dated March 12, 1998, which allows implementation of steam generator tube voltage based on repair criteria for ODSCC indications at tube to tube support plate intersections.	Yes	No	No	No

INSERT 6B-4d

Q 5.5-7

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-19	Wolf Creek CTS surveillance requirements 4.7.6.3c.3) and 4.9.13b.3) for safety-related ventilation system filter adsorber units include the requirement to measure flow rates within specified values, while imposing an artificial differential pressure, during system operation, when tested in accordance with ANSI N510-1980. The CTS surveillance requirements are incorporated into the ITS.	No	No	Yes	No

ADDITIONAL INFORMATION COVER SHEET

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

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:

Attachment No. 13, CTS 3/4.7 - ITS 3.7

Encl. 3A	12
Encl. 3B	13

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
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 DOC10-17-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	
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	

CHANGE NUMBER	NSHC	DESCRIPTION
09-07	A	A note is added to the [ESW] surveillance that clarifies system operability requirements. Isolation of [ESW] flow to individual components does not render the system inoperable. This change is in accordance with NUREG-1431, Rev. 1, and provides clarification only.
10-01	LG	Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
10-02	M	The APPLICABILITY and applicable ACTIONS are revised to incorporate "during movement of irradiated fuel assemblies" in addition to all MODES (i.e., MODES 1-6).
10-03	LS-7	Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
10-04	A	A new ACTION Statement is added by NUREG-1431 to require entering TS 3.0.3 immediately if two trains of the CR ventilation system are inoperable in MODES 1, 2, 3, or 4. The CTS requires entry into TS 3.0.3, since the condition of two trains inoperable is undefined, therefore, the revision has been classified as administrative.
10-05	LS-18	A new option is added to the ACTION by NUREG-1431 that allows the suspension of CORE ALTERATIONS or movement of irradiated fuel versus placing the CR ventilation system in the recirculation mode.
10-06	LG	The details and description of the required actions and the monthly SRs for train operability are relocated to the Bases. This is an example of removing details that are not required to be in TS and is consistent with NUREG-1431, Rev. 1.
10-07	LG	~~The surveillance that verifies control room temperature once per 12 hours is deleted and moved to a licensee controlled document.~~ INSERT 3A-12c Q5.5-8
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 in separate change numbers are made. A new SR is added that requires [CREVS and Emergency Exhaust System] ventilation system filter testing in accordance with the VFTP. The requirements of this specification are either: 1) moved to Section 5.5.11 of the ITS, or 2) ~~deleted since they are duplicated in Regulatory Guide (RG) 1.52, Revision 2, [ANSI N510-1980, or ASTM D 3803-1989]~~ moved to ITS SRs. Q5.5-8

The surveillance that verifies CR temperature once per 12 hours is moved to a licensee controlled document. 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.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
10-03 LS7	The SR for 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 LS18	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 CR temperature once per 12 hours is relocated 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. ~~or deleted as being duplicated in the applicable RGs or Standards.~~ A SR is added that requires [control room and emergency exhaust system] filter testing in accordance with the VFTP.	YES	YES Q 5.5-8	YES	YES
10-09 LS27	The action for an OPERABLE ventilation train not being capable of being supplied from an emergency power source is deleted.	NO. Refer to change 10-16-LG.	NO; not in CTS.	YES	YES

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 5.5-9 **APPLICABILITY:** DC, WC, CA

REQUEST: CTS 3.9.13 (3.9.12 - DCPP) and Change 12-04-A (Wolf Creek, Callaway and Diablo Canyon)

Comment: It appears that some of the CTS requirements covered by this change were deleted rather than transferred to ITS 5.5.11 as stated. Justify the individual deletions.

FLOG RESPONSE: Attached Table 5.5-9 describes where the CTS SRs for fuel building 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.

Per DOC 12-11-A, the SR to measure FHBVS flow rate is not identified as a separate SR in the ITS because it is verified during the other in-place filter tests specified in ITS 5.5.11, "Ventilation Filter Testing Program (VFTP)", and specific ITS SRs. This change does not result in a change to the technical requirements.

DOC 12-04-A has been revised to more clearly describe where the CTS SRs were moved to in the ITS.

ATTACHED PAGES:

Attachment No. 15, CTS 3/4.9 - ITS 3.9
Encl. 3A 8, 9

TABLE Q5.5-9					
DCPP CTS SR	WC CTS SR	CA CTS SR	CP CTS SR	VFTP	ITS SR
4.9.12 a	4.9.13 a	4.9.13 a	N/A		3.7.13.1
4.9.12 b 1	N/A	N/A	N/A		3.7.13.5
4.9.12 b 2	4.9.13 b 1	4.9.13 b 1	N/A	ITS 5.5.11a &b	3.7.13.2
4.9.12 b 3	4.9.13 b 2	4.9.13 b 2	N/A	ITS 5.5.11c	3.7.13.2
4.9.12 b 4	4.9.13 b 3	4.9.13 b 3	N/A	See DOC 12-11-A	
4.9.12 c	4.9.13 c	4.9.13 c	N/A	ITS 5.5.11c	3.7.13.2
4.9.12 d 1	4.9.13 d 1	4.9.13 d 1	N/A	ITS 5.5.11d	3.7.13.2
4.9.12 d 2	4.9.13 g 1	4.9.13 d 2	N/A		3.7.13.3
4.9.12 d 3	4.9.13 g 2	4.9.13 d 3	N/A		3.7.13.4 DC 3.7.13.5 WC&CA
4.9.12 e	4.9.13 e	4.9.13 e	N/A	ITS 5.5.11a	3.7.13.2
4.9.12 f	4.9.13 f	4.9.13 f	N/A	ITS 5.5.11b	3.7.13.2
N/A	4.9.13 d 2	4.9.13 d 4	N/A	ITS 5.5.11e	3.7.13.2 WC&CA

CHANGE NUMBER	NSHC	DESCRIPTION
		irradiated fuel immediately which would establish conditions outside the Applicability of the LCO.
11-03	M	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
11-04	LG	This change moves the restriction on crane operation to a licensee controlled document. ~~The restriction on crane operations may be removed because it is not in the assumptions used for the FHA. Crane operations that could adversely affect fuel stored in the spent fuel pool are controlled as analyzed in the review of heavy load movements. This change is consistent with NUREG-1431, Rev. 1, and moves requirements that do not meet the criteria for inclusion in the TS.~~ INSERT 3A-8a Q3.9-22
12-01	LS-24	The applicability would be changed to "During movement of irradiated fuel in the fuel building" instead of "Whenever irradiated fuel is in the spent fuel pool," consistent with NUREG-1431, Rev. 1. The proposed applicability is consistent with the assumptions used in the Fuel Handling Accident in the fuel building which postulates the inadvertent drop of an irradiated fuel assembly. Potential damage to fuel assemblies due to dropping of heavy loads is addressed by change 12-02-LG.
12-02	LG	Moves the restriction on crane operations over the spent fuel storage areas when the fuel building air cleanup system was inoperable. The restriction on crane operations may be moved because it is not consistent with the assumptions used for the FHA. Crane operations that could adversely affect fuel stored in the spent fuel pool are prohibited in accordance with plant procedures as analyzed in the review of heavy load movements.
12-03	A	The statement that 3.0.3 [and 3.0.4] are not applicable would be removed. This is consistent with the proposed change to integrate the emergency exhaust system requirements for irradiated fuel handling in the fuel building with the emergency exhaust system requirements in Modes 1 through 4. ITS 3.7.13 supports this integration of requirements.
12-04	A	Q5.5-9 The ~~Surveillance Requirements~~ regarding ~~filter~~ testing ~~would~~ be moved ~~to a~~ "Ventilation ~~Filter Testing~~ Program" INSERT 3A-8b

INSERT 3A-8a Q 3.9-22

The requirement to suspend crane operations over the spent fuel pool in the event pool water level is <23 feet, has been removed from the ACTION of CTS 3/4.9.11 (CTS 3/4.9.10 for Comanche Peak) in corresponding ITS 3.7.5, for the fuel pool water level. The bounding design basis fuel handling accident in the spent fuel pool assumes an irradiated fuel assembly is dropped onto [an array of irradiated fuel assemblies seated in] the spent fuel pool. Crane operations that could adversely affect fuel stored in the spent fuel pool are controlled in accordance with plant procedures as analyzed in the review of heavy loads movements. Administrative controls are employed to prevent the handling of loads that have a greater potential energy than those which have been analyzed. Also see licensees responses to NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment." Moving this information maintains consistency with NUREG-1431. The information is moved to a licensee controlled document which is controlled by a 10 CFR 50.59 change process.

INSERT 3A-8b Q 5.5-9

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 ITS 5.5.11. No technical changes to requirements or test specifics except as noted in separate DOCs are made. A new SR is added that requires [Fuel Building Emergency Exhaust System] filter testing in accordance with the VFTP. The requirements of this specification are : 1) moved to ITS 5.5.11, or 2) moved to ITS SRs.

CHANGE NUMBER	NSHC	DESCRIPTION
		Q5.5-9 ~~that is called out in administrative controls Section 5.5.11 of the ITS. This change does not result in a change to technical requirements.~~
12-05	TR-1	Revised Surveillance Requirement to allow for increased flexibility in using an actual or simulated actuation signal. Identification of the specific signal is moved to the Bases.
12-06	A	This requirement would demonstrate the operability of each train of the [Emergency Exhaust System] (including maintaining negative pressure in the building). This is consistent with current practice. This change does not result in a change to technical requirements and is consistent with NUREG-1431, Rev. 1.
		Q3.7.13.2-3
12-07	LS-25 INSERT 3A-9a	~~The proposed change would remove "STAGGERED TEST BASIS" from the 31-day SR. This represents no change in the frequency of testing since the CTS definition of STAGGERED TEST BASIS would have required testing each of the two [Emergency Exhaust System] trains every 31 days and the new STS require each train to be tested every 31 days.~~
		DCALL-001
12-08	LS-16	The proposed change would allow the $\boxed{18}$ month testing of the [Emergency Exhaust System's] ability to maintain the required pressure differential between the building and the outside atmosphere to be performed on a STAGGERED TEST BASIS.
12-09	LG	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
12-10	LS-9	The "within 31 days after removal" requirement for completion of laboratory analyses is deleted. This requirement is not contained in the ITS nor is it contained in the regulatory guide or ANSI standards.
12-11	A	The SR to measure [Emergency Exhaust 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 a. and b.). This change does not result in a change to technical requirements.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 5.5-10 **APPLICABILITY:** WC, CA

REQUEST: ITS 5.5.11.b (Callaway and Wolf Creek)

Comment: The smooth copy of the ITS still has the [ ] around the plant specific bypass value

FLOG RESPONSE: The smooth copy of the ITS has been marked to delete the brackets ([ ]) around the plant specific bypass value. A final review of the smooth ITS and ITS Bases is planned prior to resubmitting to the NRC the smooth copy of the ITS and Bases

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 5.5-12 **APPLICABILITY:** WC

REQUEST: ITS 5.5.11 and CTS 4.7.6.c.2 (Wolf Creek)

> **Comment**: The value of relative humidity is 70% in the ITS, 78% in the CTS markup, and 70% in the CTS. Is it correct to assume the CTS markup value is wrong?

FLOG RESPONSE: The 78% value for relative humidity in CTS 4.7.6.c.2 and CTS 4.7.6.d in the license amendment request is a typographical error. The correct value is 70%. The CTS mark-ups have been revised to specify the correct value.

ATTACHED PAGES:

Attachment 13, CTS 3/4.7 - ITS 3.7

Encl. 2 7-15

(New) Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP); 10-08-A

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by: 10-08-A

1) Verifying that the Control Room Emergency Ventilation System satisfies the in-place penetration and bypass leakage testing acceptance criteria; of less than 1% for HEPA filters and 0.05% for charcoal adsorbers and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm ±10% for the Filtration System and ~~2200 cfm ±10% for the Pressurization System with~~ 750 cfm ±10% ~~going through~~ for the Pressurization System filter adsorber unit; 10-08-A Q5.5-7 10-15-LG

2) Verifying, ~~within 31 days after removal~~, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and ~~[illegible]~~ 70% relative humidity, for a methyl iodide penetration of less than 2%; and 10-23-LS-13 10-08-A Q5.5-12

3) Verifying system flow rate of 2000 cfm ±10% at greater than or equal to 6.6 inches W.G. (dirty filter) for the Filtration System and ~~2200 cfm ±10%~~ 750 cfm ±10% at greater than or equal to 3.6 inches W.G. (dirty filter) for the ~~Pressurization System with 750 cfm ±10% going through the~~ Pressurization System filter adsorber unit during system operation when tested in accordance with ANSI N510-1980. 10-17-A Q5.5-7 10-15-LG

d. After every 720 hours of charcoal adsorber operation by verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and ~~[illegible]~~ 70% relative humidity, for a methyl iodide penetration of less than 2%; – Q5.5-2 10-23-LS-13 10-08-A

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.6 inches Water Gauge while operating the system at a flow rate of 2000 cfm ±10% for the Filtration System and less than 3.6 inches Water Gauge while operating the system at a flow rate of 750 cfm ±10% for the Pressurization System filter adsorber unit 10-08-A

2) Verifying that on an actual or simulated actuation ~~a Control Room Ventilation Isolation or High Gaseous Radioactivity test~~ signal, the system automatically actuates ~~switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks~~ 10-10-TR-1

ADDITIONAL INFORMATION COVER SHEET

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

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:** DC, CP, WC, CA

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: CA 5.0-003 **APPLICABILITY:** CA, WC, DC

REQUEST: ITS 5.5.10, is revised to delete the words: "and low pressure turbine disc stress corrosion cracking". This requirement is not part of the Secondary Water Chemistry Program described in CTS 6.8.4.c.

ATTACHED PAGES:

Encl. 5A	5.0-21
Encl. 6A	4
Encl. 6B	4

5.5 Programs and Manuals

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation ~~and low pressure turbine disc stress corrosion cracking~~. The program shall include: 5.5-17

CA 5.0-003

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP) B-PS

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2, and in accordance with the guidance specified below. B-PS

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < ~~0.05~~ 1% when tested in accordance with Regulatory Guide 1.52, Revision 2~~, and ASME N510-1989~~ at the system flowrate specified below ± 10%. B-PS B-PS B

(continued)

5.5-18 Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B) — DC 5.0-003

5.5-19 INSERT 6A-4g — Q 5.5-7

CHANGE NUMBER	JUSTIFICATION
5.5-15	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
5.5-16	INSERT 6A-4a — Q 5.2-1
5.6-1	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
5.5-17	INSERT 6A-4f — CA 5.0-003
5.6-2	This change deletes the Emergency Diesel Generator 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.
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 10 CFR 20 and 50.36a Changes" (From Christopher I. Grimes to Owners Groups Chairs). This change is consistent with traveler TSTF-152.
5.6-5	[ ] PORV lift settings are referenced in the PTLR section per ~~WOG-67, Rev. 1.~~ TSTF-233 — TR 5.0-003
5.6-6	INSERT 6A-4b — Q 5.2-1
5.7-1	~~This change revises High Radiation Area to incorporate changes consistent with [10 CFR 20.1601]. Specifically, distances from the radiation source are noted.~~ INSERT 6A-4c — Q 5.2-1
5.7-2	~~This change revises "unauthorized" to "inadvertent" in the High Radiation Area section to reflect the NRC's position as stated in Regulatory Guide 8.38, Section 1.5 regarding physical barriers for High Radiation Areas. This is consistent with traveler TSTF-167.~~ INSERT 6A-4d — Q 5.2-1
5.7-3	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
5.7-4	INSERT 6A-4e — Q 5.2-1

INSERT 6A-4a Q 5.2-1

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 surveillances. 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.

INSERT 6A-4b Q 5.2-1

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.

INSERT 6A-4c Q 5.2-1

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.

INSERT 6A-4d Q 5.2-1

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 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 maintaining appropriate radiation worker practices.

INSERT 6A-4e Q 5.2-1

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.

INSERT 6A-4f CA 5.0-003

ITS Section 5.5.10 is being revised consistent with CTS 8.8.4.c. The proposed change deletes the phrase "and low pressure turbine disc stress corrosion cracking" from the ITS to be consistent with the practices of the CTS which do not have this requirement for the Secondary Water Chemistry Program.

INSERT 6A-4g Q 5.5-7

5.5-19 CTS surveillance requirements 4.7.6.3c.3) and 4.9.13b.3) for safety-related ventilation system filter adsorber units include the requirement to measure flow rates within specified values, while imposing an artificial differential pressure, during system operation, when tested in accordance with ANSI N510-1980. This flow rate testing is to be performed at least once per 18 months, after any structural maintenance on the HEPA filter or charcoal adsorber housings, or following painting, fire, or chemical release in any ventilation zone communicating with the system. Therefore, the CTS surveillance requirements are incorporated into the ITS.

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

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-12	The referenced frequencis for the tests listed in the Ventilation Filter Testing Program (VFTP) were evaluated as part of the 24 month fuel cycle program for DCPP (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 revsied consistent with traveler ~~WOG-85~~ TSTF-237 [and ~~a License Amendment Request submitted December 1, 1996~~]. 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 Amendment No. 106 dated June 24 1997	Yes	Yes	Yes Q5.5-2
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 Callaway CTS.	No	No	No	Yes
5.6-1	Revises Section 5.6.4, "Monthly Operating Report," to reflect a revised submittal date.	No. DCPP CTS 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.~~ TSTF-37, Rev 1.	Yes Q5.6-2	No. Not in CTS	No. Not in CTS.	No. Not in CTS.
5.6-3	Revises report dates in ITS 5.6.2, "Annual Radiological Environmental Operating Report" to be consistent with current TS.	Yes. Consistent with CTS and LA 78/77.	Yes. See LA 42/28.	Yes	Yes

5.5-16 INSERT 6B-4a Q5.5-1

5.5-17 INSERT 6B-4b CA 5.0-003

5.5-18 INSERT 6B-4c DC 5.0-003

5.5-19 INSERT 6B-4d Q5.5-7

INSERT 6B-4a

Q 5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
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

INSERT 6B-4b

CA 5.0-003

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-17	This change deletes the phrase "and low pressure turbine disc stress corrosion cracking" from ITS 5.5.10 to make the program consistent with CTS 6.8.4.c.	Yes	No	Yes	Yes

INSERT 6B-4c

DC 5.0-003

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-18	Revises DCPP Sections 5.5.9 and 5.6.10 to reflect License Amendment 124/122, dated March 12, 1998, which allows implementation of steam generator tube voltage based on repair criteria for ODSCC indications at tube to tube support plate intersections.	Yes	No	No	No

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-19	Wolf Creek CTS surveillance requirements 4.7.6.3c.3) and 4.9.13b.3) for safety-related ventilation system filter adsorber units include the requirement to measure flow rates within specified values, while imposing an artificial differential pressure, during system operation, when tested in accordance with ANSI N510-1980. The CTS surveillance requirements are incorporated into the ITS.	No	No	Yes	No

ADDITIONAL INFORMATION COVER SHEET

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

REQUEST: "ITS 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 page
Encl. 6A	4
Encl. 6B	5

INDUSTRY TRAVELERS APPLICABLE TO SECTION 5.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-9, Rev. 1	Incorporated		NRC approved.
TSTF-37, Rev. 1	Incorporated	5.6-2	DCPP only.
TSTF-52	Incorporated	5.5-4	Incorporated draft Rev. 1 per Q3.6.1-6 Q 3.6.1-6
TSTF-65 Rev. 1	~~Not~~ Incorporated	~~NA~~ 5.2-9	~~Not~~ NRC approved ~~as of traveler cutoff date.~~ CPSES only. TR 5.0-005
TSTF-106, Rev. 1	Not Incorporated	NA	Retain CTS.
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
TSTF-238 ~~WOG-67, Rev. 1~~	Incorporated	5.6-5	NRC approved. TR 5.0-003
WOG-72	Incorporated	5.5-13	
TSTF-237 ~~WOG-85~~	Incorporated	5.5-14	Q 5.5-2
~~Proposed Traveler~~ TSTF-258	Incorporated	5.2-2, 5.5-1, 5.2-3, 5.2-6, 5.3-2, 5.5-16, 5.6-6, 5.7-1,	~~WOG mini-group Action Item 147.~~ Q 5.2-1

5.5-18 Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B) DC 5.0-003

CHANGE NUMBER	JUSTIFICATION
5.5-19	INSERT 6A-4g Q5.5-7
5.5-15	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
5.5-16	INSERT 6A-4a Q5.2-1
5.6-1	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
5.5-17	INSERT 6A-4f CA 5.0-003
5.6-2	This change deletes the Emergency Diesel Generator 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.
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 10 CFR 20 and 50.36a Changes" (From Christopher I. Grimes to Owners Groups Chairs). This change is consistent with traveler TSTF-152.
5.6-5	[ ] PORV lift settings are referenced in the PTLR section per ~~WOG-67, Rev. 1.~~ TSTF-233 TR 5.0-003
5.6-6	INSERT 6A-4b Q5.2-1
5.7-1	~~This change revises High Radiation Area to incorporate changes consistent with [10 CFR 20.1601]. Specifically, distances from the radiation source are noted.~~ INSERT 6A-4c Q5.2-1
5.7-2	~~This change revises "unauthorized" to "inadvertent" in the High Radiation Area section to reflect the NRC's position as stated in Regulatory Guide 8.38, Section 1.5 regarding physical barriers for High Radiation Areas. This is consistent with traveler TSTF-167.~~ INSERT 6A-4d Q5.2-1
5.7-3	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 6B).
5.7-4	INSERT 6A-4e Q 5.2-1

SECTION 5.0

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
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 traveler TSTF-152.	Yes	Yes	Yes	Yes
5.6-5	DCPP LTOP a[illegible] and PORV lift settings are referenced in PTLR section per ~~WOG-67, Rev. 1~~ TSTF-233	Yes TR 5.0-003	Yes	Yes	Yes
5.7-1	~~Revises High Radiation Area to incorporate changes consistent with [10 CFR 20.1601].~~ INSERT 6B-5b	Yes Q5.2-1	Yes	Yes	Yes
5.7-2	~~Changes "unauthorized" to "inadvertant" in the High Radiation Area section to reflect the NRC's position as stated in RG 8.3.8, Section 1.5 regarding physical barriers for High Radiation Areas. This change is consistent with traveler TSTF-167.~~ INSERT 6B-5C	~~Yes~~	~~Yes~~	~~Yes~~	~~Yes~~ Q5.2-1
5.7-3	This change deletes the phrase "or that cannot be continuously guarded" from the ITS for Callaway to make them consistent with the CTS.	No	No	No	Yes

5.6-6 INSERT 6B-5a Q5.2-1

5.7-4 INSERT 6B-5d Q5.2-1

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 SECTION 5.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-9, Rev. 1	Incorporated		NRC approved.
TSTF-37, Rev. 1	Incorporated	5.6-2	DCPP only.
TSTF-52	Incorporated	5.5-4	Incorporated draft Rev. 1 per Q3.6.1-6 Q 3.6.1-6
TSTF-65 Rev. 1	~~Not~~ Incorporated	~~NA~~ 5.2-9	~~Not~~ NRC approved ~~as of traveler cutoff date~~. CPSES only. TR 5.0-005
TSTF-106, Rev. 1	Not Incorporated	NA	Retain CTS.
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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	5.6-5	NRC approved. TR 5.0-003
WOG-72	Incorporated	5.5-13	
TSTF-237 ~~WOG-85~~	Incorporated	5.5-14	Q 5.5-2
~~Proposed Traveler~~ TSTF-258	Incorporated	5.2-2, 5.5-1, 5.2-3, 5.2-6, 5.3-2, 5.5-16, 5.6-6, 5.7-1,	~~WOG mini-group Action Item 147.~~ Q 5.2-1

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 5.0-ED **APPLICABILITY:** WC

REQUEST: 1) The electronic CTS page 6-18a did not have the correct text for item b. under CTS 6.8.4.g. This page has been marked to correctly reflect the CTS.

2) DOC 3-07 in Enclosure 3B should be identified as 3-07-A.

3) In ITS 5.5.11c the "≤" is deleted consistent with CTS SR 4.7.6c.2) and ASTM D3803-1979.

ATTACHED PAGES:

Encl. 2	6-18a
Encl. 3B	6
Encl. 5A	5.0-22

PROCEDURES AND PROGRAMS (Continued)

3. a kinematic viscosity within limits for ASTM 2D fuel oil,

4. a water and sediment content within the limits for ASTM 2D fuel oil;

b. Other properties for ASTM 2D fuel oil are ~~within limits~~ analyzed within ~~30~~ 31 days following sampling and addition of new fuel oil to storage tanks when tested based on ASTM D975; and 02-16-A 02-19-LS-2 WC 5.0-ED

c. Total particulate concentration of the stored fuel oil is < 10 mg/liter when tested every 31 days based on ASTM D2276, Method A. 02-12-LG

~~h. Emergency Diesel Generator Reliability Program~~

~~An emergency diesel generator reliability program that establishes the requirements and guidelines for emergency diesel generator reliability, availability, and monitoring. The program shall include the following:~~

~~a. Emergency diesel generator reliability performance goals (target reliability) based upon the station blackout coping assessment. Target reliability goal monitoring is accomplished through monitoring methods that are based upon those described in Appendix D of NUMARC 87-00,~~

~~b. Measures to ensure detailed root cause analysis of emergency diesel generator failures is performed and effective corrective actions are taken in response to failures,~~

~~c. Implementation of an emergency diesel generator preventive maintenance program that is consistent with the Maintenance Rule, and~~

~~d. Monitoring of emergency diesel generator availability and performance parameters to ensure the target reliability is met or exceeded.~~

i. 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 psig.

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-02 A	The requirement to submit a startup report is deleted from the CTS. This report required no staff approval and was submitted after the fact and, is therefore, not required to ensure safe plant operation. The approved 10 CFR 50, Appendix B, QA Plan, and FSAR startup testing program provides assurance that the affected activities are adequately performed and that appropriate corrective actions, if required, are taken.	Yes	No. Deleted from CTS per Amendment 50/36	Yes	Yes
03-03 A	Revises the annual report section to reflect the new 10 CFR Part 20 requirements and associated recommended changes noted in 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) - TSTF-152.	Yes	Yes. Except the Part 20 requirements were removed from the TS in Amendment 50/36	Yes	Yes
03-04 A	The requirement to report specific activity limit violations is deleted consistent with NUREG-1431. Serious degradation of a fission product barrier, among other more serious events are required to be reported by 10 CFR 50.73. This change is administrative in that it only affects reports and do not affect plant operations.	Yes	Yes	Yes	Yes
03-05 A	The Annual Radiological Environmental Operating Report including submittal date is revised.	No. DCPP report dates to remain as in CTS.	Yes	No. WCNOC report dates to remain as in CTS.	No. Callaway report dates to remain as in CTS
03-06 A	CTS [6.9.1.7], "Annual Radioactive Effluent Release Report" and CTS [6.14.c] is revised consistent with NUREG-1431, Rev. 1, to delete the term "Annual" and modify the submittal date.	Yes	Yes	Yes	Yes
03-07 A	CTS [6.9.1.6], "Annual Radiological Environmental Operating Report" is revised to include specific details concerning the contents of the report.	Yes	Yes	Yes	Yes WC 50-ED

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation System-Filtration	2000 cfm
Control Room Emergency Ventilation System - Pressurization	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	6500 cfm

B-PS

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal absorber shows a penetration and system bypass < [0.05]% when tested in accordance with Regulatory Guide 1.52, Revision 2~~, and ASME N510-1989~~ at the system flowrate specified below ~~[± 10%]~~.

B-PS

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation System - Filtration	2000 cfm
Control Room Emergency Ventilation System - Pressurization	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	6500 cfm

WC 5.0-006

B-PS

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ~~≤~~ 30°C and greater than or equal to the relative humidity specified below.

B-PS

B-PS

WC 5.0-ED

ESF Ventilation System	Penetration	RH
Control Room Emergency Ventilation System (Filtration/Pressurization)	2%	70%
Auxiliary/Fuel Building Emergency Exhaust	2%	70%

WC 5.0-006

B-PS

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

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 5.0-001 **APPLICABILITY:** WC

REQUEST: Amendment No. 110 was issued on September 22, 1997. This amendment modified CTS 5.3.1, "Fuel Assemblies" and CTS 6.9.1.9, "CORE OPERATING LIMITS REPORT," to add ZIRLO as fuel material and the use of limited zirconium alloy filler rods in place of fuel rods. As identified in Attachment 3 to the conversion application, the amendment request was incorporated into the conversion application. Subsequent to the submittal of the conversion application, the ZIRLO license amendment request was revised at the NRC request to include the reference to WCAP-12610 in CTS 6.9.1.9. This licensee identified item reflects in the conversion application the approval of Amendment No. 110 to the Wolf Creek license.

Additionally, the numbering in ITS 5.6.5 is corrected.

ATTACHED PAGES:

Encl. 2	6-21b
Encl. 5A	5.0-32

CORE OPERATING LIMITS REPORT (COLR) (Continued)

~~Nuclear Enthalpy Rise Hot Channel Factor~~ $\cancel{F^N_{\Delta H}}$ ~~Specification~~
~~3.9.1.b Refueling Boron Concentration).~~
j. INSERT 2-21 — WC 5.0-001

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-hydraulic limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC. ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector.~~ 03-08-A

~~SPECIAL REPORTS~~ 03-08-A
~~6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.~~

~~6.10 RECORD RETENTION~~ 03-09-LG
~~In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.~~

~~6.10.1 The following records shall be retained for at least 5 years:~~

~~a. Records and logs of unit operation covering time interval at each power level;~~

~~b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;~~

~~c. All REPORTABLE EVENTS;~~

~~d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;~~

~~e. Records of changes made to the procedures required by Specification 6.8.1;~~

~~f. Records of radioactive shipments;~~

~~g. Records of sealed source and fission detector leak tests and results; and~~

~~h. Records of annual physical inventory of all sealed source material of record.~~

new RCS Pressure and Temperature Limits Report (PTLR) 03-13-M
see insert 21

new PAM Report
see insert 22

j. NRC Safety Evaluation Reports dated July 1, 1991, "Acceptance for Referencing of Topical Report WCAP-12610, 'VANTAGE+ Fuel Assembly Reference Core Report' (TAC NO. 77258)," and September 15, 1994, "Acceptance for Referencing of Topical Report WCAP-12610, Appendix B, Addendum 1, 'Extended Burnup Fuel Design Methodology and ZIRLO Fuel Performance Models' (TAC No. M86416)" (WCAP-12610-P-A)

~~(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor - $F_Q(Z)$~~ 3-17-A

5.6 Reporting Requirements

1. NRC Safety Evaluation Report dated October 29, 1992, for the "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station" (ET-90-0140, ET 92-0103).

2. NRC Safety Evaluation Report dated January 17, 1989, for the "Acceptance for Referencing of Licensing Topical Report WCAP-11397, Revised Thermal Design Procedure."

3. NRC Safety Evaluation Report dated September 30, 1993 for the "Transient Analysis Methodology for the Wolf Creek Generating Station" (ET-91-0026, ET 92-0142, WM 93-0010, WM 93-0028).

4. NRC Safety Evaluation Report dated November 26, 1993, "Acceptance for Referencing of Revised Version of Licensing Topical Report WCAP-10216-P-A, Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification" (TAC No. M88206).

5. NRC Safety Evaluation Report dated March 10, 1993, for the "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station" (ET 92-0032, ET 93-0017).

6. NRC Safety Evaluation Report dated March 30, 1993, for the "Revision to Technical Specification for Cycle 7" (NA 92-0073, NA 93-0013, NA 93-0054).

7 ~~6.~~ NRC Safety Evaluation Report dated November 13, 1986, for "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code" (WCAP-10266-P-A, Rev. 2). WC 5.0-001

8 ~~7.~~ NRC Safety Evaluation Report dated May 17, 1988, "Acceptance for Referencing of Westinghouse Topical Report WCAP-11596 - Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores." WC 5.0-001

9 ~~8.~~ NRC Safety Evaluation Report dated June 23, 1986, "Acceptance for Referencing of Topical Report WCAP 10965-P and WCAP 10966-NP-ANC: A Westinghouse Advanced Nodal Computer Code." WC 5.0-001

10. INSERT 5A - 32 — WC 5.0-001

(continued)

10. NRC Safety Evaluation Reports dated July 1, 1991, "Acceptance for Referencing of Topical Report WCAP-12610, 'VANTAGE+ Fuel Assembly Reference Core Report' (TAC NO. 77258)," and September 15, 1994, "Acceptance for Referencing of Topical Report WCAP-12610, Appendix B, Addendum 1, 'Extended Burnup Fuel Design Methodology and ZIRLO Fuel Performance Models' (TAC No. M86416)" (WCAP-12610-P-A)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 5.0-002 **APPLICABILITY:** WC

REQUEST: The word "Manuals " is misspelled in the header on page 5.0-7 through 5.0-26. The editorial change has been identified for correction in the smooth copy of the ITS. The marked-up pages will not be included in this RAI response.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 5.0-003 **APPLICABILITY:** WC

REQUEST: Amendment No. 115 dated March 30, 1998 revised CTS Sections 6.3 and 6.12 to reflect the merger for the positions of Superintendent Radiation Protection and Superintendent Chemistry into one new position, Manager Chemistry/Radiation Protection. The CTS and ITS have been marked up to reflect incorporation of this amendment.

Wolf Creek submitted a CTS license amendment request on July 3, 1997 (letter number ET 97-0065) which position title changes of the Shift Supervisor to Shift Manager and Supervising Operator to Control Room Supervisor. In discussion with the Wolf Creek NRC Project Manager on April 10, 1998, this amendment request is to be approved in conjunction with the approval of the conversion license amendment request. The CTS and ITS have been marked up to reflect the incorporation of this license amendment request. Note that TSTF-258 would eliminate some of the changes requested in the CTS license amendment request.

ATTACHED PAGES:

Encl. 2	6-1, 6-2, 6-5, 6-6, 6-7, 6-23
Encl. 5A	5.0-1, 5.0-5, 5.0-6, new Section 5.7.2

6.1 RESPONSIBILITY

Control Room Supervisor

WC 5.0-003

6.1.1 The Plant Manager shall be responsible for overall Unit operation and shall delegate in writing the succession to this responsibility during his absence.

Manager

6.1.2 The Supervising Operator, under the Shift Supervisor, shall be responsible for the control room command function. ~~A management directive to this effect, signed by the President and Chief Executive Officer shall be reissued to all station personnel on an annual basis.~~ **During any absence of the SO 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 SO 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.** 01-01-A 01-02-A

6.2 ORGANIZATION

6.2.1 Onsite and Operating Corporation Organization

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

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

b. The Plant Manager shall be responsible for overall unit safe operation and shall have control of those onsite activities necessary for safe operation and maintenance of the plant.

c. The President and Chief Executive 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.

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

01-02-A

6.2.2 Unit Staff

The unit staff organization shall include the following: 01-06-LG

a. ~~Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.~~ **A nuclear station operator shall be assigned when fuel is in the reactor and an additional nuclear station operator shall be assigned when the unit is in MODES 1, 2, 3 or 4.** 01-07-LG

Unit Staff (Continued)

b. ~~At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the Unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Operator shall be in the control room;~~ 01-05-A

c. An individual from the Health Physics Groups*, qualified in radiation protection procedures, shall be on site when fuel is in the reactor;

d. ~~ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;~~ 01-03-A

01-08-LG

e. ~~A site Fire Brigade of at least 5 members* shall be maintained onsite 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; and~~ Manager WC 5.0-003

f. Administrative procedures shall be developed and implemented to limit the working hours of Unit Staff who perform safety related functions; e.g., Senior Operators, Operators, Health Physicists, 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). INSERT 2-2a 1-09-A Q 5.2-1

g. The Superintendent Operations or Manager Operations shall hold a senior reactor operator license.

*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.

~~TABLE 6.2-1~~ 01-06-LG

~~MINIMUM SHIFT CREW COMPOSITION~~ 01-02-A

~~POSITION~~	~~NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION~~	
	~~MODE 1, 2, 3, or 4~~	~~MODE 5 or 6~~
~~SM~~ ~~SS~~	~~1~~	~~1*~~
~~CRS~~ ~~SO~~	~~1~~	~~None~~
~~RO~~	~~2~~	~~1~~
~~NSO~~	~~4~~	~~1~~
~~STA~~	~~1**~~	~~None~~
~~CHM~~	~~1~~	~~None~~

WC 5.0-003

~~SM~~ ~~SS~~ ~~Shift Supervisor~~ ~~Manager~~ ~~with a Senior Operator license on Unit 1~~
~~CRS~~ ~~SO~~ ~~Supervising Operator~~ ~~with a Senior Operator license on Unit 1~~ ~~Control Room Supervisor~~
~~RO Individual with an Operator license on Unit 1~~
~~NSO Nuclear Station Operator~~
~~STA Shift Technical Advisor~~
~~CHM Chemistry Personnel~~

WC 5.0-003

The Shift Crew Composition may be one less than the minimum requirements of **10CFR50.54(m)(2)(i) and 6.2.2** ~~Table 6.2-1~~ 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. ~~of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.~~

Control Room Supervisor

During any absence of the Supervising Operator from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Supervising Operator from the control room while the Unit is in MODE 5 or 6, an individual with a valid Operator license shall be designated to assume the control room command function

Control Room Supervisor

WC 5.0-003

~~Manager~~

WC 5.0-003
01-03-A

~~*One individual with a Senior Operator license, either Shift Supervisor or Supervising Operator~~
~~Control Room Supervisor~~

~~**The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Supervisor or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.~~ 01-06-LG

~~Manager~~

~~6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)~~

01-04-LG

~~FUNCTION~~

~~6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, REPORTABLE EVENTS and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Chairman Nuclear Safety Review Committee.~~

~~COMPOSITION~~

~~6.2.3.2 The ISEG shall be composed of at least five, dedicated, full time engineers located on site. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field.~~

~~RESPONSIBILITIES~~

~~6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.~~

~~RECORDS~~

~~6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to Chairman Nuclear Safety Review Committee.~~

6.2.4 SHIFT TECHNICAL ADVISOR

An individual**

~~The Shift Technical Advisor (STA)~~ shall provide technical support to the ~~Shift Supervisor~~ unit operations shift crews in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the Unit.

1-15-A

Q 5.2-1

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 with the following exceptions:

a. Licensed Operators and Senior Operators shall meet or exceed the qualifications of ANSI/ANS 3.1-1981 as endorsed by Regulatory Guide 1.8, Revision 2.

~~*Not responsible for sign off function.~~

04-LG

This

undelete

**~~The STA~~ position shall be manned in MODES 1, 2, 3, and 4 unless the Shift ~~Supervisor~~ or the individual with a Senior Operator license meets the qualifications ~~for the STA~~ as required by the NRC.

Manager

WC 5.0-003

~~01-04-LG~~

1-15-A

Q 5.2-1

6.3 UNIT STAFF QUALIFICATION (Continued) Manager Chemistry/ (WC 5.0-003)

b. The position of ~~Superintendent~~ Radiation Protection shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, for a Radiation Protection Manager.

~~c. The NSRC members shall meet or exceed the requirements of ANSI/ANS 3.1-1981.~~ 01-04-LG

d. The position of Manager Operations shall hold or have previously held a senior reactor operator license for a similar unit (PWR).

~~6.4 TRAINING~~

~~6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager Training and shall meet or exceed the requirements and recommendations of Section 5 of ANSI/ANS 3.1-1978 with the following exceptions:~~ 01-04-LG

~~a. The training program for Licensed Operators and Senior Operators shall meet or exceed the requirements and recommendations of Section 5 of ANSI/ANS 3.1-1981 as endorsed by Regulatory Guide 1.8, Revision 2, and 10 CFR Part 55.~~

~~b. Training shall include familiarization with relevant industry operational experience identified by the ISEG or another plant group.~~

01-04-LG

~~6.5 REVIE[illegible] AND AUDIT~~

~~6.5.1 PL[illegible] SAFETY REVIEW COMMITTEE (PSRC)~~

~~FUNCTI[illegible]~~

~~6.5.1.1 The PSRC shall function to advise the Plant Manager on all matters related to nuclear safety.~~

~~COMPOSITION~~

~~6.5.1.2 The Plant Manager shall designate in writing the Chairman and Alternate Chairman of the PSRC. PSRC membership shall include between six and eight additional members appointed by the Chairman and an additional member appointed by the Vice President Engineering. Selected members shall include, at a minimum, management responsible for the following areas of expertise: operations, maintenance, instrumentation and controls, chemistry, health physics and engineering. A single individual may cover multiple disciplines.~~

~~ALTERNATES~~

~~6.5.1.3 All alternate members shall be appointed in writing by the PSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PSRC activities at any one time.**~~

~~**Except for the alternate for Engineering who is appointed by the Vice President Engineering.~~

~~6.11 RADIATION PROTECTION PROGRAM~~ 03-10-LG

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

6.12 HIGH RADIATION AREA 03-11-A

6.12.1 Pursuant to ~~Paragraph 20.203(c)(5) of~~ 10 CFR Part 20, in lieu of the ~~"control device" or "alarm signal" required by Paragraph 20.203(c)(2)~~ **requirements of 10 CFR 20.1601**, each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is **greater than 100 mrem/hr but** equal to or less than 1000 mR/h at ~~45 cm (18 in.)~~ **30 cm** ~~(12 in.)~~ ~~from the radiation source or from any surface which the radiation penetrates~~ 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 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 equal to or less than 1000 mR/h **at 30 cm** ~~(12 in.)~~, 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: QS.2-1 QS.2-1

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 personnel 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 Superintendent Radiation Protection~~ **health physics** ~~**supervision**~~ in the RWP. ~~Manager Chemistry/Radiation Protection~~ QS.2-1 03-11-A WC 5.0-003

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than **or equal to** 1000 mR/h at ~~45 cm (18 in.)~~ **30 cm (12 in.)** ~~from the radiation source or from any surface which the radiation penetrates~~ shall be provided with locked doors **or continuously guarded** to prevent ~~unauthorized~~ ~~**inadvertent**~~ entry, and the keys shall be maintained under the administrative control of the ~~Shift Supervisor/Supervising Operator~~ on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel 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 personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area. ← undelete 03-11-A 03-20-LS-3 03-19-A QS.2-1

Shift Manager/Control Room Supervisor — WC 5.0-003

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

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] or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.~~ 5.1.1

Manager

Control Room Supervisor

5.1.2 The ~~Shift Supervisor (SS)~~ ~~Supervising Operator (SO)~~ under the Shift ~~Supervisor~~ shall be responsible for the control room command function. During any absence of the ~~SS~~ ~~Supervising Operator~~ 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~~ ~~Supervising Operator~~ 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 B-PS B-PS

Control Room Supervisor

An individual

unit operations shift crew

5.2 Organization

Q 5.2-1

~~Shift Manager~~

WC 5.0-003

This

Shift Manager

WC 5.0-003

~~g~~f. ~~The Shift Technical Advisor (STA)~~ shall provide advisory technical support to the ~~Shift Supervisor (SS)~~ in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. ~~In addition, the STA shall~~ ~~The STA~~ position shall be manned in MODES 1, 2, 3 or 4, unless the ~~SS~~ or the individual with a Senior Operator License meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. 5.2-6

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.~~

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 with the following exceptions: ~~[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 B-PS

5.3.1.1 Licensed Operators and Senior Operators shall meet or exceed the qualifications of ANSI/ANS 3.1-1981 as endorsed by Regulatory Guide 1.8, Revision 2 and 10 CFR Part 55. 5.3-1

5.3.1.2 The position of ~~Superintendent~~ Manager Chemistry / Radiation Protection shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 for a Radiation Protection Manager. WC 5.0-003

5.3.1.2 The position of Manager Operations shall hold or have previously held a senior reactor operator license for a similar unit (PWR).

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m). 5.3-2 Q 5.2-1

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)

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 1 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: Shift Manager/Control Room Supervisor WCSD-003

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.

or health physics supervision

B-PS

(continued)

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 5.0-004 **APPLICABILITY:** WC

REQUEST: Amendment No. 106 was issued on June 24, 1997. This amendment modified CTS 6.8.5.b to provide an exception to the examination requirements of Regulatory Guide 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity" and delays the inspection of the "D" reactor coolant pump flywheel to the Fall 1997 refueling outage. As identified in Attachment 3 to the conversion application, the amendment request was incorporated into the conversion application. Subsequent to the submittal of the conversion application, the license amendment request was revised to include a one-time allowance tc extend the performance of "D" reactor coolant pump examination. This licensee identified item reflects in the conversion application the approval of Amendment No. 110 to the Wolf Creek license. New DOC 2-21-A was generated to delete the one-time extension for the "D" reactor coolant pump examination. DOC 2-21-A states: "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 are 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."

ATTACHED PAGES:

Encl. 2	6-18c
Encl. 3A	7
Encl. 3B	5

PROCEDURES AND PROGRAMS (Continued)

3. A surveillance program to ensure that the quantity of radioactivity contained in following outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste system, is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2 at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA, in the event of an uncontrolled release of the tanks' contents **02-09-A**

   a. Reactor Makeup Water Storage Tank.
   b. Refueling Water Storage Tank,
   c. Condensate Storage Tank, and
   d. Outside Temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

b. Reactor Coolant Pump Flywheel Inspection Program

Each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, dated August, 1975. # In lieu of Position C.4.b(1) and C.4.b(2), conduct a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or conduct a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels once every ten years coinciding with the Inservice Inspection schedule as required by ASME Section XI. 2-21-A WC 5.0-004

c. Containment Tendon Surveillance Program

This program provides controls for monitoring tendon performance, including the effectiveness of the tendon corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial plant operation as well as periodic testing thereafter. The Containment Tendon Surveillance Program, and its inspection frequencies and acceptance criteria, shall be in accordance with Wolf Creek Generating Station position on draft Revision 3 of Regulatory Guide 1.35, dated April, 1979.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Containment Tendon Surveillance Program inspection frequencies.

new. **Technical Specification Bases Control Program (see insert 10)** **02-11-M**

new. **Safety Function Determination Program (see insert 11)** **02-11-M**

6.9 REPORTING REQUIREMENTS

The following reports shall be submitted in accordance with 10 CFR 50.4. **03-01-A**

~~ROUTINE REPORTS~~

~~6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the NRC Regional Office unless otherwise noted.~~

~~# The volumetric examination and surface examination of the Reactor Coolant Pump "D" motor flywheel for the first 10-year Inservice Inspection interval may be delayed one cycle to coincide with the Fall 1997 refueling outage.~~ 2-21-A WC 5.0-004

02-23 LG Not Applicable to WCGS. See Conversion Comparison Table (Enclosure 3B) DC 5.0-004

CHANGE NUMBER	NSHC	DESCRIPTION
02-21	A	INSERT 3A-7b WC 5.0-004 which reference these programs, and therefore, the lack of an applicability statement in the Programs introduces confusion.
02-22	A	INSERT 3A-7a Q5.2-1
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	The requirement to submit a Startup Report is deleted from the CTS to be consistent with NUREG-1431. This report required no staff approval and was submitted after the fact and is therefore not required to ensure safe plant operation. The approved 10 CFR 50, Appendix B, QA Plan, and USAR startup testing program provides assurance that the affected activities are adequately performed and that appropriate corrective actions, if required, are taken.
03-03	A	The Annual Reports section is revised to be consistent with NUREG-1431 and traveler TSTF-152. Names and formats are revised consistent with NUREG-1431. Also, revises the annual report section to reflect the new 10 CFR Part 20 requirements and associated recommended changes noted in NRC letter dated July 28, 1995, "Changes to Technical Specifications Resulting from 10CFR20 and 50.36a Changes." (From Christopher I. Grimes to Owners Groups Chairs).
03-04	A	The requirement to report specific activity limit violations is deleted consistent with NUREG-1431. This report is a history of Reactor Coolant System (RCS) specific activity Limiting Conditions for Operation (LCO) entries. GL 83-43 and revised reporting requirements in the regulations intended that LCO entry reports no longer be required. The reporting requirements in regulations cover situations such as seriously degraded barriers (fuel failure). Therefore, every violation of the RCS specific activity LCO need not be reported. Serious degradation of a fission product barrier, among other more serious events are required to be reported by 10 CFR 50.73. This change is administrative in that it only affects reports and do not affect plant operations.
03-05	A	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
03-06	A	CTS [6.9.1.7], "Annual Radioactive Effluent Release Report" and CTS [6.14.c] is revised consistent with NUREG-

f. Demonstrate at least once per 18 months for each of the ESF systems that following the creation of an artificial Delta P across the combined HEPA filters, the prefilters, and the charcoal absorbers of not less than the value specified below (dirty filter conditions), that the flowrate through these flow paths is with ± 10% of the value specified below when tested in accordance with ANSI N510-1980.

ESF Ventilation System	Delta P	Flowrate
Control Room Filtration System	6.6 in. W.G.	2000 cfm
Control Room Pressurization System	3.6 in. W.G.	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	4.7 in. W.G.	6500 cfm

LIST OF COMMITMENTS

The following table identifies those actions committed to by Wolf Creek Nuclear Operating Corporation (WCNOC) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Michael J. Angus, Manager Licensing and Corrective Action at Wolf Creek Generating Station, (316) 364-8831, extension 4077.

COMMITMENT	Due Date/Event
A supplement to Reference 3 (ET 97-0050 - conversion license amendment request) will be submitted at a later date.	Prior to issuance of SER
The Withdrawn Specimen Test Results Report will be submitted to the NRC by the end of September 1998.	9/30/98

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops – MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fission product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

redline Q 3.4.Gen-1

The reactor coolant is circulated through four loops connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

(continued)

BASES (continued)

~~SL~~ 3.4.Gen-1

APPLICABLE SAFETY ANALYSES (continued)

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the limit values, 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.~~ 10 CFR 50.36(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 at rated power.

redline Q3.4Gen-1 CA3.4-004

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.

APPLICABILITY

In MODES 1 and 2, the reactor ~~is critical and thus~~ when critical 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)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

redline Q3.4.Gen-1

The reactor coolant is circulated through four RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

redline Q3.4.Gen-1

APPLICABLE SAFETY ANALYSES

Whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM.

Therefore, in MODE 3 with ~~RTBs in the closed position and~~ the Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least two RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when

redline Q3.4.Gen-1

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to be consistent with MODE 3 accident analyses.

The operation of one RCP in MODES 3, 4, and 5 provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during RCS boron concentration reductions. With no reactor coolant loop in operation in either MODES 3,4, or 5, boron dilutions must be terminated and dilution sources isolated. The boron dilution analysis in these MODES take credit for the mixing volume associated with having at least one reactor coolant loop in operation. (Ref. 1)

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops – MODE 3 satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36(c)(2)(ii).

redline Q3.4.Gen-1

LCO

The purpose of this LCO is to require that at least two RCS loops be OPERABLE. In MODE 3 with the ~~RTBs in the closed position and~~ Rod Control System capable of rod withdrawal, two RCS loops must be in operation. Two RCS loops are required to be in operation in MODE 3 with the ~~RTBs closed and~~ Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

~~With the RTBs in the open position, or the CRDMs de-energized.~~ When the Rod Control System is not capable of rod withdrawal ~~therefore,~~ only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that ~~safety analyses limits are met~~redundancy for heat removal is maintained.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If the required RCS loop is not in operation, and the ~~RTBs are closed and~~ Rod Control System is capable of rod withdrawal, the Required Action is either to restore the required RCS loop to operation or to place the Rod Control System in a condition incapable of rod withdrawal (e.g., by de-energizing ~~de-energize~~ all CRDMs, by opening the RTBs or de-energizing the motor generator (MG) sets). When the ~~RTBs are in the closed position and~~ Rod Control System is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the Rod Control System must be rendered incapable of rod withdrawal. ~~the RTBs must be opened.~~ The Completion Times of 1 hour to restore the required RCS loop to operation or defeat the Rod Control System ~~de-energize all CRDMs~~ is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

D.1, D.2, and D.3
redline Q3.4.Gen-1

If four RCS loops are inoperable or no RCS loop is in operation, except as during conditions permitted by the Note in the LCO section, place the Rod Control System in a condition incapable of rod withdrawal (e.g., by de-energizing all CRDMs, ~~must be de-energized~~ by opening the RTBs or de-energizing the MG sets).

All operations involving a reduction of RCS boron concentration must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Addition of borated water with a concentration greater than or equal to the minimum required RWST concentration but less than the actual RCS boron concentration shall not be considered a reduction in boron concentration. (Ref. 2). Boron dilution requires forced circulation for proper mixing, and defeating the Rod Control System ~~opening the RTBs or de-energizing the MG sets~~ removes the possibility of an inadvertent rod withdrawal. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

6 Q3.4.Gen-1

This SR requires verification every 12 hours that the required loops are in operation. Verification may include flow rate, temperature, ~~and~~or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

redlined - Q3.4.Gen-1

The reactor coolant is circulated through four RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for decay heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. ~~The RCS and RHR loops provide this circulation.~~

The operation of one RCP in MODES 3, 4, and 5 provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during RCS boron concentration reductions. With no reactor coolant loop in operation in either MODES 3, 4, or 5, boron dilutions must be terminated and dilution sources isolated. The boron dilution analysis in thes MODES take credit for the mixing volume associated with having at least one reactor coolant loop in operation. (Ref. 1).

RCS Loops - MODE 4 ~~have been identified in the NRC Policy Statement as important contributors to risk reduction.~~ satisfies criterion 4 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be removed from operation ~~de-energized~~ for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are required to be performed without flow or pump noise ~~designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again.~~ The 1 hour time period is adequate to perform the necessary testing, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by ~~initial startup~~ test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

redlined Q3.4.Gen-1

Note 2 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ ~~350°F~~ 368°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. ~~275~~ Q3.4.Gen-1

(continued)

BASES

LCO (continued)

stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by ~~initial startup~~ test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

redline Q3.4.Gen-1

Note 3 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with any RCS cold leg temperature ≤ ~~350~~ 368°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. ~~275~~ Q 3.4.Gen-1

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required. An OPERABLE SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

(continued)

BASES

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

Note 1 permits all RHR pumps to be ~~de-energized~~ **removed from operation** for ≤ **1 hour** ~~15 minutes when switching from one loop to another~~. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet temperature is maintained ≥ 10°F below saturation temperature. ~~The Note prohibits boron dilution or draining operations when RHR forced flow is stopped.~~ **The Note requires reactor vessel water level be above the vessel flange to ensure the operating RHR pump will not be intentionally deenergized during mid-loop operations.**

at least

WC 3.4-001

redline

Q 3.4.Gen-1

Note 2 allows one RHR loop to be inoperable for a period of ≤ **2 hours**, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

Q3.4.7-2

APPLICABILITY

In MODE 5 with loops not filled ~~(as defined in plant procedures)~~, this LCO requires core heat removal and coolant circulation by the RHR System. **One RHR loop provides sufficient capability for this purpose. However, one additional RHR loop is required to be OPERABLE to meet single failure considerations.**

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
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 3.9.5, "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)

BASES

APPLICABILITY (continued)

The Applicability is modified by a Note stating that entry into MODE 5-Loops Not Filled from MODE 5-Loops Filled is not permitted while the LCO is not met. This Note specifies an exception to LCO 3.0.4 and would prevent draining the RCS, which would eliminate the possibility of SG heat removal, while the RHR function was degraded.

ACTIONS

A.1

If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no required RHR loops are OPERABLE or in operation, except during conditions permitted by Note 1, all operations involving a reduction of RCS boron concentration must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. Addition of borated water with a concentration greater than or equal to the minimum required RWST concentration but less than the actual RCS boron concentration shall not be considered a reduction in boron concentration. (Ref. 2). Boron dilution requires forced circulation from at least one RCP for proper mixing so that inadvertent ~~uniform dilution, and the margin to~~ criticality can be prevented ~~must not be reduced in this type of operation~~. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

6 - Q3.4.Gen-1

This SR requires verification every 12 hours that one loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

(continued)

BASES

BACKGROUND (continued)

a loss of single phase natural circulation and decreased capability to remove core decay heat. Two groups of backup pressurizer heaters are normally powered via the Class 1E 4.16kV buses. The heater loads will be shed after a safety injection or bus undervoltage signal and manually sequenced back onto the Class 1E 4.16kV buses.

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the UFSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

LCO

redline Q3.4.Gen-1

The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ ~~1240~~ 1657 cubic feet, which is equivalent to 92%, ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires two groups of backup pressurizer heaters to be OPERABLE ~~pressurizer heaters~~, each with a capacity ≥ ~~125~~ 150 kW, capable of being powered from either the offsite power source or the emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer

(continued)

BASES (continued)

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 UFSAR (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:

redline Q3.4.Gen-1

undelete

a. Uncontrolled rod withdrawal from full power;

b. ~~Loss of reactor coolant flow~~ Feedwater line break; Q3.4.10-1

c. Loss of external electrical load/turbine trip; Q3.4.10-1

d. Loss of normal feedwater;

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

f. Locked rotor.

g.

h. Rod cluster control assembly ejection Q3.4.10-1

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in ~~events c, d, and e (above)~~ the above events 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.~~ 10 CFR 50.36(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.

(continued)

LCO (continued)

The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

redline Q 3.4. Gen-1

APPLICABILITY

In MODES 1, 2, and 3, ~~and portions of MODE 4 above the LTOP arming temperature,~~ OPERABILITY of three valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 ~~and portions of MODE 4 are~~ is conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4, ~~when all RCS cold leg temperatures are ≤ 320°F or in~~ MODE 5, or MODE 6 with the reactor vessel head on because LTOP is ~~provided~~ in service. Overpressure protection is not required in MODE 6 with the reactor vessel head ~~detensioned~~ removed (vent path ≥ 2.0 square inches).

The Note allows entry into MODES 3 ~~and 4~~ with the lift settings outside the LCO limits. This method permits the inplace testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

redline Q 3.4. Gen-1

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

(continued)

ACTIONS (continued)

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, 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 4 ~~with any RCS cold leg temperatures ≤ 320°F~~ 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. With any RCS cold leg temperatures at or below ~~320~~ 368°F, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

~~275~~ Q3.4.Gen-1

redline Q3.4.Gen-1

Addition to the RCS of borated water with a concentration greater than or equal to the minimum required RWST concentration shall not be considered a positive reactivity change. Cooldown of the RCS for restoration of operability of a pressurizer code safety valve, with a negative moderator temperature coefficient, shall not be considered a positive reactivity change provided the RCS is borated to the COLD SHUTDOWN, xenon-free condition per specification 3.1.1. (Ref. 5)

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

~~The pressurizer safety valve setpoint is ± [3]% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.~~

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.

2. ~~FSAR~~ USAR, Chapter 15.

3. WCAP-7769, Rev. 1, June 1972.

(continued)

BASES (continued)

BACKGROUND (continued)

Pressure – High reactor trip setpoint following a step reduction of 50% of full load with steam dump.

In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature protection (LTOP). See LCO 3.4.12, "Low Temperature Protection (LTOP) System."

APPLICABLE SAFETY ANALYSES

Plant operators may employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are ~~used~~ also modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria, pressurizer volume, or hot leg saturation are examined (Ref. 2) ~~are critical~~. By assuming PORV ~~manual~~ actuation, the primary pressure remains below the high pressurizer pressure trip setpoint.~~; thus, t~~The DNBR calculation is more conservative and the transient pressurizer water volume is maximized, and the hot leg saturation temperature is reduced for those transients assuming PORV operation. As such, this actuation is not required to mitigate these events, and PORV automatic operation is, therefore, not an assumed safety function. ~~The automatic actuation of the PORVs is not assumed in any of the design basis accidents during MODES 1, 2 or 3. Events that assume this condition include a turbine trip, and the loss of normal feedwater (Ref. 2).~~ – Q 3.4.Gen-1

Pressurizer PORVs satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (c)(2)(ii).

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

(continued)

ACTIONS (continued)

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

~~and B.3~~ s Q3.4.Gen-1

If one PORV 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 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 not~~ may not be capable of mitigating an ~~overpressure~~ event ~~when placed in manual control~~ if the inoperable block valve is not full open. If the block valve is restored within the Completion Time of 72 hours, the power will be restored ~~and~~ to 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.

INSERT B 3.4-5 Q3.4.11-4

(continued)

APPLICABLE SAFETY ANALYSIS (continued)

not occur, which either of the LTOP overpressure protection means cannot handle:

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

b. Deactivating the accumulator discharge isolation valves in their closed positions or by venting the affected accumulator; and

redline Q3.4.Gen-1

c. ~~Disallowing~~ Precluding start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCO 3.4.5, "RCS Loops - MODE 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," provide this protection.

Operation below 350°F but greater than 325°F with all centrifugal charging and safety injection pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic safety injection actuation signals except Containment Pressure - High are blocked. In normal conditions a single failure of the ESF actuation circuitry will result in the starting of at most one train of safety injection (one centrifugal charging pump, and one safety injection pump). For temperatures above 325°F, an overpressure event occurring as a result of starting two pumps can be successfully mitigated by operation of both PORV's without exceeding Appendix G limit. Given the short time duration that this condition is allowed and the low probability of a single failure causing an overpressure event during this time, the single failure of a PORV is not assumed. Initiation of both trains of safety injection during this 4-hour time frame due to operator error or a single failureoccurring during testing of a redundant channel are not considered to be credible accidents.

Although LTOP is required to be OPERABLE when RCS temperature is less than 368°F, operation with all centrifugal charging pumps and both safety injection pumps OPERABLE is acceptable when RCS temperature is greater than 350°F. Should an inadvertent safety injection occur above 350°F, a single PORV has sufficient capacity to relieve the combined flow rate of all pumps. Above 350°F, two RCPs and all pressurizer safety valves are required to be OPERABLE. Operation of an RCP eliminates the possibility of a 50°F difference existing between indicated and actual RCS temperature as a result of heat transport effects. Considering instrument uncertainties only, an indicated RCS temperature of 350°F is sufficiently high to allow full RCS pressurization in

(continued)

BASES

LCO (continued)

two independent means to prevent a pump start in accordance with SR 3.4.12.2.

Note 2 recognizes the Applicability overlap between LCO's 3.4.12 and 3.5.2 and states that two safety injection pumps and two centrifugal charging pumps may be capable of injecting into the RCS:

redline Q3.4.Gen-1

(a) In MODE 3 with any RCS cold leg temperature < 368° F and ECCS pumps OPERABLE pursuant to LCO 3.5.2, "ECCS-Operating", and

(b) For up to 4 hours after entering MODE 4 from MODE 3 or the temperature of one or more RCS cold legs decreases below 325°F, whichever comes first.

Note 3 states that one or more safety injection pumps may becapable of injecting into the RCS in MODES 5 and 6 when the RCS water level is below the top of the reactor vessel flange for the purpose of protecting the decay heat removal function.

the may be unisolated Q3.4.12-2

Note 4 states that accumulator ~~isolation is only required~~ when the accumulator pressure is ~~greater than or equal to~~ the maximum RCS pressure for the existing RCS cold leg temperature as allowed by the P/T limit curves provided in the PTLR. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

less than

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

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

~~1~~a. 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.

(bracketed text)
redline Q3.4.Gen-1

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

(continued)

BASES

LCO (continued)

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.

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

Q3.4.Gen-1

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

redline (bracketed text)

An RCS vent is OPERABLE when open with an area of ≥ ~~2.07~~ 2.0 square inches.

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

APPLICABILITY

This LCO is applicable in MODE 3 when ~~the temperature of~~ any RCS cold leg temperature is ≤ 368°F, 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 in MODES 1, 2, and 3 ~~above 275°F~~. When the reactor vessel head is off, overpressurization cannot occur. ~~With fuel off loaded, the reactor vessel head may be placed on the vessel for radiological conditions but not bolted. Overpressure protection is maintained because sources for thermally induced overpressure are not available and the reactor vessel head will lift and relieve at low pressure if hydraulically induced pressure is present.~~

Q3.4.12-4

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 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.

(continued)

BASES

ACTIONS

C.1, D.1 and D.2~~, D.3~~ (continued)

the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either ~~one~~ of which must be performed in the next 12 hours. By increasing the RCS temperature to ~~>275~~ 368°F, an accumulator pressure of ~~693~~ 648 psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

~~600~~
Q3.4.Gen-1

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

E.1

In MODE 3 with any RCS cold leg temperature ≤368°F or MODE 4 ~~when any RCS cold leg temperature is ≤ [275]°F~~, with one required RCS relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS relief valves in any combination of the PORVS and the RHR suction relief valves are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

Q3.4.Gen-1

redline (bracketed text)

The Completion Time considers the facts that only one of the RCS relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

F.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS relief valves inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.

(continued)

BASES

ACTIONS

F.1 (continued)

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS relief valve to protect against overpressure events.

G.1

The RCS must be depressurized and a vent must be established within 8 hours when:

a. Both required RCS relief valves are inoperable; or

b. A Required Action and associated Completion Time of Condition A, B, D, E, or F is not met; or

redline (bracketed text) Q3.4 Gen-1

c. The LTOP System is inoperable for any reason other than Condition A, B, C, D, E, or F.

~~2.07~~ Q3.4.Gen-1

The vent must be sized ≥ ~~2.98~~ 2.0 square inches to ensure that the flow capacity is greater than that required for the worst case 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

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of zero safety injection pumps and a maximum of ~~two~~ one charging pump~~s~~ are verified to be ~~in~~capable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and ~~locked out~~ with power removed from the valve operator.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.4 (continued)

available to the operator in the control room that verify the RHR suction isolation valves ~~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

~~2.07~~ Q3.4Gen-1

The RCS vent of ≥ ~~2.98~~ 2.0 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., for a vent valve, a valve that is locked, sealed, or otherwise secured in position). A removed pressurizer safety valve or open manway 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 3.4.12~~b~~.d.

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 in the main control room. This Surveillance is only required to be performed if the PORV is being used to meet this ~~satisfies the~~ LCO.

redline (bracketed text) Q3.4 Gen-1

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.6 (continued)

develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.7 Not Used.

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valve are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.4 for the RHR suction valve Surveillance and for a description of the requirements of the Inservice Testing Program.) This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.~~

block of text missing that would be lined/struck through — Q 3.4 Gen-1

SR 3.4.12.8

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ ~~275~~ 368°F and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

The 12 hour allowance ~~Frequency~~ considers the unlikelihood of a low temperature overpressure event during this time.

A Note has been added indicating that this SR is not required to be ~~met~~ performed until 12 hours after decreasing RCS cold leg temperature to ≤~~275~~ 368°F. ~~The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES.~~ WC 3.4-008

(continued)

BASES

LCO (continued)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the ~~SLB~~ accident analys~~i~~es **involving secondary steam discharge to the atmosphere**. Violation of this LCO could exceed the offsite dose limits for th~~is~~**ese** accidents. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. <u>Primary to Secondary LEAKAGE through Any One SG</u>

redline $\boxed{\text{Q3.4.Gen-1}}$

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 leak~~ed~~age is through many cracks, then 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

<u>A.1</u>

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

(continued)

BASES

ACTIONS (continued)

~~with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.~~

Q3.4.14-2

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~~.

Q3.4.14-2

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 timeframe considers the time required to complete the Action and the low probability of a second valve failing during this time period.

Q3.4.14-2

Block of text missing but is lined through (see next page) Q34Gen-1

B.1 and B.2

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 status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. 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.

(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].

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 PIV.** The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

~~or~~

Q 3.4.Gen-1

~~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

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 status, the plant must be brought to MODE 3

(continued)

ACTIONS (continued)

C.1

The inoperability of the RHR ~~System autoclosure~~ suction isolation valve interlock renders the ~~RHR~~ suction ~~isolation valves~~ ~~incapable of isolating in response to a high pressure condition and preventing~~ could allow inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure. If the ~~RHR System autoclosure~~ RHR suction isolation valve interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated ~~automatic~~ remote manual valve within 4 hours. This Action accomplishes the purpose of the ~~autoclosure function~~ interlock.

WC3.4-009 undelete

WC 3.4-009

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Q3.4.14-2

Performance of leakage testing on each RCS PIV ~~or isolation valve~~ used to satisfy Required Action A.1 ~~and Required Action A.2~~ is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

redline Q3.4.Gen-1 redline Q3.4.Gen-1

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. ~~8~~ 7) as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. ~~7~~ 6), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

WC 3.4-010

WC 3.4-010

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere particulate and gaseous radioactivity monitors. The check gives reasonable confidence that the channel ~~is~~ are operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the required containment atmosphere particulate and gaseous radioactivity monitors. The test ensures that the monitors can perform ~~its~~ their function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.15.3, SR 3.4.15.4, and SR 3.4.15.5

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

redline Q3.4.Gen-1

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.

2. Regulatory Guide 1.45.

3. ~~FSAR~~ USAR, Section 5.2.5

4. NUREG-609, "Asymmetric Blowdown Loads on PWR Primary Systems," 1981.

5. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated pipe Breaks in PWR Primary Main Loops."

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.1-1 **APPLICABILITY:** DC, CP, WC, CA

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	2-15
Encl. 3A	8
Encl. 3B	7

Attachment 10 CTS 3/4.4 ITS 3.4

Encl. 5A	Traveler Status page, 3.4-4
Encl. 5B	B 3.4-7
Encl. 6A	7
Encl. 6B	6

ACTION: (Continued)

4. ~~Identify and c~~Correct ~~the cause of~~ the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION 1.b and/or 3, above; subsequent POWER OPERATION may proceed provided that the indicated RCS total flow rate is demonstrated to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels: 05-05-LG

   a. A nominal 50% of RATED THERMAL POWER,

   b. A nominal 75% of RATED THERMAL POWER, and

   c. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.5.1 The provisions of Specification 4.0.4 are not applicable to Specification 3.2.5.c.

4.2.5.2 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.3 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. 05-12-A ~~[illegible]~~ CP 3.2-001

undelete undelete

4.2.5.4 The RCS total flow rate shall be ~~measured~~ ~~determined by precision heat balance measurement~~ at least once per 18 months.# ~~Within 7 days prior to performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.~~ Q 3.4.1-1 05-07-M 05-04-LG

~~4.2.5.5 The feedwater venturi shall be inspected for fouling and cleaned as necessary at least once per 18 months.~~ 05-09-LG

Not required to be performed until 7 days after achieving ≥ 95 % RTP. 05-07-M

CHANGE NUMBER	NSHC	DESCRIPTION
		~~elimination. The proposed changes, based on traveler WOG-95, which clarify that power reductions must be completed within 2 hours after each QPTR determination and permit achieving equilibrium conditions for measuring peaking factors are not considered to be relaxations of current requirements. This is because completion times for these activities are not specified in current TS.~~ Q3.2-6
04-11	A	INSERT 3A-8 WC 3.2-001
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	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
05-03	LG	CP 3.2-001 ~~Consistent with NUREG-1431, the requirement to perform a CHANNEL CALIBRATION on the RCS flow meters at least once per 18 months and the requirement to normalize the channels are moved to the Bases for the RCS flow - low reactor trip function in ITS Section 3.3.1.~~ Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
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 example of removing unnecessary details from the TS and is acceptable based on the guidance provided in 10 CFR 50.36.~~ INSERT 3A-8a Q3.4.1-1
05-05	LG	Consistent with NUREG-1431, Wolf Creek specific REQUIRED ACTIONS would be modified to move details regarding identifying the cause of RCS low flow rate to the Bases. This is acceptable because it would remove details that are not required to be in TS to provide operational safety while retaining the limiting conditions for operation.
05-06	LS-8	In accordance with NUREG-1431, if any of the DNB-related parameters of pressure, temperature, or RCS flow are found to be outside their limits, the time period required to perform a power reduction would be extended to 6 hours. The DNB-related parameters of Reactor Coolant System (RCS) average temperature, pressurizer pressure, and RCS flow rate are

INSERT 3A-8 WC 3.2-001

04-11 A A note is added to Wolf Creek CTS SR 4.2.4.2 to indicate that the surveillance is not required to be performed until 12 hours after input from one Power Range Neutron Flux channel is inoperable with THERMAL POWER > 75% RTP. This change is considered an administrative change since Action A.2 provides a frequency of 12 hours to determine QPTR when QPTR has exceed 1.02. Further justification is based on the fact that under normal circumstances, QPTR would not be expected to change significantly within a 12 hour period. If a significant change in QPTR were to occur, it would likely be the result of control rod misalignment which would likely be detectable immediately by means of the rod deviation monitor or rod bottom lights.

Additionally, a note is added to CTS SR 4.2.4.1 to indicate that CTS SR 4.2.4.2 may be performed in lieu of this surveillance requirement to confirm the indication of the remaining three excore channels. As identified in the NUREG-1431, Rev. 1, definition of QPTR and the SR 3.2.4.2 Bases, QPTR is a ratio of excore detector outputs and, for the purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating tilt.

INSERT 3A-8a Q 3.4.1-1

CTS 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.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
4-11 A	INSERT 3B-7 — WC 3.2-001				
04-10 ~~LS-14~~	~~The allowed time for the requirement to reset the Power Range Neutron Flux-High setpoint during power reduction required by QPTR ACTIONS would be extended to 72 hours for Wolf Creek.~~ Not Used.	~~No~~ NA	~~No~~ NA	~~Yes~~ NA	~~No~~ NA Q3.2-6
05-01 LG	The designation of how instrument uncertainties are treated (nominal, in the analysis, or in the development of the CTS limit) is moved to the Bases.	Yes	Yes	Yes	Yes
05-02 LS-7	The CPSES specific requirement to verify that the total RCS flow is within limits using the plant computer or elbow tap output voltage on a monthly basis is deleted.	No	Yes	No	No
05-03 LG	The requirement ~~to perform a CHANNEL CALIBRATION at least once per 18 months and the requirement~~ to normalize the RCS loop flow rate indicators is ~~channels are~~ moved to the Bases for the surveillance requirements for ~~the RCS flow - low reactor trip function in~~ ITS ~~3.3.1~~. 3.4.1	~~Yes~~ No-Not in CTS	Yes	~~Yes~~ No-Not in CTS	~~Yes~~ No-Not in CTS CP 3.2-001
05-04 LG	~~Consistent with industry traveler TSTF-105, the explicit requirements that the RCS flow be measured through the use of a precision heat balance measurement and that the instrumentation used in the performance of the calorimetric flow measurement be calibrated within a specified time period of performing the measurement is moved to the Bases.~~ The details regarding the timing of calibration of the instrumentation used in the performance of the calorimetric flow measurement is moved to the Bases.	No - Requirement not in CTS.	Yes	Yes	Yes Q3.4.1-1
05-05 LG	The Wolf Creek required actions would be modified to move details regarding identification of the cause for low RCS flow rate 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 05-11-A.	Yes	Yes

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TR..VELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev ~~2~~ 3	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		Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. ~~1~~ 2	Incorporated	3.4-31	Approved by NRC. TR 3.4-004
TSTF-93 Rev. 3	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~~	~~3.4-28~~	Q 3.4.1-1
TSTF-108, Rev. 1	Not Incorporated	NA	LCO 3.4.19 does not apply.
TSTF-113, Rev. ~~3~~ 4	Incorporated	3.4-39	Q 3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev. ~~1~~ 2	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..
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
TSTF-285 ~~WOG-51, Rev. 1~~	Incorporated	3.4-23, 3.4-45, 3.4-52	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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	3.4-10	DCPP only Approved by NRC. 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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.1.1	Verify pressurizer pressure is ≥ ~~[2200]~~ 2220 psig.	12 hours	B-PS
SR 3.4.1.2	Verify RCS average temperature is ≤ ~~[581]~~ 590.5°F.	12 hours	B-PS
SR 3.4.1.3	Verify RCS total flow rate is ≥ ~~[284,000]~~ 37.1 x 10^4 gpm.	12 hours	B-PS
SR 3.4.1.4	-----NOTE----- Not required to be performed until ~~24 hours~~ 7 days after ≥ 95 % RTP. ----- undelete Verify ~~by precision heat balance that~~ measured RCS total flow rate is ≥ ~~[284,000]~~ 37.1 x 10^4 gpm.	Q 3.4.1-1 18 months	B-PS 3.4-40 ~~3.4-38~~ B-PS B

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The installed flow instrumentation provides indication as a percentage of total flow rate based on the precision calorimetric heat balance. Plant procedures specify the percentage of the total flow rate required to meet the RCS total flow rate limit. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

SR 3.4.1.4

undelete

Measurement of RCS total flow rate ~~by performance of a precision calorimetric heat balance~~ once every 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. ~~This verification is performed via a precision calorimetric heat balance.~~ When performing a precision heat balance, the instrumentation used for determining steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi Δp in the calorimetric calculations shall be calibrated within 7 days prior to performing the heat balance. Q 3.4.1-1 Q3.4.1-1

The Frequency of 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 ~~24 hours~~ 7 days after ≥ ~~90~~ 95% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of ~~90~~ 95% RTP to obtain the stated RCS flow accuracies and the test is only a confirmation of SR 3.4.1.4. The Surveillance shall be performed within ~~24 hours~~ 7 days after reaching ~~90~~ 95% RTP.

REFERENCES

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

CHANGE
NUMBER | JUSTIFICATION

purposes (per Bases). This allowance is properly presented as an SR Note. A properly placed exception (i.e., an SR Noted 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 ~~WOG-50~~ TSTF-288 Q3.4.11-2

3.4-36

SR 3.4.13.1 and LCO 3.4.15 are revised per traveler TSTF-116. The note addresses the concern that an RCS water inventory balance connot 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 WCGS. See Conversion Comparison Table (Enclosure 3B).

3.4-38

Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B)

Q 3.4.1-1

~~Consistent with TSTF-105, the details on the method by which the RCS flow rate are verified are moved from 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.~~

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. INSERT 6A-7a 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. Q3.4.11-6

3.4-40

Consistent with traveler ~~WOG-99~~ TSTF-282, the Note to SR 3.4.1.4 would be modified to provide additional time to perform an RCS precision flow rate measurement. The time allowed would be changed from 24 hours to 7 days. This change is acceptable because other indication of RCS flow is available (SR 3.4.1.3, RCS total flow meters) and additional time normally would be required to establish Q3.4.1-2

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	
3.4-38	Consistent with the 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. ~~Consistent with TSTF-105, the details on the method by which the RCS flow rate are verified are moved from the SR 3.4.1.4 to the Bases.~~	Yes	~~Yes~~ No	~~Yes~~ No	~~Yes~~ No	Q3.4.1-1
3.4-39	The shutdown requirements of ITS 3.4.11 would require the plant 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 all 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 reactor 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 - Specific to DCPP	No	No	No	
3.4-43	A new Condition 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	Q345-2 Q345-3

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.1-2 **APPLICABILITY:** DC, CP, WC, CA

REQUEST: Difference 3.4-40

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

FLOG RESPONSE: WOG-99 has been designated TSTF-232 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:

Attachment 8, CTS 3/4.2 - ITS 3.2
- Encl. 3A 9

Attachment 10, CTS 3/4/4 - ITS 3.4
- Encl. 5A Traveler Status page
- Encl. 6A 7

CHANGE NUMBER	NSHC	DESCRIPTION
		maintained within specified limits in order to ensure consistency with the assumed initial conditions of the accident analyses. The limits placed on the RCS temperature, pressure, and flow ensure that the minimum departure from Nucleate Boiling ratio (DNBR) will be met for each of the transients analyzed. Compliance with the above limits is verified every 12 hours. If a parameter is found to be outside the required limit, 2 hours are allowed in order to restore the parameter to within the limit. If the parameter is not restored to compliance within the required time, the plant must be shut down. The revised completion time of 6 hours is acceptable to allow transition to the required plant conditions in an orderly manner without unnecessarily initiating any undue plant transients and on the small likelihood of a severe event occurring during the extended time period.
05-07 TSTF-282 Q 3.4.1-2	M	This surveillance for measuring RCS flow by precision heat balance is modified to add a footnote that corresponds to the Note for ITS SR 3.4.1.4. The footnote requires that the surveillance be performed within 7 days of achieving 95% RTP. This is more restrictive in that it ties the surveillance to the beginning of a cycle. This is acceptable because other indication of RCS flow is available (RCS flow meters) and time is provided to establish plant conditions suitable for the precision heat balance. This is consistent with traveler WOG-99. In addition, the THERMAL POWER specified in the footnote would be changed from the generic value provided in NUREG-1431 to a plant-specific value of ≥ 95 % RTP. This change is acceptable because it specifies a THERMAL POWER in better agreement with current operating procedures for performing a precision heat balance. Current TS do not specify a power level for this measurement.
05-08		Not used.
05-09	LG	The requirements for inspecting and cleaning the feedwater flow venturi would be moved to licensee controlled documents.. These details are not contained in NUREG-1431. This is an example of moving unnecessary detailed information from the TS and is acceptable.
05-10	A	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
05-11	A	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
5-12	A	INSERT 3A -9 — CP 3.2-001

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC.
TSTF-27, Rev. ~~2~~ 3	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		Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. ~~1~~ 2	Incorporated	3.4-31	Approved by NRC. TR 3.4-004
TSTF-93 Rev. 3	Incorporated	3.4-17	Approved by NRC. Q3.4.9-3
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. ~~3~~ 4	Incorporated	3.4-39	Q 3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev. ~~1~~ 2	Incorporated	3.4-36	Q3.4.13-2
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..
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
TSTF-285 ~~WOG-51, Rev. 1~~	Incorporated	~~3.4-23~~, 3.4-45, 3.4-52	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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	3.4-10	DCPP only Approved by NRC. 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
	purposes (per Bases). This allowance is properly presented as an SR Note. A properly placed exception (i.e., an SR Noted 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 ~~WOG-50~~ TSTF-288 Q3.4.11-2
3.4-36	SR 3.4.13.1 and LCO 3.4.15 are revised per traveler TSTF-116. The note addresses the concern that an RCS water inventory balance connot 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 WCGS. See Conversion C rison Table (Enclosure 3B).
3.4-38	Q 3.4.1-1 Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B) ~~Consistent with TSTF-105, the details on the method by which the RCS flow rate are verified are moved from 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.~~
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. INSERT 6A-7a Q3.4.11-6 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.
3.4-40	Consistent with traveler ~~WOG-99~~ TSTF-282, the Note to SR 3.4.1.4 Q3.4.1-2 would be modified to provide additional time to perform an RCS precision flow rate measurement. The time allowed would be changed from 24 hours to 7 days. This change is acceptable because other indication of RCS flow is available (SR 3.4.1.3, RCS total flow meters) and additional time normally would be required to establish

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.2-1 **APPLICABILITY:** DC, CP, WC, CA

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 3.4-10
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 ~~2~~ 3	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		Minor change that is adequately addressed in the Bases.
TSTF-87, Rev. ~~1~~ 2	Incorporated	3.4-31	Approved by NRC. TR 3.4-004
TSTF-93 Rev. 3	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~~	~~3.4-28~~ Q 3.4.1-1	
TSTF-108, Rev. 1	Not Incorporated	NA	LCO 3.4.19 does not apply.
TSTF-113, Rev. ~~3~~ 4	Incorporated	3.4-39	Q 3.4.11-3
TSTF-114	Incorporated	NA	Approved by NRC.
TSTF-116, Rev. ~~1~~ 2	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..
TSTF-151 Rev. 1	Incorporated	NA	TR 3.4-009
TSTF-153	Incorporated	3.4-01	Approved by NRC. TR 3.4-004
TSTF-162	Incorporated	NA	Approved by NRC. TR 3.4-006
TSTF-285 ~~WOG-51, Rev. 1~~	Incorporated	~~3.4-23~~, 3.4-45, 3.4-52	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
TSTF-233 ~~WOG-67, Rev. 1~~	Incorporated	3.4-10	DCPP only Approved by NRC. 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

BASES

ACTIONS

A.1 (continued)

requirements within the allowed Completion Time. In accordance with plant procedures, the plant is brought to MODE 3 to allow for more stable plant conditions prior to resumption of power operation. 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 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 ~~30 minutes~~ 12 hours. ~~when [T_{avg}-T_{ref} deviation, low low T_{avg}] alarm not reset and any RCS loop T_{avg} < [547]°F.~~

~~The Note modifies the SR. When any RCS loop average temperature is < [547]°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~~ and takes into account indications and alarms that are continuously available to the operator in the control room, and is consistent with other routine Surveillances which are typically performed once per shift. Q3.4.2-1

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. Q3.4.2-1

REFERENCES

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

CHANGE NUMBER	JUSTIFICATION
	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 overpressurization potential when operating within a CONDITION. This option should be retained as exists in the current Technical Specifications.
3.4-31	The 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, the 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 rods cannot be withdrawn, replaces these specific methods 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 traveler TSTF-87 Rev. 1.
3.4-32	In accordance with 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.
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 the 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. Q3.4.2-1
3.4-34	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
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

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.3-1 **APPLICABILITY:** DC, CP, WC, CA

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>

Wolf Creek removed surveillance specimen capsule V during the ninth refueling outage in November 1997. This capsule is currently being evaluated, and the methodology of WCAP-14040-NP-A, Rev. 1 is being utilized for the Cold Overpressure Mitigating System setpoint and heatup and cooldown curve development. The Withdrawn Specimen Test Results Report will be submitted to the NRC by the end of September 1998. Therefore, Wolf Creek believes it is acceptable to reference WCAP-14040-NP-A, since the WCAP has been approved by the NRC, and Wolf Creek is following the methodology provided in the WCAP.

ATTACHED PAGES:

None

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.4-1 **APPLICABILITY:** DC, CP, WC, CA

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 USAR, 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: Q 3.4.5-1 **APPLICABILITY:** WC, CA

REQUEST: Change 1-14 LS-22. (Callaway and Wolf Creek)

Comment: The change discussion is not adequate. The NSHC contains the necessary justification.

FLOG RESPONSE: DOC 1-14-LS-22 is revised to read:

"The LCO and ACTION b of Specification 3.4.1.2, "Reactor Coolant System, Hot Standby," would be revised to require that two reactor coolant loops be OPERABLE. Loop operation requirements would also be revised to be contingent on Rod Control System status. The requirement to have a third OPERABLE reactor coolant loop would be deleted, consistent with NUREG-1431. This is acceptable because the MODE 3 decay heat removal requirements are sufficiently low that a single RCS loop with one RCP running is adequate to remove core decay heat. A second RCS loop ensures redundant capability for decay heat removal. When the Rod Control System is capable of rod withdrawal, two loops must be in operation to ensure accident analysis assumptions are satisfied. When rod withdrawal is precluded, only one loop is required to be in operation to satisfy MODE 3 accident analyses. The MODE 3 accident analyses which assume only two RCS loops in operation include the Uncontrolled RCCA Bank Withdrawal from Subcritical and the hot zero power RCCA ejection events. The initial conditions and analysis assumptions for these events will be unchanged since two loops must still be in operation during MODE 3 when the Rod Control System is capable of rod withdrawal. These reactivity transients rely on the Nuclear Instrumentation System's high flux trips for event termination which occurs very rapidly (on the order of seconds). There would be no benefit of having a third RCS loop OPERABLE for these transients since by the time the loop could be brought into operation, the event would be over for all practical purposes."

ATTACHED PAGES:

Encl. 3A 3

CHANGE NUMBER	NSHC	DESCRIPTION
1-11	M	This change adds a new surveillance for verification of breaker alignment and power availability to the required pump not in operation. This change is in conformance with NUREG-1431 Rev. 1.
1-12	M	The Actions are changed to separate the required actions for only one required RHR loop OPERABLE and no required RHR loops OPERABLE. These revised Actions are consistent with the Actions which are required under this LCO in NUREG-1431 Rev. 1, and are more conservative than current required actions.
1-13	M	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
1-14	LS-22	~~The LCO and Action b are revised to require only two loops OPERABLE with two loops in operation when the rod control system is capable of rod withdrawal and one loop in operation when the rod control system is not capable of rod withdrawal. This change is consistent with NUREG-1431 Rev. 1.~~ INSERT 3A-3a Q3.4.5-1
1-15	M	A steam generator (SG) level corresponding to 10% of the wide range does not cover all of the SG tubes. To qualify as a valid heat sink, the tubes must be covered. This is a more restrictive change. [.] INSERT 3A-3b Q3.4.5-2 Q3.4.5-3
1-16	A	Consistent with the intent of traveler TSTF-153, this change revises the note that permits up to 1 hour "deenergization" of RCP/RHR pumps. The revised wording clarifies the intent of the note to allow the pumps to be "removed from operation" instead of "deenergized", thus permitting other means of removing the pumps from service. With this change the pumps are not <u>required</u> to be deenergized to use the note (e.g. the pumps may be isolated, etc.). The change is considered to be administrative because from the standpoint of providing an exception to the LCO requirements (to maintain the operability and operation of the pumps), the revised wording is equivalent.
1-17	LG	Not applicable to WCGS. See conversion Comparison Table (Enclosure 3B).

INSERT 3A-3a

Q 3.4.5-1

The LCO and ACTION b of Specification 3.4.1.2, "Reactor Coolant System, Hot Standby," would be revised to require that two reactor coolant loops be OPERABLE. Loop operation requirements would also be revised to be contingent on Rod Control System status. The requirement to have a third OPERABLE reactor coolant loop would be deleted, consistent with NUREG-1431. This is acceptable because the MODE 3 decay heat removal requirements are sufficiently low that a single RCS loop with one RCP running is adequate to remove core decay heat. A second RCS loop ensures redundant capability for decay heat removal. When the Rod Control System is capable of rod withdrawal, two loops must be in operation to ensure accident analysis assumptions are satisfied. When rod withdrawal is precluded, only one loop is required to be in operation to satisfy MODE 3 accident analyses. The MODE 3 accident analyses which assume only two RCS loops in operation include the Uncontrolled RCCA Bank Withdrawal from Subcritical and the hot zero power RCCA ejection events. The initial conditions and analysis assumptions for these events will be unchanged since two loops must still be in operation during MODE 3 when the Rod Control System is capable of rod withdrawal. These reactivity transients rely on the Nuclear Instrumentation System's high flux trips for event termination which occurs very rapidly (on the order of seconds). There would be no benefit of having a third RCS loop OPERABLE for these transients since by the time the loop could be brought into operation, the event would be over for all practical purposes.

INSERT 3A-3b

Q 3.4.5-2
Q 3.4.5-3

Six percent of the narrow range span is specified at the higher temperatures of MODES 3 and 4 whereas 66% of the wide range span is specified for MODE 5. Both values ensure SG tubes are covered. The Emergency Operating Procedures cite the 6% narrow range level to ensure heat sink adequacy.

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: Q 3.4.5-2 **APPLICABILITY:** CA, WC

REQUEST: ITS SR 3.4.5.2 (also SR 3.4.6.2 and SR 3.4.7.2) (Callaway)
Change 1-15M

Comment: The sections of the ITS use the phrase "or equivalent" yet the term is not explained in the change or in t ITS Bases. According to the information provided narrow range level is used at th higher temperatures (Modes 3 and 4) and wide range level is used at the lower temperatures (Mode 5). If "or equivalent" means using the wide range at higher temperatures and the narrow range at lower temperatures are the levels specified applicable at the different temperatures? If not, what are the equivalent levels to the values specified in the ITS and how were they determined?

FLOG RESPONSE: At Callaway, the top of the highest steam generator (SG) tube is 344 inches above the tube sheet. The wide range instrumentation provides level indication from 7 inches above the tube sheet (0% indication) to the moisture separators (a range of 559 inches). The narrow range instrumentation provides level indication between 438 and 566 inches above the tube sheet for its 0-100% indication (the use of a common upper tap results in 100% level indication on both wide range and narrow range nominally being at the same 566 inches above the tube sheet). A calculation was performed to correlate the top of the highest tube to the wide range scale for MODE 5 conditions (the wide range instrumentation is calibrated for cold conditions), with margins added in for instrument loop errors and readability, resulting in the specified 67% wide range level. A minor error in the calculation was corrected, resulting in the specified 66% wide range level value cited in the attached pages for Callaway. Since the zero reference for the narrow range level instrumentation is nominally 96 inches above the top of the highest tube, the 4% value specified for MODES 3 and 4 was chosen since it is used throughout the EOPs for heat sink indication and is familiar to the operators. In the main control room there is one Class 1E wide range level indicator per SG and there are four narrow range level indicators per SG, of which three per SG are Class 1E. The "or equivalent" phrase would allow the use of wide range level instrumentation in MODES 3 and 4 in the unlikely event all narrow range level instrumentation were unavailable for a required SG; in MODE 3 this unlikely scenario would result in ITS LCO 3.3.3 non-compliance and would invoke Required Action(s) under PAM Instrumentation. Conversely, the "or equivalent" phrase would allow the use of narrow range level instrumentation in MODE 5 if the one wide range level indicator per SG were unavailable. This flexibility is similar to the approach under which Vogtle was licensed wherein their MODES 3-5 RCS specifications required SG water level to be above the highest point of the SG U-tubes. We are specifying water levels that ensure the same, yet allow the use of all available instrumentation. Before the "or equivalent" instrumentation were used in a given MODE, process measurement effects on the alternate instrument's calibrated span would be considered. Due to the unlikely event of either scenario presenting operational limitations, given the reduced RCS loop requirements in MODES 3-5 and the instrumentation redundancy, we do not see the need for a pre-determined correlation between the wide range and narrow range level indications; however, we reserve the right to exercise that option should the need arise.

Wolf Creek reviewed this particular comment for applicability to Wolf Creek and concurs with the use of the phrase "or equivalent" in the ITS and ITS Bases. Wolf Creek believes that it is appropriate to change their plant-specific value to 6% narrow range (including uncertainties) since it is used throughout the Emergency Operating Procedures (EMGs), it has operator awareness because of the EMG familiarity, and ensures an SG water level approximately 100 inches above the top of the highest SG tube. Wolf Creek has done a review of the drawings and design documents and has determined that for MODE 5 conditions (the wide range instrumentation is calibrated for cold conditions), 66% wide range level corresponds to the top of the highest tube, with margins added in for instrument loop errors and readability. The need for flexibility to use either narrow range or wide range indication is most evident when placing the SGs in wet layup conditions. The narrow range instruments are "jumpered" to indicate a constant 50% level. This precludes a feedwater isolation signal at approximately 78%. The operators use SG wide range indication to maintain and monitor SG level. Additionally, the narrow range instruments are calibrated for normal operating pressure and temperature conditions while the wide range instruments are calibrated for shutdown conditions.

The Callaway and Wolf Creek ITS Bases have been modified to explain the "or equivalent" phrase.

ATTACHED PAGES:

Encl. 2	4-2, 4-4, 4-5
Encl. 3A	3
Encl. 5A	3.4-11, 3.4-13, 3.4-15, 3.4-17
Encl. 5B	B 3.4-26, B 3.4-31, B 3.4-32, B 3.4-33, B 3.4-35, B 3.4-36
Encl. 6A	8
Encl. 6B	6

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least ~~three of the~~ two reactor coolant loops listed below shall be OPERABLE and at least two of these reactor coolant loops shall be in operation **when the Rod Control System is capable of rod withdrawal and at least one reactor coolant loop in operation when the Rod Control System is not capable of rod withdrawal**:* **1-14-LS-22**

a. Reactor Coolant Loop A ~~and its associated steam generator and reactor coolant pump,~~ **1-01-LG**

b. Reactor Coolant Loop B ~~and its associated steam generator and reactor coolant pump,~~

c. Reactor Coolant Loop C ~~and its associated steam generator and reactor coolant pump, and~~

d. Reactor Coolant Loop D ~~and its associated steam generator and reactor coolant pump.~~

APPLICABILITY: MODE 3.~~**~~ **1-04-M**

ACTION:

a. With ~~less than the above~~ one required reactor coolant loops ~~OPERABLE~~ **inoperable**, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours. **1-19-M**

b. With only one reactor coolant loop in operation, ~~restore at least two loops to operation within 72 hours or within 1 hour open the Reactor Trip System breakers~~ **and the rod control system capable of rod withdrawal, within 1 hour restore two loops to operation or place the rod control system in a condition incapable of rod withdrawal.** **1-14-LS-22**

c. With **four RCS loops inoperable or** no reactor coolant loop in operation, **immediately place the rod control system in a condition incapable of rod withdrawal,** suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and ~~immediately~~ initiate corrective action to return ~~the required~~ one reactor coolant loop to OPERABLE status and operation. **3-04-LS-29** **1-19-M**

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

6%, or equivalent

Q3.4.5-2, Q3.4.5-3

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side ~~wide~~ narrow range water level to be greater than or equal to 10% at least once per 12 hours. **1-15-M**

4.4.1.2.3 ~~At least two~~ The required reactor coolant loops shall be verified in operation ~~and circulating reactor coolant~~ at least once per 12 hours. **1-14-LS-22** **1-01-LG**

*All reactor coolant pumps may be ~~deenergized~~ removed from operation for up to 1 hour per 8 hour period provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature. **1-16-A** **1-06-M**

~~**See Special Test Exception Specification 3.10.4.~~ **1-04-M**

***No RCP shall be started with any RCS cold leg temperature ≤368°F unless the secondary side water temperature of each steam generator is ≤ 50°F above each of the RCS cold leg temperatures.** **1-05-M**

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s) and/or RHR pump, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side ~~wide~~ narrow range water level to be greater than or equal to ~~10%~~ at least once per 12 hours. 1-15-M

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation ~~and circulating reactor coolant~~ at least once per 12 hours. 1-01-LG

6%, or equivalent — Q 3.4.5-2 Q 3.4.5-3

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*** and either: 1-08-LS-2

a. One additional RHR loop shall be OPERABLE#, or

66%, or equivalent

Q3.4.5-2
Q3.4.5-3

b. The secondary side water level of at least two steam generators shall be greater than 10% of the ~~wide~~ narrow range. 1-15-M

wide

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator level as soon as possible.

b. With **required RHR loops inoperable or** no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to **OPERABLE status and** operation. 1-10-M

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours. 1-01-LG

(NEW) Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation at least once per 7 days. 1-11-M

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

##A reactor coolant pump shall not be started **with any RCS cold leg temperature ≤368°F** unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures. 1-05-M

*The RHR pump may be ~~deenergized~~ **removed from operation** for up to 1 hour **per 8 hour period** provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature. 1-16-A 1-06-M

**** All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.** 1-08-LS-2

CHANGE NUMBER	NSHC	DESCRIPTION
1-11	M	This change adds a new surveillance for verification of breaker alignment and power availability to the required pump not in operation. This change is in conformance with NUREG-1431 Rev. 1.
1-12	M	The Actions are changed to separate the required actions for only one required RHR loop OPERABLE and no required RHR loops OPERABLE. These revised Actions are consistent with the Actions which are required under this LCO in NUREG-1431 Rev. 1, and are more conservative than current required actions.
1-13	M	Not applicable to WCGS. See Conversion Comparison Table (Enclosure 3B).
1-14	LS-22	~~The LCO and Action b are revised to require only two loops OPERABLE with two loops in operation when the rod control system is capable of rod withdrawal and one loop in operation when the rod control system is not capable of rod withdrawal. This change is consistent with NUREG-1431 Rev. 1.~~ INSERT 3A-3a Q 3.4.5-1
1-15	M	A steam generator (SG) level corresponding to 10% of the wide range does not cover all of the SG tubes. To qualify as a valid heat sink, the tubes must be covered. This is a more restrictive change. [.] INSERT 3A-3b Q3.4.5-2 Q3.4.5-3
1-16	A	Consistent with the intent of traveler TSTF-153, this change revises the note that permits up to 1 hour "deenergization" of RCP/RHR pumps. The revised wording clarifies the intent of the note to allow the pumps to be "removed from operation" instead of "deenergized", thus permitting other means of removing the pumps from service. With this change the pumps are not <u>required</u> to be deenergized to use the note (e.g. the pumps may be isolated, etc.). The change is considered to be administrative because from the standpoint of providing an exception to the LCO requirements (to maintain the operability and operation of the pumps), the revised wording is equivalent.
1-17	LG	Not applicable to WCGS. See conversion Comparison Table (Enclosure 3B).

INSERT 3A-3a Q 3.4.5-1

The LCO and ACTION b of Specification 3.4.1.2, "Reactor Coolant System, Hot Standby," would be revised to require that two reactor coolant loops be OPERABLE. Loop operation requirements would also be revised to be contingent on Rod Control System status. The requirement to have a third OPERABLE reactor coolant loop would be deleted, consistent with NUREG-1431. This is acceptable because the MODE 3 decay heat removal requirements are sufficiently low that a single RCS loop with one RCP running is adequate to remove core decay heat. A second RCS loop ensures redundant capability for decay heat removal. When the Rod Control System is capable of rod withdrawal, two loops must be in operation to ensure accident analysis assumptions are satisfied. When rod withdrawal is precluded, only one loop is required to be in operation to satisfy MODE 3 accident analyses. The MODE 3 accident analyses which assume only two RCS loops in operation include the Uncontrolled RCCA Bank Withdrawal from Subcritical and the hot zero power RCCA ejection events. The initial conditions and analysis assumptions for these events will be unchanged since two loops must still be in operation during MODE 3 when the Rod Control System is capable of rod withdrawal. These reactivity transients rely on the Nuclear Instrumentation System's high flux trips for event termination which occurs very rapidly (on the order of seconds). There would be no benefit of having a third RCS loop OPERABLE for these transients since by the time the loop could be brought into operation, the event would be over for all practical purposes.

INSERT 3A-3b Q 3.4.5-2
Q 3.4.5-3

Six percent of the narrow range span is specified at the higher temperatures of MODES 3 and 4 whereas 66% of the wide range span is specified for MODE 5. Both values ensure SG tubes are covered. The Emergency Operating Procedures cite the 6% narrow range level to ensure heat sink adequacy.

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 narrow range water levels are ≥ ~~17%~~ ~~10% narrow range~~ 6%, or equivalent for required RCS loops.	12 hours Q3.4.5-2 Q3.4.5-3 ~~B-PS~~ 3.4-44
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

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
~~B.~~ ~~One required RHR loop inoperable.~~ ~~AND~~ ~~Two required RCS loops inoperable.~~	~~B.1~~	~~Be in MODE 5.~~	~~24 hours~~	3.4-02
~~C~~B. Required ~~RCS or RHR~~ loops inoperable. OR No RCS or RHR loop in operation.	~~C~~B.1 AND ~~C~~B.2	Suspend all operations involving a reduction of RCS boron concentration. Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately	3.4-02

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	12 hours	
SR 3.4.6.2	Verify SG secondary side narrow range water levels are ≥ ~~17~~ ~~10% narrow range~~ 6%, or equivalent for required RCS loops.	12 hours Q3.4.5-2 Q3.4.5-3	~~B-PS~~ 3.4-44

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or Q3.4.5-2 Q3.4.5-3

b. The secondary side wide range water level of at least two steam generators (SGs) shall be ≥ ~~17~~ ~~10% narrow range~~ 66%, or equivalent. B ~~B-PS~~ 3.4-44

NOTES

1. The RHR pump of the loop in operation may be ~~de-energized~~ removed from operation for ≤ 1 hour per 8 hour period provided: 3.4-01

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

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

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

3. No reactor coolant pump shall be started with ~~one or more~~ any RCS cold leg temperature~~s~~ ≤ ~~275°F~~ 368°F unless the secondary side water temperature of each SG is ≤ 50°F above each of the RCS cold leg temperatures. Ed. B-PS B

4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.7.2	Verify SG secondary side wide range water level is ≥ ~~17~~ ~~10% narrow range~~ 66%, or equivalent, in required SGs.	12 hours Q3.4.5-2 Q3.4.5-3
SR 3.4.7.3	Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days

~~B-BS~~ 3.4-44

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1 (continued)

considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 10% 6%, or equivalent for required RCS loops. The wide range level instrumentation may be used in MODES 3 and 4 in the event all narrow range instrumentation were unavailable for a required SG. If the SG secondary side narrow range water level is < 10% 6%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

Q3.4.5-2
Q3.4.5-3

SR 3.4.5.3

Verification that the required RCPs are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs.

REFERENCES

~~None.~~ 1. USAR, Section 15.4.6

2. NRC letter (W. Reckley to N. Carns) dated November 22, 1993: "Wolf Creek Generating Station - Positive Reactivity Addition; Technical Specification Bases Change."

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.2

6%, or equivalent, 6% Q3.4.5-2 Q3.4.5-3

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 10% for required RCS loops. If the SG secondary side narrow range water level is < 10%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

The wide range level instrumentation may be used in MODES 3 and 4 in the event all narrow range instrumentation were unavailable for a required SG.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

~~None.~~ 1. USAR, Section 15.4.6

2. NRC letter (W. Reckley to N. Carns) dated November 22, 1993: "Wolf Creek Generating Station - Positive Reactivity Addition; Technical Specification Bases Change."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops – MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 3) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification, but is not sufficient for the boron dilution analysis discussed below.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary side water levels above ~~10~~% to provide an alternate method for decay heat removal via ~~natural~~ natural circulation (Ref. 3).

wide range 66

Q3.4.5-2
Q3.4.5-3

(continued)

INSERT 3A-7a Q 5.2-1

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 surveillances. 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.

INSERT 3A-7b WC 5.0-004

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 are 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.

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 ITS SR 3.8.3.3. TSTF-237 — QS.5-1	No. Not in CTS.	No. Not in CTS.	Yes	Yes
02-17 LS-1	"Reactor Coolant Pump Flywheel" is being revised consistent with ~~WOG-85~~. The proposed changes provide an exception to the examination requirements in Regulatory Guide 1.14, Revison 1, "Reactor Coolant Pump Flywheel Integrity."	Yes	No. See Section 3/4.4, CN 10-03-LS	No. LAR submitted 12/3/96.	Yes
02-18 A	Revise the Radioactive Effluent Controls Program dose rate limits to reflect changes to 10 CFR Part 20. ~~a draft Generic Letter and a proposed traveler.~~	~~No. Already in CTS.~~ Yes QS.2-1	~~No. Already in CTS.~~ Yes	Yes	Yes QS.2-1
02-19 LS-2	The surveillance interval for verifying that other properties are within 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 (See CN 01-60-LS24).	No. Addressed in 3/4.8 (See CN 01-60-LS24).	Yes	Yes
02-20 A	Add the provisions of Specification SR 3.0.2 and SR 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
03-01 A	The method for submitting all reports is revised to be in accordance with 10 CFR 50.4.	Yes	Yes	Yes	Yes

02-21 A INSERT 3B-5b — WC 5.0-004

02-22 A INSERT 3B-5a — QS.2-1

02-23 LG INSERT 3B-5C — DC 5.0-004

INSERT 3B-5a Q 5.2-1

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE 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

INSERT 3B-5b WC 5.0-004

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
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 are 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

INSERT 3B-5c DC 5.0-004

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMMANCHE PEAK	WOLF CREEK	CALLAWAY
02-23 LG	DCPP Administrative Programs, CTS 6.8.4.d, "Backup Method for Determining Subcooling Margin," and 6.8.4.f, "Containment Poar and Turbine Building Cranes," were evaluated for reloaction outside the TS to a licensee-controlled document consiste with 10 CFR 50.36 screening criteria.	Yes	No	No	No

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 5.0-005 **APPLICABILITY:** WC

REQUEST: NRC letter dated February 27, 1998 reissued CTS page 6-18b due to an administrative error which inadvertently omitted a sentence from CTS 6.8.5. The CTS has been marked up to reflect the issuance of page 6-18b.

ATTACHED PAGES:

Encl. 2 6-18b

PROCEDURES AND PROGRAMS (Continued)

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion 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 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 $\leq 0.005\ L_a$ when pressurized to ≥ 10 psig.

The provisions of Technical Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

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

6.8.5 a. Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks

The program shall include:

1. The limits for concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM and a surveillance program to ensure the limits are maintained.

2. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an UNRESTRICTED AREA in the event of an uncontrolled release of the tanks' contents, consistent with Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases due to Waste Gas System Leak or Failure."

6.8.5 The following programs, relocated from the Technical Specifications to USAR Chapter 16, shall be implemented and maintained:

WC 5.0-005

ADDITIONAL INFORMATION COVER SHEET

ADDITIONAL INFORMATION NO: WC 5.0-006 **APPLICABILITY:** WC

REQUEST: ITS Section 5.5.11 is revised to correctly reflect that the Control Room Emergency Ventilation System - Filtration is part of the ITS 5.5.11b. program. Also editorial changes are made for consistency.

ATTACHED PAGES:

Encl. 5A 5.0-22, 5.0-23

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation System-Filtration	2000 cfm
Control Room Emergency Ventilation System - Pressurization	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	6500 cfm

B-PS

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal absorber shows a penetration and system bypass < [0.05]% when tested in accordance with Regulatory Guide 1.52, Revision 2~~, and ASME N510-1989~~ at the system flowrate specified below ~~[± 10%]~~. B-PS

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation System - Filtration	2000 cfm
Control Room Emergency Ventilation System - Pressurization	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	6500 cfm

WC 5.0-006

B-PS

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ~~≤~~ 30°C and greater than or equal to the relative humidity specified below. B-PS B-PS WC 5.0-ED

ESF Ventilation System	Penetration	RH
Control Room Emergency Ventilation System (Filtration / Pressurization)	2%	70%
Auxiliary/Fuel Building Emergency Exhaust	2%	70%

WC 5.0-006

B-PS

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

(continued)

5.5 Programs and Manuals

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 absorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, ~~and ASME N510-1989]~~ at the system flowrate specified below ~~[± 10%]~~. 5.5-11 B-PS

Emergency Ventilation System - WC 5.0-006

Emergency Ventilation System - WC 5.0-006

ESF Ventilation System	Delta P	Flowrate
Control Room Filtration System	6.6 in. W. G.	2000 cfm
Control Room Pressurization System	3.6 in. W. G.	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	4.7 in. W. G.	6500 cfm

B-PS

e. Demonstrate at least once per 18 months th. the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1975. 5.5-11 B-PS

Emergency Ventilation System - Pressurization

ESF Ventilation System	Wattage
Control Room	5 ± 1 kW
Auxiliary/Fuel Building	37 ± 3 kW

B-PS

Emergency Exhaust

WC 5.0-006

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

f. INSERT 5.0-23 - Q 5.5-7

5.5-19

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks ~~or fed into the offgas treatment system~~, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, Revision 0, July 1981, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Revision 2, July 1981, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures." B-PS B B-PS B-PS

(continued)