UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY	)	Docket No. 50-275 Facility Operating License No. DPR-80
Diablo Canyon Power Plant Units 1 and 2	)	Docket No. 50-323 Facility Operating License No. DPR-82

License Amendment Request
No. 97-09

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company hereby applies to amend its Diablo Canyon Power Plant Facility Operating License Nos. DPR-80 and DPR-82 (Licenses). The proposed changes convert the Technical Specification (TS) to improved TS based on NUREG-1431, Revision 1.

Information on the proposed TS change is provided in Attachments 3 through 21. The changes have been reviewed and do not involve a significant hazards consideration as defined in 10 CFR 50.92 or an unreviewed environmental question. Further, there is reasonable assurance that the proposed change will not adversely affect the health and safety of the public.

Sincerely,

Gregory M. Rueger

Subscribed and sworn to before me
this 2nd of June, 1997
County of San Francisco

Notary Public

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My Comm. Expires JUL 30, 1999

TECHNICAL SPECIFICATION CONVERSION APPLICATION GENERAL DESCRIPTION AND ASSESSMENT

A. BACKGROUND

The nuclear industry and the NRC have been working for several years to improve plant Technical Specifications (TS). In September 1992, the NRC issued NUREG-1431, Revision 0, Standard Technical Specifications - Westinghouse Plants," (Revision 1 was issued in April 1995),[1] as the basis for the improved Standard Technical Specification (ISTS) for Westinghouse plants. The ISTS accomplishes the following:

- Provides a new Use and Applications Section (Chapter 1) to provide a clear and detailed explanation for use of the ISTS (the format of the ISTS is completely revised to be more user friendly).

- Simplifies the TS by relocating various specifications, surveillance requirements (SRs), and much of the current detail of the TS to other licensing basis documents.

- Incorporates improvements in the TS such as eliminating unnecessary specifications, extending the time to perform required actions, and reducing the frequency of certain surveillance requirements.

- Provides a greatly expanded Bases section which includes the basis of each limiting condition for operation (LCO), action, and SR.

The NRC has been strongly encouraging the industry to adopt the NUREG-1431 format.

Joint Technical Specification Conversion

In October 1995, PG&E joined with Wolf Creek Nuclear Operating Corp. (Wolf Creek Generating Station), UE Company (Callaway Plant), and TU Electric (Comanche Peak Steam Electric Station) in a joint effort to convert the current TS (CTS). Mr. Don Woodlan is the chairman of the Joint Licensing Subcommittee (JLS) that coordinated the effort of the four utilities. A meeting was held with the director of NRR and the NRC staff on November 14, 1995, to discuss the joint effort of those utilities in converting to the ISTS. A working

[1] Throughout this submittal, any reference to the ISTS or NUREG-1431 specifically is a reference to the version of NUREG-1431 available on the NRC's bulletin board in April 1995.

level meeting was held with the NRC on December 14, 1995. The first joint meeting to review a conversion package was held on January 15 and 16, 1996.

The JLS is attempting to reduce the costs and approval times for the conversion application and for other license amendment requests (LARs). It is also a goal of the four utilities to make the improved TS (ITS) for all four plants as similar as possible. Such commonality should enhance operations and reduce costs in the future.

The conversion application was produced in a cooperative effort involving TU Electric, PG&E, Wolf Creek Nuclear Operating Corporation, and Union Electric Company (hereafter the "Group").

The NRC staff has stressed the value of licensees working together to increase standardization and to reduce the NRC resources needed to act on licensing matters. In response to these recommendations and in recognition of the benefits that result, PG&E chose to work jointly with the Group in the conversion of the Diablo Canyon Power Plant (DCPP) TS. PG&E believes the benefits, both near term and long term, clearly justify this action.

The Group jointly developed conversion applications based on NUREG-1431, Revision 1 (Ref. 1). Submittals for all four utilities address the generic features of the Group members CTS in an identical fashion, include comparison tables to correlate the Group member's conversion applications, and are being docketed at approximately the same time. The Group anticipates an approximately nine month review by the NRC with the resultant review cost savings for each utility as outlined at the previous meetings between the Group and the NRC. This conversion process has been based on the following understandings reached with the NRC:

1. Each plant may maintain its licensing basis as established by its CTS in the conversion process. With appropriate justification, a given utility may optimize their ITS based on another Group member's CTS. The goal is to maximize commonality.

2. Plant specific LARs will continue to receive timely consideration during the conversion process and especially during the NRC review cycle. The Group will screen and limit these to the extent possible, yet it must be recognized that LARs in support of reloads and LARs representing either safety issues or significant cost savings will receive due consideration. When possible, LARs submitted during the next 18 months will be jointly developed and submitted by the Group to conserve NRC review resources.

3. The effective date for new surveillance requirements with a fuel cycle frequency, imposed as a result of the conversion, will be the next refueling outage occurring after the implementation of the individual plant's amendment. A specific license condition is proposed below to incorporate this item.

4. Given the commitment to convert the TS, enforcement discretion will not be denied or delayed solely on the basis that a given plant has not yet converted, especially when the basis of the requested discretion is NUREG-1431, Revision 1. Each request for such discretion will be judged on its own merit.

Conversion Application

The proposed amendment represents a conversion from the CTS to ITS based on NUREG-1431, Revision 1, dated April 1995. As part of this submittal, the criteria contained in 10 CFR 50.36(c)(2)(ii) were applied to the CTS and using NUREG-1431 as a basis, used to develop the proposed ITS.

Enclosure 5A of each attachment provides a list of applicable travelers for the associated section of the ITS. The list identifies the traveler number (the Technical Specification Task Force (TSTF) number is provided if assigned, otherwise the owners group number is used), the traveler's status with respect to that section (i.e., incorporated, not incorporated), the difference number (s) used to discuss the differences from NUREG-1431, Revision 1, and comments. The comments are used to explain the manner in which the traveler is being addressed in the associated section when such explanations are deemed to be beneficial.

The JLS members generally incorporated travelers into the applicable TS section as they became available. Travelers may not have been incorporated for various reasons including: (1) the traveler was disapproved by the NRC, (2) the traveler was received too late to incorporate into the package, (3) the traveler contained changes which were not consistent with the plant specific design or CTS, etc. October 1996 was the cut-off date for travelers for this conversion effort. If a traveler was approved by the Westinghouse Owners Group (WOG) mini-group (or higher), it was considered and addressed in the conversion application (see Enclosure 5A in Attachments 4 through 18). Later travelers were only considered if there was a safety impact or a significant operational improvement.

Travelers are generally written to reflect a single change, and it would not be appropriate to incorporate a portion of the traveler without incorporating the entire traveler. However, in a few cases, multiple changes were incorporated

in a single traveler. Some of those changes may be appropriate for a given plant while other changes may not. Since travelers are approved by a majority vote, a majority of the WOG members may be served properly by the traveler but some individual plants may not. Travelers were generally incorporated in their entirety or not at all; however, in a few rare cases, only portions of a traveler were incorporated.

The traveler process is dynamic. Travelers continue to be generated, changed, approved, denied, and denied with comment. For those travelers which have changed status during the LAR review (e.g., have been revised or denied by the NRC), the JLS members will work with the NRC to properly address the changed status in the conversion LARs. In general, it is anticipated that most travelers which are denied by the NRC will be removed in supplements to each utilities LARs.

The JLS members used the NRC bulletin board version of NUREG-1431, Revision 1, dated April 1995. When the NRC made corrections to the bulletin board version, these editorial corrections were incorporated into the mark-up of NUREG-1431 without justification.

In order to address new SRs imposed by the TS approved and issued as a result of this LAR, the following license condition is proposed:

> For SRs not previously performed by existing SRs or other plant tests, the requirement will be considered met on the implementation date and the next required test will be at the interval specified in the TS.

B. DESCRIPTION OF PROPOSED TECHNICAL SPECIFICATION REQUESTS

The overall format for the conversion application is as follows:

- Cover letter
- Oath and Affirmation (Attachment 1)
- General Description and Assessment (Attachment 2)
- Tables of changes not within the scope of full conversion to the ISTS and of pending or proposed LARs which could impact the conversion application review (Attachment 3)
- Specific change descriptions and evaluations (Attachments 4 through 18)

Each of these attachments (4 through 18) include:

- Cover Sheet
- Index of Enclosures
- Enclosure 1 - Cross-Reference Tables

- Enclosure 2 - Mark-up of CTS (NUREG-1151)
- Enclosure 3A - Description of Changes to CTS
- Enclosure 3B - Conversion Comparison Table - CTS
- Enclosure 4 - No Significant Hazards Consideration (NSHC)
- Enclosure 5A - Mark-up of NUREG-1431 Specifications
- Enclosure 5B - Mark-up NUREG-1431 BASES
- Enclosure 6A - Differences From NUREG-1431
- Enclosure 6B - Conversion Comparison Table - NUREG-1431

Attachments 19 and 20 include the ITS and Bases (with proposed changes to the ISTS incorporated).

Evaluations of TS in accordance with 10 CFR 50.36 to determine the acceptability of relocating TS to licensee control were performed and are included in Attachment 21. This information was previously submitted to the NRC in LAR 96-01.

Electronic copies of various portions of this submittal (e.g., Attachments 2, 19 and 20; Enclosures 3A, 4, and 6A,) will be provided to facilitate review. These copies are being provided in a WordPerfect 6.1 format.

The conversion application does not contain a separate criteria application report. As previously discussed with the NRC, the same information has been integrated into the application. There will be no matrix of the LCOs versus the 10 CFR 50.36 criteria. The NRC and the Group agreed that a separate criteria application report was not necessary based on the degree to which each of the JLS members have already completed the "split" activity.

The methodologies used to mark-up the CTS and the ISTS are explained in the appropriate enclosures and attachments. These methodologies explain the techniques used and any abbreviations employed. As described in the methodology for Enclosure 2, the CTS has been marked up to denote the technical changes needed to convert the CTS to the ITS. The exceptions are the notes used to identify MODE change restrictions which are added to selected specifications. These notes retain needed restrictions which are otherwise removed by the change of scope in LCO 3.0.4 from the CTS to the ITS. These notes are not included in the CTS markup for the affected specification but are listed in the "LCO 3.0.4 evaluation matrix attached to LS-1 in Enclosure 4 of Attachment 6 (CTS Section 3/4.0 / ITS Section 3.0). Additionally, the mark-up of the CTS and ITS reflect the CTS as of April 1,1997. Changes made to the CTS as the result of LAs subsequent to April 1, 1997 will be reflected in a supplemental submittal to this LAR.

The conversion application also identifies changes to the Bases. Material deleted from the Bases is identified using the strikeout feature of WordPerfect. Material added to the Bases is identified using the redline feature of WordPerfect. Identification numbers are not assigned and justifications are not submitted for these changes. This approach had been discussed with the NRC during previous meetings and was determined to be acceptable.

Brackets ("[ ]") are used in some descriptions in Enclosures 3A, 3B, 4, 6A, and 6B. Brackets provide a clear, convenient means of denoting plant specific differences. This was determined to be the most efficient and effective way to identify such differences. Additionally, due to a delay in the completion of the reviews for DCPP, an additional set of brackets ("{ }") are used to identify changes incorporated into the DCPP submittal, but not included in the associated attachments of the other utilities.

The movement of a requirement from one specification in the CTS to a different specification in the ITS is denoted through the use of an "A" item number and description along with the cross-reference table.

In order to achieve as much consistency in the license requirements as possible, the JLS members adopted the following policy with respect to renumbering LCOs, Conditions, Required Actions, or SRs when converting from the ISTS to the ITS.

- In general, LCOs will not be renumbered if an LCO is deleted. The JLS members felt that if licensees renumber the LCOs, a strength of the ISTS would be weakened in that it will be more difficult to compare one plant to another. The JLS members may choose to renumber specifications if a traveler that does so is approved by the NRC. The JLS members will encourage the WOG mini-group responsible for traveler review to not include the renumbering of LCOs in future travelers.

- In Conditions and Required Actions, the steps will be re-lettered. The JLS members concluded that the use of "Not Used" for deleted steps was not conducive to clear understanding by the operator especially under the stress of abnormal plant conditions. Specifications 3.3.1, "Instrumentation - Reactor Protection System Instrumentation," and 3.3.2, "Instrumentation, Engineered Safety Features Actuation System Instrumentation are exceptions to this rule. The conditions in these two specifications are not being re-lettered even though some conditions may have been deleted for some plants.

- SRs will not be renumbered. The numbers for deleted SRs will be retained and labeled "Not Used" in the specification. If the SR is the last one in the specification, it will be deleted entirely.

C. ANALYSIS

The proposed changes to the CTS have been categorized into five general groupings. These groupings can be characterized as administrative changes, relocated changes, moved changes, more restrictive changes, and less restrictive changes.

Non-technical administrative changes ("A" changes) were intended to incorporate human factors principles into the form and structure of the ITS so that they would be easier to use for Operations personnel. Administrative changes are editorial in nature or involve the reorganization or reformatting of requirements without affecting technical content or operational requirements. The proposed changes include: (1) adopting the form and format of the ISTS and (2) reorganizing the specifications and the information within the specifications in a manner consistent with the ISTS.

Relocated changes ("R" changes), those current TS requirements which do not satisfy or fall within any of the four criteria specified in 10 CFR 50.36(C)(2)(ii) may be relocated to appropriate "licensee controlled documents." In the attachments, the document to which requirements are being relocated is generally identified. The relocated LCO portion of the CTS, which includes the system description, design limits, functional capabilities, and performance levels, will be relocated to a licensee controlled document. Changes to these licensee controlled documents will be made pursuant to 10 CFR 50.59 or other appropriate control mechanisms. These changes reduce the number of current TS requirements but the actual commitment to continue to perform the requirement will be unchanged upon implementation of ITS.

Material is relocated to the types of documents (licensee controlled documents) described below:

- Documents which have controls defined by regulations e.g., the Quality Assurance Program, 10 CFR 50.54(a), the Security Plan, 10 CFR 50.54(p), they Emergency Plan, 10 CFR 50.54 (q), and the Final Safety Analysis Report (FSAR), 10 CFR 50.59.

- Documents which have controls established by License Conditions (e.g., the Fire Protection Report for most plants).

- Documents which have controls established by the programs and manuals section of the Administrative Controls in the TS (TS 5.5 in the ISTS). For example, the Offsite Dose Calculation Manual, Ventilation Filter Testing Program, and TS Bases are documents whose controls are established by the TS.

- Documents which are incorporated into one of the documents identified above by reference and, as such, come under the same controls as the document into which it is incorporated (e.g., some licensees have created a specific a document [Equipment Control Guidelines for DCPP] which contains those specification relocated to licensee control. The ECGs containing relocated TS will be incorporated into the FSAR by reference, thus falling under 10 CFR 50. 59).

Moved changes ("LG" changes) are a subset of the relocated changes. Moved changes are those current TS descriptions or details which do not establish requirements but do provide information on how requirements are satisfied. As such, moved changes do not satisfy or fall within any of the four criteria specified in 10 CFR 50.59 or other appropriate control mechanisms. These changes reduce the complexity and detailed prescriptive nature of the TS not required to remain in the TS; however, these moved descriptive details will be unchanged upon implementation of the ITS.

More restrictive changes ("M" changes) are those which either are more conservative than corresponding requirements in the CTS, or are additional restrictions which are contained in NUREG-1431 but are not contained in the CTS. Examples of the more restrictive requirements include: planning an LCO on plant equipment which is not required by the CTS to be operable; more restrictive requirements to restore inoperable equipment; more restrictive SRs.

Less restrictive changes ("LS" and "TR" changes) are those where current requirements are relaxed or eliminated, or new flexibility is provided. The more significant "less restrictive" requirements are justified on a case-by-case basis. When requirements have been shown to provide little or no safety benefit, their removal from the TS may be appropriate. In most cases, relaxations result of: (a) generic NRC actions, (b) new NRC staff positions that have evolved from technological advancements and operating experience, or (c) resolution of the Owners Groups' comments on the ISTS. Generic relaxations contained in NUREG-1431 were reviewed by the staff and found to be acceptable because they are consistent with current licensing practices and NRC regulations. The licensee's design was reviewed to determine if the specific design and licensing basis are consistent with the technical and licensing basis are consistent with the technical basis for the model requirements in NUREG-1431, and thus, provides a basis for these revised TS. To be conservative, some

items have been identified as "less restrictive" even though the revision could be considered in compliance with the CTS. Making the item "less restrictive" is not intended to be an admission that the plants may not have been in compliance with the CTS in the past but rather is an attempt to avoid a potential area for unnecessary debate as the change can be properly addressed as a "less restrictive" change.

These administrative, relocated, moved, more restrictive, and less restrictive changes to the requirements of the CTS do not result in operations that will alter assumptions relative to mitigation of an analyzed accident or transient event.

In addition to the changes described above, the licensee proposed certain changes to the CTS that are both less restrictive and/or are not within the scope of application for conversion to the guidance of NUREG-1431. All of the differences will be reviewed by the NRC staff and a determination will be made regarding the approval or disapproval of each item as a part of this license amendment request. Specifically, the licensee identifies the instances where their submittal varied for the provisions of NUREG-1431 (see Attachment 3).

D. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Separate enclosures have been provided in attachments 4 through 18 to provide NSHC evaluations for the changes provided in the associated attachments. The conclusion of each of the evaluations is that a NSHC determination is justified.

E. ENVIRONMENTAL EVALUATION

An evaluation of the proposed changes has determined that these changes do not involve (i) a significant hazard consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increasing individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 50.22(b), an environmental assessment of the proposed changes is not required.

F. REFERENCES

1. NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 1, April 1995.

2. NUREG-1366, "Improvements to Technical Specification Surveillance Requirements."

3. Generic Letter 93-05, "Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation."

4. NRC letter from Mr. William T. Russell to Messrs. Lee Bush, Blair Wunderly, Brian Woods, and Ray Barker dated October 25, 1993.

5. NUREG-1024, "Technical Specification - Enhancing the Safety Impact."

6. NRC Administrative Letter 96-04, "Efficient Adoption of Improved Standard Technical Specifications," dated October 9, 1996.

7. Nuclear Energy Institute (NEI) 96-06, "Improved Technical Specifications Conversion Guidance," dated July 1996.

DIABLO CANYON TABLE OF CHANGES NOT WITHIN THE SCOPE OF FULL CONVERSION TO THE ISTS[2]

ITS Change No./CTS Change No.	Description
	ITS 1.0/CTS 1.0
ITS 1.1-7 CTS 01-01-A CTS 01-30-A	The definition of a CHANNEL FUNCTIONAL TEST (CFT) from the current Technical Specifications (CTS) was included in the ITS. However, this definition was revised in the improved Technical Specifications (ITS) to allow an actual signal, a required actuation, or any series of overlapping tests to be credited for satisfying the requirements of the test. The same changes were made to the CHANNEL OPERATIONAL TEST (COT) via TSTF-39. A COT and a CFT test similar channel functions.
	ITS 3.3/CTS 3/4.3
ITS 3.3-104 CTS 02-36-M	Action 15 was added to CTS 3.3.2 to describe the actions required when both first or second level 4kV undervoltage relays are inoperable. This change has been proposed in License Amendment Request (LAR) 97-02.
ITS 3.3-29 CTS 02-29-M	An Engineered Safety Features Actuation System function for the refueling water storage tank (RWST) level channels is added. This change will also be included in a separate LAR to be submitted by approximately June 30, 1997.
	ITS 3.4/CTS 3/4.4
ITS 3.4-45 CTS 04-01-LS	The low temperature overpressure protection (LTOP) system LCO note on centrifugal charging pump (CCP) swap is revised to allow both CCPs to be capable of injecting into the RCS for up to 4 hours throughout the LTOP applicability.
ITS N/A CTS 05-03-A	(See ITS 5.5.9, c.4.a.8, and Section 3/4.0, CN 1-15-A.) The definition of "Tube Inspection" is clarified to eliminate potential misunderstanding with regard to the required point of entry.
	ITS 3.5/CTS 3/4.5
ITS N/A CTS 01-08-A	The volume of the RWST and accumulators is revised to be expressed in percent level rather than gallons, as specified

[2] Changes to the ISTS except those which involve the incorporation of plant specific design information, which were developed as part of the industry traveler process, which are simple editorial corrections, or which incorporate CTS information; and changes to the CTS that do not merit a separate LAR.

	in the ITS and the CTS.
ITS 3.6/CTS 2/4.6	
ITS 3.6-10 CTS 09-01-A	The volume of the spray additive tank is revised to be expressed in percent volume rather than gallons as indicated in the ITS and the CTS.
ITS 3.6-13 CTS 07-10-LS9	A note is added to delete the surveillance requirement to leak test containment ventilation isolation valves with resilient seals if the flow path is isolated by a leak tested blank flange.
ITS 3.7/CTS 3/4.7	
ITS 3.7-01 CTS 01-01-A	This change modifies the power range neutron flux high trip setpoints to reflect a new algorithm used to determine the setpoints. The algorithm was introduced in Westinghouse Nuclear Safety Advisory Letter 94-001. This change will also be included in a future, separate LAR.
ITS - N/A CTS 05-02-LS11	The allowed outage time for the MSIVs is increased to 8 hours. NUREG-1431 has been revised via a traveler to provide an allowed outage time of 72 hours for the MSIVs.
ITS 3.7-15 CTS 09-01-M	Verification that a motive force is available to assure that valves in the ASW system that must be re-positioned can be repositioned is added to SR 3.7.8.1. This requirement is not in the CTS
ITS 3.8/CTS 3/4.8	
ITS 3.8-47 CTS 01-48-M	LCO 3.8.3, ACTION B. regarding stored diesel generator (DG) lube oil was changed from a per DG format to a plant wide, shared system bases similar to the diesel fuel oil supply
ITS 5.0/CTS 6.0	
ITS 5.7-1 CTS 03-11-A	Limitations in the Radioactive Effluent Controls Program, reporting requirements for the Occupational Radiation Exposure Report and the Annual Radiological Environmental Operating Report, and radiation limits for High Radiation Areas are revised to reflect the requirements of revised 10 CFR 20 (proposed specifications 5.5.4, 5.6.1, 5.6.3, and 5.7)

DIABLO CANYON PENDING OR PROJECTED LARS WHICH COULD IMPACT THE REVIEW OF THIS CONVERSION APPLICATION

Pending LARs:

LAR Number	Description	Comment
95-07	Relocation of Selected Technical Specifications in accordance with NRC Final Policy Statement and NUREG-1431, Rev. 1	LAR 97-09 will be revised to incorporate changes following approval of this LAR.
96-10	Revision of Technical Specification to Support Extended Fuel Cycles to 24 Months: Submittal #3	LAR 97-09 will be revised to incorporate changes following approval of this LAR.
97-01	Revision of Technical Specification to Support Extended Fuel Cycles to 24 Months: Submittal #4	LAR 97-09 will be revised to incorporate changes following approval of this LAR.
97-02	Revision of Technical Specifications 3/4.8.1.1 and 3/4.3.2	LAR 97-09 will be revised to incorporate changes following approval of this LAR.
97-03	Voltage-Based Alternate Steam Generator Tube Repair Criteria for Outside Diameter Stress Corrosion Cracking at Tube Support Plate Intersections	LAR 97-09 will be revised to incorporate changes following approval of this LAR.
97-04	Steam Generator Tube Alternate Repair Criteria for Indications in the Westinghouse Explosive Tube Expansion (WEXTEX) Region	LAR 97-09 will be revised to incorporate changes following approval of this LAR.
97-05	Revision of Technical Specification 3/4.7.3.1	This LAR adds requirements to the TS to maintain pressure on the component cooling water (CCW) surge tank. LAR 97-09 will be revised to incorporate changes following approval of this LAR.

97-07	Revision of Technical Specification to Support Extended Fuel Cycles to 24 Months: Submittal #5	LAR 97-09 will be revised to incorporate changes following approval of this LAR.
97-08	Revision of Technical Specification s to Apply Westinghouse Generic Best Estimate LOCA Analysis Methodology	LAR 97-09 will be revised to incorporate changes following approval of this LAR.

Anticipated LARs:

Description	Comments
Revision of TS 3/4.7.1.1 to Revise the Power Range Neutron Flux High Trip Setpoints with Inoperable Main Steam Safety Valves	The changes proposed in this have already been incorporated into LAR 97-09 and will be included in a separate LAR expected to be submitted approximately June 30, 1997.
Uprating of Unit 1 to 3411	
Revision of TS 3/4.3.2 to add ESFAS requirements for the Refueling Water Storage Tank Level Channels	The need for this LAR was identified during follow-up of an NRC finding. The LAR is expected to be submitted by approximately June 30, 1997. LAR 97-09 will be revised to incorporate the RWST level channels LAR upon approval.
Revision of TS 3/4.3.2 to Revise the First and Second Level Undervoltage Relay Setpoints Upon Installation of Automatic Load Tap Changing Startup Transformers	This LAR is required to support completion of the startup transformer replacement project scheduled for completion during the Unit 2 eighth refueling outage scheduled for January 1998. LAR 97-09 will be revised to incorporate the FLUR/SLUR setpoint change LAR upon approval.
Revision of TS 3/4.4.6.2 to Revise the Requirements Associated with the Measurement of Reactor Coolant System (RCS) Controlled Leakage	This LAR would clarify the method of performing testing to measure RCS controlled leakage (i.e. reactor coolant pump seal flow). Upon approval of this LAR, LAR 97-09 will be revised to reflect the clarification of the methodology. This LAR is expected to be submitted by approximately October 1, 1997.

Revision of TS to Add Requirements to Time Response Test the Main Feedwater Pump Trip Function	Submittal of this LAR is expected by August 1, 1997.

ACRONYMS AND ABBREVIATIONS

Below is provided a list of the more broadly used acronyms and abbreviations. The list is not intended to be a complete list of all acronyms. Acronyms of abbreviations which have only limited use and which are properly defined where used are not included in this list.

"[ ]"	Brackets which are used in enclosures 3A, 3B, 4, 6A and 6B to enclosure portions of the application which are specific to the conversion application in which the portions are contained. Other applications may have different in formation in that part of an otherwise generic part of the parallel conversion applications. Empty brackets indicate that one or more of the other parallel applications have plant specific information in that location.
"A"	Change code for an Administrative Change to the CTS
ADV	Atmospheric Dump Valve
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater
AOT	Allowed outage time
APP	Applicability
ASP	Alternate Shutdown Panel
"B"	Change code for "Bracketed" information in the ISTS which indicates that the bracketed information was adopted in the ITS
BDMS	Boron Dilution Mitigation System
"B-PS"	Change code for "Plant Specific" information which has been inserted in a "Bracketed" portion of the ISTS
BOP	Balance of plant
BWOG	Boiling Water Owners Group
BWR	Boiling Water Reactor
CEOG	Combustion Engineering Owners Group
CFR	Code of Federal Regulations
CN	Change number - a number assigned to a change to the CTS or the ISTS in the conversion application
COLR	Core Operating Limits Report
COT	Channel Operational Test
CPSES	Comanche Peak Steam Electric Station
CR	Control Room
CRC	Corporate Review Committee - generic term for the various corporate safety committees
CRVS	Control Room Ventilation System

ACRONYMS AND ABBREVIATIONS (cont.)

CTS	Current Technical Specifications
DBA	Design Basis Accident as defined by the plant specific licensing basis
DC	Diablo Canyon
DCPP	Diablo Canyon Power Plant
"ED"	Change code used to identify "Editorial" changes made to the ISTS as part of the conversion application
EFPD	Effective Full Power Days
encl.	Enclosure
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Feature Actuation System
FHA	Fuel Handling Accident as defined by the plant specific licensing basis
FHBVS	Fuel Handling Building Ventilation System
FLUR	First Level Undervoltage Relay
FSAR	Updated Final Safety Analysis Report per 10 CFR 50.71(e)
FW	Feedwater
Group	The four licensees (PG&E, TU, UE, and WCNOC) which have joined together to convert the CTS and to produce parallel conversion applications
Improved STS	Improved Standard Technical Specifications, NUREG-1431, Rev. 1, April 1995
Improved TS	Improved Technical Specifications - the proposed plant specific Technical Specifications developed from the ISTS
IR	Intermediate Range
ISTS	Improved Standard Technical Specifications, NUREG-1431, Rev. 1, April 1995
ITS	Improved Technical Specifications - the proposed plant specific Technical Specifications developed from the ISTS
JCRC	Joint Corporate Review Committee - A subcommittee of the CRCs for PG&E, TU, UE, and WCNOC organized to perform an initial joint CRC review for the various licensees.
JLS	Joint Licensing Subcommittee - A working group composed of members from PG&E, TU, UE and WCNOC to share resources and to work together in common licensing matters
LA	License Amendment

ACRONYMS AND ABBREVIATIONS (cont.)

LAR	License Amendment Request
LCD	Licensee Controlled Document - A plant specific document which has change controls which include the change criteria established by 10 CFR 50.59 (e.g., the FSAR), similar regulatory requirements (e.g., 10 CFR 50.54a for the QA Plan), or the Administrative Controls Section of the ITS (e.g., the ODCM).
LCO	Limiting Condition for Operation
LDCR	Licensing Document Change Request - the document or form to initiate changes to licensing documents such as the FSAR, TS, etc.
LER	Licensee Event Report
"LG"	Change code for a Less Restrictive Generic Change (moving technical or descriptive information to a licensee controlled document) to the CTS
LOOP	Loss of Offsite Power
"LS"	Change code for a Less Restrictive change to the CTS
LSSS	Limiting Safety System Setting
"M"	Change code for an More Restrictive change to the CTS
MFIV	Main Feedwater Isolation Valve
mini-group	WOG MERITS Mini-Group - the group of utilities within the WOG that are acting on potential generic changes to the ISTS
MSIV	Main Steam Isolation Valve
MSSV	Main Steam Safety Valve
N/A	Not applicable
NA	Not applicable
NEI	Nuclear Energy Institute
Not Used	Generic term use to hold a place in the numbering system for LCOs, SRs, etc to indicate a generic requirement which does not apply to that specific unit
NRC	U.S. Nuclear Regulatory Commission
NSHC	No Significant Hazards Consideration evaluation per 10 CFR 50.92
NSSS	Nuclear steam supply system
NUREG-	Generic designator used to identify reports issued by the NRC or NRC contractors
NUREG-1431	Improved Standard Technical Specifications, NUREG-1431, Rev. 1, April 1995
ODCM	Offsite Dose calculation Manual
OL	Operating License

ACRONYMS AND ABBREVIATIONS (cont.)

OOS	Out of Scope or beyond the scope of an ITS conversion
PAM	Post Accident Monitoring
PAMS	Post Accident Monitoring System
Para	Paragraph
PG&E	Pacific Gas and Electric Co.
PR	Power Range
"PS"	Change code for a Plant Specific change to the ISTS
PWR	Pressurized Water Reactor
QA	Quality Assurance
QPTR	Quadrant Power Tilt Ratio
"R"	Change code for a Relocation change (relocation to a licensee controlled document outside of TS) to the CTS
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RO	Reactor Operator
RSP	Remote Shutdown Panel
RTB	Reactor Trip Breaker
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWST	Reactor Water Storage Tank
SDM	Shutdown Margin
SE	NRC issued Safety Evaluation
SFDP	Safety Function Determination Program
SG	Steam Generator
SI	Safety Injection
SIS	Safety Injection Signal
SL	Safety Limit
SLUR	Second Level Undervoltage Relay
SR	Surveillance Requirement
SR	Source Range
SRC	Safety Review Committee - Generic term for the various safety committees for the participating licensees
SRO	Senior Reactor Operator
SSPS	Solid State Protection System
STA	Shift Technical Advisor
STB	Staggered Test Basis
STS	Standard Technical Specifications
TADOT	Trip Actuating Device Operational Test
TDAFW	Turbine-driven Auxiliary Feedwater
TRM	Technical Requirements Manual
TS	Technical Specifications

ACRONYMS AND ABBREVIATIONS (cont.)

TSTF	Technical Specification Task Force
TU	TU Electric
UFSAR	Updated Final Safety Analysis Report per 10 CFR 50.71(e)
Updated FSAR	Updated Final Safety Analysis Report per 10 CFR 50.71(e)
USAR	Updated Final Safety Analysis Report per 10 CFR 50.71(e)
UV	Undervoltage
"TR"	Change code for a Technical Change (recurring - less restrictive) to the CTS
UE	Union Electric Co.
WC	Wolf Creek
WCAP-	Generic designator used to identify reports issued by Westinghouse
WCNOC	Wolf Creek Nuclear Operation Corp.
VFTP	Ventilation Filter Testing Program
WOG	Westinghouse Owners Group

SCREENING FORMS FOR TS TO BE RELOCATED

Screening Forms for the following TS are attached:

Reactivity Control Systems

3.1.2.1	Boration Systems Flow Path - Shutdown
3.1.2.3	Charging Pumps - Shutdown
3.1.2.4	Charging Pumps - Operating
3.1.2.5	Borated Water Source - Shutdown
3.1.2.6	Borated Water Sources - Operating

Instrumentation

3.3.3.2	Movable Incore Detectors
3.3.3.4	Meteorological Instrumentation
3.3.3.10	Explosive Gas Effluent Monitoring Instrumentation

Refueling Operations

3.9.3	Decay Time
3.9.5	Communications
3.9.6	Manipulator Crane
3.9.7	Crane Travel - Fuel Handling Building
3.9.10.2	Water Level - Reactor Vessel (Control Rods)
3.9.13	Spent Fuel Shipping Cask Movement

Special Test Exceptions

3.10.1	Special Test Exceptions - Shutdown Margin
3.10.4	Position Indication System - Shutdown

Radioactive Effluents

3.11.1.4	Liquid Holdup Tanks
3.11.2.5	Explosive Gas Mixture
3.11.2.6	Gas Storage Tanks

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.1.2.1 BORATION FLOW PATHS - SHUTDOWN

Applicable Modes: Modes 5 and 6

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
__	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
__	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification (TS) shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

The Bases for this limiting condition for operation (LCO) state that the purpose is to assure negative reactivity control is available during each mode of facility operation.

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

The boration subsystem is not assumed to operate to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS, which causes a boron dilution event; the response, or that required by the operator, is to close the appropriate valves in the reactor makeup system before the SDM is lost. Operation of the boration subsystem is not assumed to mitigate this event. Furthermore, Ref. 3 notes that the normal capability to control reactivity with boron is not credited in the

accident analysis. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully withdrawn. When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

Based on the foregoing, the boration subsystem is not installed instrumentation that is used to detect or indicate a significant degradation of the reactor coolant pressure boundary (RCPB); therefore, this TS does not satisfy criterion 1.

The boration subsystem TS is not associated with a process variable, design feature, or operating restriction that is an initial condition of an event that assumes failure of or challenges the integrity of a fission product barrier. Therefore, the boration subsystem TS does not satisfy criterion 2.

For these events, the primary success path for mitigation includes isolating the dilution flowpath. The subsequent actuation of equipment to establish a boron injection flowpath is intended to regain the required SDM. This is desirable, but beyond the scope of a primary success path action. The boration subsystem TS does not apply to a system that is part of the primary success path, and which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; therefore, the TS does not satisfy criterion 3.

The boration flow paths at shutdown are not modeled in the Diablo Canyon Power Plant (DCPP) Individual Plant Examination (IPE), as the IPE only considers power operation (Mode 1). However, there is no indication that this function would be identified as risk significant if it was modeled in probabilistic risk assessment (PRA) models. Therefore, this TS does not satisfy criterion 4.

(4) CONCLUSION

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.1.2.3 CHARGING PUMPS - SHUTDOWN

Applicable Modes: Modes 5 and 6

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
__	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
__	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

The Bases for this LCO state that the purpose is to assure negative reactivity control is available during each mode of facility operation. Equipment required to perform this function includes: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power source from operable diesel generators.

The boration subsystem is not assumed to operate to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS, which causes a boron dilution event; the response, or that required by the operator, is to close the appropriate valves in the reactor makeup system before the SDM is lost. Operation of the boration subsystem is not assumed to mitigate this event. Furthermore, Ref. 3 notes that the normal capability to control reactivity with boron is not credited in the accident analysis. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully

withdrawn. When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

The boration subsystem TS is not applicable to installed instrumentation used to detect or indicate a significant degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The boration subsystem TS is not associated with a process variable, design feature, or operating restriction that is an initial condition of an event that assumes failure of or challenges the integrity of a fission product barrier. Therefore, the boration subsystem TS does not satisfy criterion 2.

For these events, the primary success path for mitigation includes isolating the dilution flowpath. The subsequent actuation of equipment to establish a boron injection flowpath is intended to regain the required SDM. This is desirable, but beyond the scope of a primary success path action. The boration subsystem TS does not apply to a SSC that is part of the primary success path and which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; therefore, the TS does not satisfy criterion 3.

The charging pumps at shutdown are not modeled in the DCPP IPE, as the IPE only considers power operation (Mode 1). However, there is no indication that this function would be identified as risk significant if it was modeled in PRA models. Therefore, this TS does not satisfy criterion 4.

(4) CONCLUSION

___ This Technical Specification is retained.

_X_ The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.1.2.4 CHARGING PUMPS - OPERATING

Applicable Modes: Modes 1, 2, 3 and 4#

(# a maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F).

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
__	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
__	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

The Bases for this LCO state that the purpose is to assure negative reactivity control is available during each mode of facility operation. The equipment required to perform this function includes: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from operable diesel generators.

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

The boration subsystem is not assumed to operate to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS, which causes a boron dilution event; the response, or that required by the operator, is to close the appropriate valves in the reactor makeup system before the SDM is lost. Operation of the boration subsystem is not assumed to mitigate this event. Furthermore, Ref. 3 notes that the normal capability to control reactivity with boron is not credited in the accident analysis. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully withdrawn. When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

Based on the foregoing, the boration subsystem is not installed instrumentation that is used to detect or indicate a significant degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The boration subsystem TS is not associated with a process variable, design feature, or operating restriction that is an initial condition of an event that assumes failure of or challenges the integrity of a fission product barrier. Therefore, the boration subsystem TS does not satisfy criterion 2.

For these events, the primary success path for mitigation includes isolating the dilution flowpath. The subsequent actuation of equipment to establish a boron injection flowpath is intended to regain the required SDM. This is desirable, but beyond the scope of a primary success path action. The boration subsystem TS does not apply to a system that is part of the primary success path, and which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; therefore, the TS does not satisfy criterion 3. Ref. 3 also notes that operability of the charging pumps, the refueling water storage tank (RWST) and associated flowpaths is required as part of the emergency core cooling system (ECCS) TS.

For the main steamline break (MSLB) event, the sequence of events takes the plant to cold shutdown conditions and; therefore, boration of the RCS is necessary. However, the boration flowpath in this case is required as part of the ECCS function.

The ECCS function of the charging pumps is explicitly modeled in the DCPP IPE; this function is being retained in the DCPP ECCS TS. The boration function of the charging pumps is not explicitly modeled in the DCPP PRA; however, the boration function in response to anticipated transient without scram (ATWS) events is considered. The ATWS contribution to core damage is small, less than 1E-6. Thus, it can be concluded that this TS does not satisfy criterion 4.

(4) **CONCLUSION**

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.1.2.5 BORATED WATER SOURCE - SHUTDOWN

Applicable Modes: Modes 5 and 6

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
__	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
__	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

The Bases for this LCO state that the purpose is to assure negative reactivity control is available during each mode of facility operation. Equipment required to perform this function includes, depending on operating conditions, a combination of: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power source from operable diesel generators.

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

The boration subsystem is not assumed to operate to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS, which causes a boron dilution event; the response, or that required by the operator, is to close the

appropriate valves in the reactor makeup system before the SDM is lost. Operation of the boration subsystem is not assumed to mitigate this event. Furthermore, Ref. 3 notes that the normal capability to control reactivity with boron is not credited in the accident analysis. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully withdrawn. When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

The boration subsystem TS is not applicable to installed instrumentation used to detect or indicate a significant degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The boration subsystem TS is not associated with a process variable, design feature, or operating restriction that is an initial condition of an event that assumes failure of or challenges the integrity of a fission product barrier. Therefore, the boration subsystem TS does not satisfy criterion 2.

For these events, the primary success path for mitigation includes isolating the dilution flowpath. The subsequent actuation of equipment to establish a boron injection flowpath is intended to regain the required SDM. This is desirable, but beyond the scope of a primary success path action. The boration subsystem TS does not apply to a system that is part of the primary success path, and which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; therefore, the TS does not satisfy criterion 3.

The borated water sources at shutdown are not modeled in the DCPP IPE, as the IPE only considers power operation (Mode 1). However, there is no indication that this function would be identified as risk significant if it was modeled in PRA models. Therefore, this TS does not satisfy criterion 4.

(4) CONCLUSION

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.1.2.6 BORATED WATER SOURCES - OPERATING

Applicable Modes: Modes 1, 2, 3, and 4

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
__	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
__	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

The Bases for this LCO state that the purpose is to assure negative reactivity control is available during each mode of facility operation. The equipment required to perform this function includes, depending upon operating conditions, combinations of: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from operable diesel generators.

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

The boration subsystem is not assumed to operate to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS, which causes a boron

dilution event; the response, or that required by the operator, is to close the appropriate valves in the reactor makeup system before the SDM is lost. Operation of the boration subsystem is not assumed to mitigate this event. Furthermore, Ref. 3 notes that the normal capability to control reactivity with boron is not credited in the accident analysis. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully withdrawn. During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCOs 3.1.3.5 and 3.1.3.6, for rod insertion.

Based on the foregoing, the boration subsystem TS is not applicable to installed instrumentation used to detect or indicate a significant degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The boration subsystem TS is not associated with a process variable, design feature, or operating restriction that is an initial condition of an event that assumes failure of or challenges the integrity of a fission product barrier. Therefore, the boration subsystem is not a design feature required to be operable to mitigate these events, and this TS does not satisfy criterion 2.

For these events, the primary success path for mitigation includes isolating the dilution flowpath. The subsequent actuation of equipment to establish a boron injection flowpath is intended to regain the required SDM. This is desirable, but beyond the scope of a primary success path action. The boration subsystem TS does not apply to a system that is part of the primary success path, and which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; therefore, the TS does not satisfy criterion 3. Ref. 3 also notes that operability of the charging pumps, the RWST and associated flowpaths is required as part of the ECCS TS.

For the MSLB event, the sequence of events takes the plant to cold shutdown conditions and; therefore, boration of the RCS is necessary. However, the boration flowpath in this case is required as part of the ECCS function.

The ECCS function of the RWST is explicitly modeled in the DCPP IPE; this function is being retained in the DCPP ECCS TS. The boration function of the RWST and boric acid storage system is not explicitly modeled in the DCPP PRA; however, the boration function in response to ATWS events is considered. The ATWS contribution to core damage is small, less than 1E-6. Thus, it can be concluded that this TS does not satisfy criterion 4.

(4) CONCLUSION

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.3.3.2 MOVABLE INCORE DETECTORS

Applicable Modes: When the Movable Incore Detection System is used for:
a. Recalibration of the Excore Neutron Flux Detection System, or
b. Monitoring the Quadrant Power Tilt Ratio, or
c. Measurement of $F^N_\Delta H$, $F_Q(Z)$ and F_{xy}.

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
—	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
—	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

This TS requires the movable incore detectors to be operable, within defined conditions, whenever the system is used for recalibration of excore detectors, monitoring the quadrant power tilt ratio, or measurement of F_Q and F^N-Delta H. If the system is not operable, the required action is not to use the system for these purposes. The requirements for maintaining F_Q and F^N-Delta H within limits are addressed in the TS for power distribution limits. Furthermore, the measurements are used in a confirmatory manner and do not provide direct input to reactor protection system or engineered safety features actuation system functions.

Ref. 1 states that the operability of the movable incore detectors ensures the accurate measurement of spatial neutron flux distribution of the core.

Ref. 3 notes that the movable incore detector system is not installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the RCPB. Also, the system is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Further, the movable incore detector system is not an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The movable incore detector TS is not applicable to installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the RCPB. The movable incore detector TS is associated indirectly with monitoring an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, this initial condition is only required to be monitored periodically by incore detectors.

The movable incore detector TS does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

From Ref. 3, the movable incore detectors have not been shown to be a significant risk contributor to public health and safety by either operational experience or PSA. The detectors are used only for periodic surveillance of the core power distribution and for calibration of the excore detectors and do not initiate any automatic protection action. The detectors are not modeled in the DCPP IPE.

Based on the above, the TS does not satisfy criteria 1, 2, 3 or 4.

(4) CONCLUSION

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.3.3.4 METEOROLOGICAL INSTRUMENTATION

Applicable Modes: At all times

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
—	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
—	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

The meteorological instrumentation ensures that data is available to estimate potential radiological doses to the public from accidental or routine releases of radioactive materials to the atmosphere. The instrumentation is used to assess the need for recommending protective measures following an accident. The meteorological instrumentation is not used to mitigate a DBA or transient.

Ref. 3 evaluated this instrumentation and concluded that it is not installed instrumentation that is used to detect degradation of the RCPB. Neither is it assumed to function in the safety analysis and is not an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The meteorological instrumentation TS is not applicable to installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the RCPB. Therefore, this TS does not satisfy criterion 1.

The meteorological instrumentation TS is also not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 2.

The meteorological instrumentation TS does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

The meteorological instrumentation is not modeled in the DCPP PRA. However, there is no indication that this function would be identified as risk significant if it was modeled in PRA models. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.3.3.10 EXPLOSIVE GAS MONITORING INSTRUMENTATION

Applicable Modes: During Gaseous Radwaste System operation

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
—	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
—	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident (DBA) or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

The explosive gas monitoring instrumentation provides the capability to detect the concentration of oxygen and hydrogen in the waste gas holdup system and provide an alarm if the concentrations exceed prescribed limits. According to LCO 3.3.3.10, this TS assures the operability of the instrumentation required for TS 3.11.2.5, "Explosive Gas Mixture of the Gaseous Effluents." According to the Bases of TS 3.3.3.10 and 3.11.2.5, the purpose of the limits on explosive gas concentrations and the monitoring instrumentation is to prevent an explosion in the waste gas holdup system. An explosion could result in a release of radioactive materials contained in the gaseous waste holdup system. Although release of the contents of a waste gas decay tank is an analyzed DBA, the analysis assumes that the tank fails and the contents are released without any mitigating circumstances. Therefore, the explosive gas limits are not an initial condition of a DBA.

The explosive gas monitoring instrumentation is not applicable to installed instrumentation used to detect and indicate in the control room a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The explosive gas monitoring instrumentation is not applicable to a process variable, design feature, or operating restriction that is an initial condition of any DBA or transient analysis since the tank failure is assumed as the initiating event for the release. Thus, this TS does not satisfy criterion 2.

The explosive gas monitoring instrumentation is not assumed to function in the safety analysis. It is not a part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 3.

From Ref. 3, the explosive gas monitoring instrumentation has not been shown to be a significant risk contributor to public health and safety by either operational experience or PSA. The function of this instrumentation is to preclude inadvertent radioactivity releases from the waste gas holdup system due to a tank failure from a waste gas explosion. Severe accidents dominate public risk, not inadvertent releases. This system is not modeled in the DCPP IPE. Thus, this instrumentation does not satisfy criterion 4.

(4) CONCLUSION

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

(The TS will be relocated but a program statement will be added to Administrative Controls Section)

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.9.3 DECAY TIME

Applicable Modes: During movement of irradiated fuel in the reactor vessel

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
—	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
—	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

This specification places a time limit on reactor subcriticality prior to the movement of irradiated fuel assemblies in the reactor vessel. This ensures that sufficient time has elapsed for the radioactive decay of short-lived fission products. The decay of short-lived fission products is assumed in the fuel handling accident.

Decay time is not installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. Decay time does not satisfy criterion 1.

Decay time is an operating restriction that is an initial condition of a DBA that assumes the failure of the integrity of a fission product barrier. However, it was agreed upon in Industry/NRC meetings during the development of NUREG-1431 that this LCO may be relocated. This LCO is not contained in Ref. 2. However, the requirement for a minimum decay time of 100 hours prior to fuel handling is contained in the Bases of NUREG-1431, Rev. 1 (B 3.9.7). DCPP will be consistent with the

decay time limit in the Bases of NUREG-1431, Rev. 1 upon implementation of the new standard TS. Based on NRC determination in Ref. 2, the screening criterion application question 2 may be answered with a "no".

Decay time is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either the failure of or presents a challenge to the integrity of a fission product barrier. Decay time does not satisfy criterion 3.

Decay time is not modeled in the DCPP IPE. However, there is no indication that this function would be identified as risk significant if it was modeled in PRA models. Therefore, this TS does not satisfy criterion 4.

(4) CONCLUSION

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.9.5 COMMUNICATIONS

Applicable Modes: During Core Alterations

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
—	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
—	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

This specification requires communication between the control room and the refueling bridge to ensure that any abnormal change in the facility status or core reactivity observed on the control room instrumentation can be communicated to the refueling bridge personnel during core alterations.

The TS requirements for communications are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The communications TS is not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this requirement does not meet criterion 2.

The TS for refueling communications does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient

that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, the requirements do not satisfy criterion 3.

Communications during core alterations is not modeled in the DCPP IPE. However, there is no indication that this function would be identified as risk significant if it was modeled in PRA models. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.9.6 MANIPULATOR CRANE

Applicable Modes: During movement of control rods or fuel assemblies within the reactor vessel

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
—	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
—	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

This TS ensures that the lifting device on the manipulator crane has adequate capacity to lift the weight of a fuel assembly and a rod control cluster assembly, and that an automatic load limiting device is available to prevent damage to the core internals or reactor vessel. This TS also ensures that the auxiliary hoist on the manipulator crane has adequate capacity for movement of control rods and fuel assemblies.

The TS requirements for the manipulator crane are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The manipulator crane TS is not associated with a process variable, design feature, or operating restriction that is monitored and controlled and is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this requirement does not meet criterion 2.

The TS for the manipulator crane does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

The requirements of this technical specification are not a significant risk contributor to public health and safety by either operational experience or PSA. The manipulator crane is used to transport fuel assemblies during refueling operations. The DCPP IPE models the plant during power operations, and therefore does not include the manipulator crane in any risk quantifications. However, if the manipulator crane were included in the model, its significance would be negligible. Therefore, these requirements do not satisfy criterion 4.

(4) CONCLUSION

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING

Applicable Modes: With fuel assemblies in the spent fuel pool

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
—	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
—	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

This specification ensures that loads in excess of one fuel assembly containing a control rod, plus the weight of the fuel handling tool, will not be moved over other fuel assemblies stored in the spent fuel storage racks. Therefore, in the event of a drop of this load, the activity released is limited to that contained in one fuel assembly. This also prevents any possible distortion of fuel assemblies in the storage racks from achieving a critical configuration. This specification applies to prevention of a heavy load drop accident and assures that the damage caused by the load is limited to the equivalent of one spent fuel assembly. This assumption is consistent with the activity release assumed in the DBA safety analyses for a fuel handling accident.

The TS requirements for crane travel are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The fuel handling building crane travel TS is associated with an operating restriction for a heavy load drop event. This specification is not applicable to a process variable, design feature, or operating restriction that is monitored and controlled during power

operation and is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this requirement does not meet criterion 2. This conclusion is consistent with the corresponding evaluation in Ref. 4.

The TS for crane travel does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, these requirements do not satisfy criterion 3.

From Ref. 3, the fuel handling building crane has not been shown to be a significant risk contributor to public health and safety by either operational experience or PSA. Ref. 3 reviewed several environmental reports related to these cranes, and found their risk significance to be minimal. The spent fuel storage facility crane is not modeled in the DCPP IPE. Therefore, these requirements do not satisfy criterion 4.

(4) CONCLUSION

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.9.10.2 WATER LEVEL - REACTOR VESSEL (CONTROL RODS)

Applicable Modes: During movement of control rods within the reactor pressure vessel while in Mode 6

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
__	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
__	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

This specification places a lower limit on the amount of water above the top of the fuel assemblies in the reactor vessel during movement of control rods. The Bases state that this ensures the water removes 99 percent of the assumed 10 percent iodine gap activity released from the rupture of an irradiated fuel assembly in the event of a fuel handling accident (FHA) during core alterations. However, the movement of control rods is not associated with the initial conditions of an FHA, and the Bases do not address any concerns regarding inadvertent criticality which could lead to a breach of the fuel rod cladding. Inadvertent criticality during Mode 6 is prevented by maintaining proper boron concentration in the coolant in accordance with LCO 3.9.1.

The TS requirements for water level - reactor vessel are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The water level - reactor vessel (control rods) TS are not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this requirement meets criterion 2.

The TS for water level - reactor vessel do not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, these requirements do not satisfy criterion 3.

The reactor water level during movement of control rods while in Mode 6 is not modeled in the DCPP IPE. However, there is no indication that this function would be identified as risk significant if it was modeled in PRA models. Therefore, this TS does not satisfy criterion 4.

(4) CONCLUSION

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.9.13 SPENT FUEL SHIPPING CASK MOVEMENT

Applicable Modes: During all cask handling operations

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
__	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
__	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

The restriction on spent fuel shipping cask movement ensures that no fuel assemblies will be ruptured in the event of a spent fuel shipping cask accident. The dose consequences of this accident are within the dose guideline values of 10 CFR Part 100.

Spent fuel cask handling and the spent fuel cask drop accident are addressed in FSAR Section 9.1.2.3.1 as revised by a PG&E approved 10 CFR 50.59 evaluation. Cask handling is addressed in general terms for a typical fuel cask shown in FSAR Figure 9.1-4, which is used as the basis for discussion of cask handling and the spent fuel pool cask drop accident in the FSAR. Prior to cask movement in the fuel handling building, a detailed evaluation and analysis using specific cask parameters will be performed in accordance with 10 CFR 50.59, and cask handling procedures will be revised or developed, as necessary, to reflect the results of the evaluations.

This specification does not contain requirements for installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. This specification does not satisfy criterion 1.

This specification does not contain requirements on a process variable, design feature, or operating restriction that is monitored or controlled during power operation and is an initial condition of DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. This specification does not satisfy criterion 2.

This specification does not contain requirements for a SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. This specification does not satisfy criterion 3.

Spent fuel shipping cask movement is not modeled in the DCPP IPE. However, there is no indication that this function would be identified as risk significant if it was modeled in PRA models. Therefore, this TS does not satisfy criterion 4.

(4) CONCLUSION

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.10.1 SPECIAL TEST EXCEPTION - SHUTDOWN MARGIN

Applicable Modes: Mode 2

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
__	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
__	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

Ref. 4 states: "Special Test Exceptions 3.10.1 through 3.10.4 may be included with corresponding LCOs which are remaining in TS. Special Test Exception 3.10.5 [which is DCPP TS 3.10.4] may be relocated outside of TS along with LCO 3.1.3.3."

LCO 3.10.1 is only applicable in Mode 2. The shutdown margin requirements for Modes 1 and 2 are retained in other reactivity control system TS; therefore, LCO 3.10.1 may be deleted. This conclusion is consistent with Ref. 4. However, DCPP has chosen to relocate TS 3.10.1 to the DCPP Equipment Control Guidelines and will address the entire reactivity control system TS during the conversion phase of the new standard technical specification program.

Shutdown margin during physics tests is not modeled in the DCPP PRA. However, there is no indication that this function would be identified as risk significant if it was modeled in PRA models. Therefore, this TS does not satisfy criterion 4.

(4) CONCLUSION

___ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.10.4 SPECIAL TEST EXCEPTION - POSITION INDICATION SYSTEM - SHUTDOWN

Applicable Modes: Modes 3, 4, and 5 during performance of rod drop time measurements and during surveillance of digital rod position indicators for Operability

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
—	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
—	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

Ref. 4 states that Special Test Exceptions 3.10.1 through 3.10.4 may be included with corresponding LCOs which are remaining in TS. Furthermore, Ref. 4 states that Special Test Exception 3.10.5 (DCPP TS 3.10.4) may be relocated outside of TS along with LCO 3.1.3.3.

In accordance with LAR 95-07, Attachment D, "Screening Form for TS 3.1.3.3," may be relocated from the TS. Therefore, TS 3.10.4 may be relocated.

(4) CONCLUSION

— This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.11.1.4 LIQUID HOLDUP TANKS

Applicable Modes: At all times

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
__	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
__	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES" then the TS shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

The liquid holdup tank specifications impose limits on the quantity of radioactive material contained in specific outdoor tanks that may contain radwaste. Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentration would be less than the limits of 10 CFR 20, Appendix B. Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area. The tanks addressed by this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

The TS requirements for liquid holdup are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The liquid holdup TS are not associated with a process variable, design feature, or operating restriction that is monitored or controlled during power operation and is an initial condition of a DBA or transient analysis that either assumes the failure of or

presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The TS for liquid holdup do not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

From Ref. 3, the liquid holdup tanks, which hold radwaste, have not been shown to be a significant risk contributor to public health and safety by either operational experience or PSA. Risk of radioactivity release is dominated by severe accidents, not releases of radionuclides generated from normal operations. The liquid holdup tanks are not modeled in the DCPP IPE. Therefore, this TS do not satisfy criterion 4.

(4) **CONCLUSION**

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

(The TS may be relocated but a program statement will be added to Administrative Controls section).

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.11.2.5 EXPLOSIVE GAS MIXTURE

Applicable Modes: At all times

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
—	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
—	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
—	X	(4)	An SSC which operating experience or probabilistic safety assessment (PSA) has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the TS shall be included in the new TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining these limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of GDC 60 of Appendix A to 10 CFR 50. The safety analysis concerning the gaseous radwaste system assumes that a storage tank ruptures, from unspecified causes, and releases its contents without mitigation.

The TS requirements for explosive gas mixtures are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The explosive gas mixture TS are not associated with a process variable, design feature, or operating restriction that is monitored or controlled during power operation and is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS

does not meet criterion 2. This conclusion is consistent with the corresponding evaluation in Ref. 4.

The TS for explosive gas mixture does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

The explosive gas mixture of the waste gas holdup tanks has not been shown to be a significant risk contributor to public health and safety by either operational experience or PSA. Risk of radioactivity release is dominated by severe accidents, not releases of radionuclides generated from normal operations. In addition, from Ref. 3 the quantity of radioactivity contained in each pressurized gas storage tank in the waste gas holdup system is limited to assure a release would be substantially below the dose guideline values of 10 CFR 100. The waste gas holdup tanks are not modeled in the DCPP IPE. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

__ This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

(The TS may be relocated but a program statement will be added to Administrative Controls section).

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.11.2.6 GAS STORAGE TANKS

Applicable Modes: At all times

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
__	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
__	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(3)	A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to 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.
__	X	(4)	An SSC which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the (TS) shall be retained in the TS.

If the answer to all four of the above questions is "NO", the TS may be relocated to a controlled document.

(3) DISCUSSION

The gas storage tank specifications impose limits on the quantity of radioactive material contained in those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another TS. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a member of the public at the nearest site boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3 and Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure." The safety analysis concerning the gaseous radwaste system assumes a rupture of a storage tank without mitigation.

The TS requirements for gas storage tanks are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The gas storage tank TS are associated with a process variable, design feature, or operating restriction that is monitored or controlled during power operation and is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, the barrier in this case is the tank itself which is

not a barrier that is monitored and controlled during power operation of the plant. Therefore, this TS does not satisfy criterion 2. This conclusion is consistent with the corresponding evaluation in Ref. 4.

The TS for gas storage tanks does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

From Ref. 3, the waste gas holdup tanks, which hold radwaste, have not been shown to be a significant risk contributor to public health and safety by either operational experience or PSA. In addition, from Ref. 3 the quantity of radioactivity contained in each pressurized gas storage tank in the waste gas holdup system is limited to assure a release would be substantially below the dose guideline values of 10 CFR 100. The waste gas holdup tanks are not modeled in the DCPP IPE. Therefore, this TS does not satisfy criterion 4.

(4) CONCLUSION

— This Technical Specification is retained.

X The Technical Specification may be relocated to a licensee controlled document.

(The TS may be relocated but a program statement will be added to Administrative Controls section).

REFERENCES:

1. DCPP Units 1 and 2 Technical Specifications and Bases (NUREG-1151) as amended.

2. Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Rev. 1 dated April 1995

3. J. D. Andrachek, et. al., Methodically Engineered, Restructured, and Improved Technical Specifications, MERITS Program - Phase II Task 5, Criteria Application, WCAP-11618, November 1987.

4. NRC letter to Westinghouse Owners Group (T. Murley to R. Newton), "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," May 9, 1988.

5. 10CFR50.36, "Technical Specifications," dated July 19, 1995 (Federal Register Vol. 60, No. 138, Page 36959).

6. Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, NRC, Federal Register, Page 39132, Vol. 58, No. 138.

7. PG&E letter, DCL-92-087, "Response to Generic Letter 88-20, Individual Plant Examination," dated April 14, 1992.