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December 3, 2020

PG&E Letter DCL-20-092

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Washington, D.C. 20555-0001

10 CFR 50.90

Diablo Canyon Units 1 and 2
Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
License Amendment Request 20-03
Proposed Technical Specifications and Revised License Conditions for the Permanently Defueled Condition

References:

1. PG&E Letter DCL-18-096, Certification of Permanent Cessation of Power Operations, dated November 27, 2018 (ML18331A553)

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) hereby requests approval of the enclosed proposed amendment, to Facility Operating License(s) (FOL) Numbers DPR-80 and DPR-82. The proposed amendment would revise the Operating Licenses, including Appendix D, Additional Conditions, and Appendix A, Technical Specifications (TS), to reflect the permanent cessation of reactor operation, for Diablo Canyon Power Plant (DCPP) Units 1 and 2.

In Reference 1, PG&E notified the U.S. Nuclear Regulatory Commission (NRC) that the decision was made to permanently cease operations at DCPP Units 1 and 2 upon expiration of the operating licenses. The FOL for DCPP Unit 1 expires on November 2, 2024, and the FOL for DCPP Unit 2 expires on August 26, 2025. Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessels for Units 1 and 2, in accordance with 10 CFR 50.82(a)(1)(i) and (ii), the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement of fuel in the reactor vessels in accordance with 10 CFR 50.82(a)(2). The proposed changes revise the FOL, Additional Conditions, and TS including Section 5.0, Administrative Controls, to align with the requirements for a permanently shutdown and defueled condition.

In accordance with site administrative procedures and the Quality Assurance Program, the proposed amendment has been reviewed by the Plant Staff Review Committee.

Pursuant to 10 CFR 50.91, PG&E is notifying the State of California of this license amendment request by transmitting a copy of this letter and enclosure to the California Department of Public Health.

The Enclosure provides a detailed description and evaluation of the proposed changes. Attachments 1 through 4 contain the markups of the current FOLs, Additional Conditions, TS, and TS Bases pages. Attachments 5 through 7 contain the retyped FOLs, TS, and TS Bases pages.

PG&E requests a 180-day implementation period from the effective date of the amendment. PG&E requests that the approved amendments become effective after the following conditions have been met:

- docketing of the certifications required by 10 CFR 50.82(a)(1)(i and ii) for DCPP Units 1 and 2,
- DCPP Units 1 and 2 have both been shutdown for at least 45 days,
- and a Certified Fuel Handler Training and Retraining Program has been implemented in accordance with 10 CFR 50.2.

There are no new or revised regulatory commitments (as defined by NEI 99-04) in this submittal.

If you have any questions or require additional information, please contact Mr. Philippe Soenen at 805-459-3701.

I state under penalty of perjury that the foregoing is true and correct.

Executed on December 3, 2020.

Sincerely,

Maureen R. Zawalick
Vice President, Generation Business and Technical Services

Enclosure
cc: Diablo Distribution
cc/enc: Samson S. Lee, NRR Senior Project Manager
Scott A. Morris, NRC Region IV Administrator
Donald R. Krause, NRC Senior Resident Inspector
Gonzalo L. Perez, Branch Chief, California Dept of Public Health

Evaluation of the Proposed Changes

Subject: Proposed Changes to Licenses and Technical Specifications to Reflect a Permanently Defueled Condition

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ATTACHMENTS:

1. Proposed Facility Operating License Changes (DPR-80 and DPR-82) – Markup
2. Proposed Changes to Additional Conditions – Markup
3. Proposed Technical Specification Changes – Markup
4. Proposed Technical Specification Bases Changes – Markup
5. Proposed Facility Operating License Changes (DPR-80 and DPR-82) – Clean
6. Proposed Technical Specifications – Clean
7. Proposed Technical Specification Bases – Clean

EVALUATION

1.0 SUMMARY DESCRIPTION

On November 27, 2018, Pacific Gas and Electric Company (PG&E) notified the U.S. Nuclear Regulatory Commission (NRC) that it would permanently cease power operations at Diablo Canyon Power Plant (DCPP) Units 1 and 2 upon expiration of the Facility Operating License(s) (FOL) (Reference 1). The FOL for DCPP Unit 1 expires on November 2, 2024, and the FOL for DCPP Unit 2 expires on August 26, 2025.

Pursuant to 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit,” PG&E proposes an amendment to FOL DPR-80 and DPR-82, including Appendix D, Additional Conditions, and Appendix A, Technical Specification(s) (TS). The proposed amendment would revise the FOL, Additional Conditions, and the associated TS to reflect the permanent cessation of reactor operation and permanent defueling of the reactors for DCPP Units 1 and 2.

The proposed changes to the FOL and TS are in accordance with 10 CFR 50.36(c)(1) through (c)(5). The proposed changes also include administrative changes to content format and revised page numbering. The TS Table of Contents is revised accordingly.

The proposed changes will not be applicable until permanent cessation of operations and once the following conditions have been met for DCPP Units 1 and 2:

- docketing of the certifications required by 10 CFR 50.82(a)(1)(i and ii) for DCPP Units 1 and 2,
- DCPP Units 1 and 2 have both been shut down for at least 45 days, and
- a Certified Fuel Handler Training and Retraining Program has been implemented in accordance with 10 CFR 50.2.

Upon docketing of the certifications required by 10 CFR 50.82(a)(1)(i and ii), the 10 CFR Part 50 licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors or placement of fuel in the reactor vessels in accordance with 10 CFR 50.82(a)(2).

The current DCPP TS contain Limiting Condition(s) for Operation (LCO) that provide for appropriate functional capability of equipment required for safe operation of the facility, including safe storage and management of irradiated fuel. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing TS provide an appropriate level of control. However, the majority of the existing TS are only applicable with the reactor in an operational mode. LCO and associated Surveillance Requirements (SR) that will not apply in the permanently defueled condition are proposed for deletion. The remaining portions of the TS are being proposed for revision and incorporation as the Permanently Defueled Technical

Specifications (PDTS) to provide a continuing acceptable level of safety which addresses the reduced scope of postulated design basis accidents (DBAs) associated with a permanently defueled plant.

This license amendment request (LAR) also proposes changes to the staffing and training requirements for the staff contained in Section 5.0, Administrative Controls, of the TS. To support activities at DCPP Units 1 and 2, once the site is in a permanently shutdown and defueled condition, some administrative controls will no longer be applicable and can be deleted or revised. Additional changes are proposed to certain required reports and programs that will no longer be applicable once DCPP Units 1 and 2 are permanently defueled.

In the development of the proposed changes, PG&E reviewed the requirements from other plants, such as Davis Besse (References 2, 3, and 4, original submittals and initial response to requests for additional information), Three Mile Island (References 5 and 6), Oyster Creek (References 7 and 8), Fort Calhoun (Reference 9), and Crystal River (Reference 10) .

This LAR provides a discussion and description of the proposed FOL, Additional Conditions, and TS changes; a technical evaluation of the proposed FOL, Additional Conditions, and TS changes; and information supporting a finding of No Significant Hazards Consideration.

Related Licensing Actions

The licensing actions listed below impact the information included in this LAR:

- On September 11, 2020, the NRC issued License Amendment Numbers 237 and 239 to FOL DPR-80 and DPR-82, respectively (Reference 11). The implementation of these License Amendments are reflected in this LAR, but have not yet been implemented at the facility.

- On August 31, 2020, PG&E submitted LAR 20-02 titled, "Non-Voluntary License Amendment Request to Revise Technical Specifications 3.2.1, $F_Q(Z)$, to Implement Methodology from WCAP-17661, Revision 1, 'Improved RAOC and CAOC F_Q Surveillance Technical Specifications'" (Reference 46). The proposed LAR 20-02 impacts TS 3.2.1 and 5.6.5b. As discussed below in this LAR, TS 3.1 through 3.9 that are deleted in their entirety are identified as such below and shown in the table of contents, but the associated deleted pages are not included in Attachment 3. In addition, the changes proposed in LAR 20-02, do not impact the proposed deletion in this LAR. The markups to TS 5.6.5b proposed in this LAR assume approval of Reference 46 by the NRC. If Reference 46, as provided, is not approved by the NRC, PG&E will supplement this LAR with the necessary revisions.

2.0 DETAILED DESCRIPTION AND BASIS FOR CHANGES

The proposed amendment would revise the DCPP Units 1 and 2 FOL, Additional Conditions, and transform the operating TS into the DCPP Units 1 and 2 PDTS to align with a permanently defueled condition. To support the proposed changes, PG&E has evaluated the DBAs that will be applicable in a permanently shutdown and defueled condition. PG&E has also evaluated the General Design Criteria (GDC) with respect to compliance in the permanently shutdown and defueled condition. The DBAs and GDC evaluations provide the framework and basis for the proposed changes.

2.1 General Analysis Applicable to Proposed Changes

Chapter 15 of the DCPP Units 1 and 2 Updated Final Safety Analysis Report (UFSAR) describes the DBAs and transient scenarios applicable during power operations. During normal power operations, the forced flow of water through the reactor coolant system (RCS) removes the heat generated by the reactor. The RCS, operating at high temperatures and pressures, transfers this heat through the steam generator (SG) tubes to the secondary system. The most severe postulated accidents for nuclear power plants involve damage to the reactor core and the release of large quantities of fission products to the RCS. Many of the accident scenarios postulated in the UFSAR involve failures or malfunctions of systems which could affect the reactor core.

DCPP Units 1 and 2 will permanently cease operation and remove all nuclear fuel from the reactor vessels. In this condition, the number of credible accidents/transients is significantly smaller than for a plant authorized to operate the reactor or place fuel in the reactor vessel. During decommissioning, the spent nuclear fuel (SNF) will be stored in the spent fuel pool (SFP) and in the independent spent fuel storage installation (ISFSI). In this configuration, the SFP and its systems are dedicated to SNF storage.

With the SNF being stored in the SFP and the ISFSI, the reactor, RCS, and secondary system will no longer be in operation and have no function related to the storage of SNF. Upon permanent cessation of power operation and the permanent removal of the fuel from the reactor core, the accident/transient initial conditions/initial reactor power level of the reactor core, cannot be achieved and, as such, most of the accident/transient scenarios are not possible. Therefore, the postulated UFSAR Chapter 15 accidents/transients involving failure or malfunction of the reactor, RCS, and secondary system are no longer applicable.

Chapter 15 of the UFSAR describes the safety analysis aspects of the plant that were evaluated to demonstrate that the plant could be operated safely and that radiological consequences from postulated accidents do not exceed regulatory requirements. The full spectrum of abnormal situations and accidents are evaluated by dividing plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

(1) Condition I: Normal Operation and Operational Transients (Initial Conditions)

(2) Condition II: Faults of Moderate Frequency

(3) Condition III: Infrequent Faults

(4) Condition IV: Limiting Faults

In addition, UFSAR Chapter 11 provides an evaluation of tank ruptures associated with storage of radioactive waste. A list of the Chapter 15 DBAs and Chapter 11 tank ruptures and whether the accident applies to DCPP Units 1 and 2 in a permanently defueled condition is provided below in Table 2.1.1.

TABLE 2.1.1 - DCPP Units 1 and 2 Design Basis Accidents and Tank Ruptures

Postulated Accident or Transient	Defueled Applicability
Chapter 11	
11.3.3.9 Rupture of a Gas Decay Tank	Applicable
11.2.3.12 Rupture of a Liquid Holdup Tank	Applicable
11.2.3.12 Rupture of Volume Control Tank	Applicable
Chapter 15	
15.2 Condition II - Faults of Moderate Frequency	
15.2.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition	Not Applicable
15.2.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Not Applicable
15.2.3 Rod Cluster Control Assembly Mis-operation	Not Applicable
15.2.4 Uncontrolled Boron Dilution	Not Applicable
15.2.5 Partial Loss of Forced Reactor Coolant Flow	Not Applicable
15.2.6 Startup of an Inactive Reactor Coolant Loop	Not Applicable
15.2.7 Loss of External Electrical Load and/or Turbine Trip	Not Applicable
15.2.8 Loss of Normal Feedwater	Not Applicable
15.2.9 Loss of Offsite Power to the Station Auxiliaries	Not Applicable
15.2.10 Excessive Heat Removal Due to Feedwater System malfunctions	Not Applicable
15.2.11 Sudden Feedwater Temperature Reduction	Not Applicable
15.2.12 Excessive Load Increase Incident	Not Applicable

Postulated Accident or Transient	Defueled Applicability
15.2.13 Accidental Depressurization of the Reactor Coolant System	Not Applicable
15.2.14 Accidental Depressurization of the Main Steam System	Not Applicable
15.2.15 Spurious Operation of the Safety Injection System at Power	Not Applicable
15.3 Condition III - Infrequent Faults	
15.3.1 Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes that Actuate Emergency Core Cooling System	Not Applicable
15.3.2 Minor Secondary System Pipe Breaks	Not Applicable
15.3.3 Inadvertent Loading of a Fuel Assembly into an Improper Position	Not Applicable
15.3.4 Complete Loss of Forced Reactor Coolant Flow	Not Applicable
15.3.5 Single Rod Cluster Control Assembly Withdrawal at Full Power	Not Applicable
15.4 Condition IV - Limiting Faults	
15.4.1 Major Reactor Coolant System Pipe Ruptures (LOCA)	Not Applicable
15.4.2 Major Secondary System Pipe Ruptures	Not Applicable
15.4.3 Steam Generator Tube Rupture (SGTR)	Not Applicable
15.4.4 Single Reactor Coolant Pump Locked Rotor	Not Applicable
15.4.5 Fuel Handling Accident	Applicable
15.4.6 Rupture of a Control Rod Drive Mechanism Housing	Not Applicable

The UFSAR Chapter 15 DBAs and Chapter 11 tank ruptures that remain applicable to DCPP Units 1 and 2 in a permanently shut down and defueled condition are a fuel handling accident (FHA) in the fuel handling building (FHB), and the rupture of a gas decay tank (GDT), liquid holdup tank (LHUT), or the volume control tank (VCT).

In accordance with 10 CFR 50.2, "Definitions," safety-related structures, systems, and components (SSCs) are those relied upon to remain functional during and following design basis events to assure:

1. the integrity of the reactor coolant pressure boundary (RCPB);
2. the capability to shut down the reactor and maintain it in a safe shutdown condition; or,
3. the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.43(a)(1) or 10 CFR 100.11.

The first two criteria (integrity of the RCPB and safe shutdown of the reactor) are not applicable to a plant in a permanently defueled condition. The third criterion is related to preventing or mitigating the consequences of accidents that could result in potential offsite exposures exceeding limits. However, after termination of reactor operations at DCPP Units 1 and 2, permanent removal of fuel from the reactor vessel, and following 45 days of decay time after shutdown (as discussed below), there are no active SSCs that meet the definition of a safety-related SSC stated in 10 CFR 50.2.

10 CFR 50.36, "Technical Specifications," provides the regulatory requirements related to the content of TS. As detailed in Section 3.1 of this proposed amendment, this regulation lists four criteria to define the scope of equipment and parameters that must be included in TS. In a permanently defueled condition, the scope of equipment and parameters that must be included in the DCPP PDTS is limited to those needed to address the remaining applicable accidents, so that the consequences of the accident are maintained within acceptable limits. The applicable accidents are as follows:

Tank Ruptures

As discussed in UFSAR Section 11.3.3.9, "Rupture of a Gas Decay Tank," the quantity of radioactive material contained in each GDT is limited to less than or equal to 10^5 curies noble gases (considered as Xe-133 equivalent), to ensure 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 be less than 0.5 rem.

UFSAR Section 11.2.3.12, discusses the rupture of an LHUT and a VCT and concludes the following:

- The dose consequences at both the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) due to airborne releases following a LHUT rupture are estimated to be below the acceptance criteria of 0.5 rem whole body and 3 rem thyroid.

- The dose consequences at both the EAB and the LPZ due to airborne releases following a VCT rupture are estimated to be below the acceptance criteria of 0.5 rem whole body and 3 rem thyroid.

These analyses do not credit any active components for mitigating the consequences of the rupture. These analyses remain valid and bounding during decommissioning and therefore, will not be further addressed in this submittal.

Fuel Handling Accident Analysis for Permanently Defueled Condition

The FHA in the FHB is the remaining Chapter 15 accident with a radiological consequence to the health and safety of the public and control room (CR). A calculation (Reference 12) was performed to assess the dose consequences of a postulated FHA after permanent cessation of operations. The calculation demonstrates that radiological dose consequences at the EAB and in the CR are within allowable limits of 10 CFR 50.67 and Regulatory Guide (RG) 1.183, Revision 0 (Reference 13) without crediting the control room ventilation system (CRVS) (e.g., filtration, flowrates, intake location if beneficial), CR structure (e.g., shielding, reduced source volume), or the FHB ventilation system (which exhausts through the plant vent) after a 30-day fuel decay period following permanent reactor shutdown. With the exception of not taking credit for protection provided by the building ventilation/structure at the receptor location, use of the worst case location for the release point, and use of a reduced depth of water credited above the damaged assembly, the FHA dose consequence analysis as well as the supporting atmospheric dispersion calculations are consistent with the assumptions and methodology outlined in PG&E's Alternative Source Term LAR 15-03 (Reference 14), including supplements and accepted by the NRC via License Amendment Numbers 230 and 232 (Reference 15). The above changes in the assumptions utilized to assess the dose consequences of the FHA are intended to support a longer-term decommissioning initiative to minimize the number of systems and structures required to support the plant following permanent cessation of operations.

The accident is assumed to occur during handling of a spent fuel assembly in the SFP. One fuel assembly is damaged, releasing all of the fuel gap activity associated with that assembly. A radial peaking factor of 1.65 is applied to the activity release. The activity (consisting of noble gases, halogens, and alkali metals) is released in a "puff" to the SFP, which has a minimum of 21 feet of water above the damaged fuel assembly. A halogen decontamination factor of 142 is applied to the gap release (for all halogen species) to account for the scrubbing effect of the 21 feet of water above the damaged fuel assembly. The listed decontamination factor is developed using guidance provided in Reference B-1 of RG 1.183 (Reference 13). Noble gas and un-scrubbed iodines rise to the water surface where they are mixed in the minimum available air space in the FHB above the SFP. Per Appendix B of RG 1.183, Revision 0 (Reference 13), all of the alkali metals are retained in the SFP water.

In accordance with RG 1.183, Revision 0 (Reference 13) and current licensing basis, the environmental release due to the accident occurs over a 2-hour period. The analysis uses bounding atmospheric dispersion factors associated with the limiting release location between the plant vent and the worst-case location in the FHB structure, for all of the radioactivity released following an FHA.

The dose consequences were calculated using the RADTRAD 3.0.3 computer software. The acceptance criteria used in the calculation are described below:

- EAB acceptance criteria is based on the U.S. Environmental Protection Agency (EPA) Protective Action Guides (PAG) Manual (Reference 16) which requires the dose at the EAB (inclusive of the contribution of ground shine) to be less than 1 rem when projected over four days;

- Acceptance criteria for on-site habitability is based on the requirements in RG 1.183, Revision 0 (Reference 13) which requires a 30-day integrated dose of less than 5 rem total effective dose equivalent (TEDE).

The acceptance criteria for the EAB is more conservative than the limits described in RG 1.183, Revision 0 (Reference 13). The results of the analysis demonstrate that after 30 days of decay the radiological doses at the EAB and the CR meet the acceptance criteria specified above and are therefore within the limits of 10 CFR 50.67, RG 1.183, Revision 0 (Reference 13), and the PAG Manual (Reference 16). The dose at the LPZ was not specifically addressed because it would be less than the dose at the EAB and therefore would be below the limits specified in 10 CFR 50.67, RG 1.183, Revision 0 (Reference 13), and the PAG Manual (Reference 16). It was determined that 45 days after reactor shutdown the resultant dose in the CR would be below the current licensing basis value of 1 rem. Therefore, PG&E will implement the revised FHA and the PDTS 45 days after both Units have been shutdown.

In conclusion, the FHA analysis for DCPP Units 1 and 2 shows that, following 45 days of decay time after reactor shutdown and provided the SFP water level requirements of TS 3.7.15[1] are met, the dose consequences for the CR, the EAB, and the LPZ remain below the acceptance criteria, without relying on active components remaining functional for accident mitigation during and following the event.

Conclusion

The remaining accident and tank ruptures that support the permanently shutdown and defueled condition do not rely on any active safety system for mitigation. The new FHA analysis, after 45 days of permanent shutdown, demonstrates that the unmitigated release will not exceed the regulatory limits for the CR, EAB, or LPZ. As discussed above, the current analysis for tank ruptures is valid and remains bounding during decommissioning. Upon permanent shutdown and cooldown, the source term contained within these tanks represents the highest (worst case) source term. Potential subsequent additions to the tanks during decommissioning resulting from water management activities would be less than the final shutdown and cooldown source term.

Detailed Review of General Design Criteria After Permanent Defueling

As stated and described in DCPP UFSAR Section 3.1, DCPP Units 1 and 2 are designed to comply with the “General Design Criteria for Nuclear Power Plant Construction Permits,” published by the Atomic Energy Commission (AEC) in July, 1967 (i.e., the 1967 GDCs). However, PG&E has made subsequent commitments to GDCs

[1] TS 3.7.15, "Spent Fuel Pool Water Level," requires the spent fuel pool water level to be greater than or equal to 23 feet over the top of irradiated fuel assemblies seated in the storage racks. TS 3.7.15 is applicable during movement of irradiated fuel assemblies in the spent fuel pool.

issued later (e.g., 1971 GDC, and the 1987 revision to GDC 4). In addition, DCPP UFSAR Appendix 3.1A briefly discusses the extent to which the original DCPP principal design features (the 1967 GDCs plus additional design features) for plant SSCs conform to the intent of the AEC "General Design Criteria for Nuclear Power Plants" published in February 1971 as 10 CFR Part 50, Appendix A (i.e., the 1971 GDCs). As part of the initial licensing of DCPP Units 1 and 2, each 10 CFR Part 50, Appendix A 1971 GDC was addressed by including a summary of how the DCPP principal design features (the 1967 GDCs plus additional design features) demonstrates conformance to the intent of or exceptions to the criterion. The discussion of how the plant design conformed to the intent of the 1971 GDCs was included in Appendix 3.1A of the original Final Safety Analysis Report, and was reviewed by the NRC to conclude that DCPP's design conformed to the intent of the 1971 GDCs. The degree to which the DCPP design conforms to the intent of the 1971 GDCs, as summarized in DCPP UFSAR Appendix 3.1A, establishes additional DCPP licensing basis which must be reviewed when evaluating facility changes.

With the termination of reactor operations at DCPP Units 1 and 2, and the permanent removal of fuel from the reactors, as certified in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and pursuant to 10 CFR 50.82(a)(2), the majority of the GDCs in the UFSAR will no longer be applicable. During decommissioning, all of the SNF stored in the SFPs will be transferred to the ISFSI until it is shipped offsite in accordance with the schedules provided in the Post-Shutdown Decommissioning Activities Report (Reference 17) and the Irradiated Fuel Management Plan (Reference 18). The RCS, steam system, and turbine generator will no longer be in operation and will have no function related to the safe storage and management of the SNF. In general, the GDCs that relate only to reactor operation or the systems that support reactor operation will no longer be applicable when the facility is in a permanently defueled condition. However, since SNF and radioactive waste will still be stored at the facility, the GDCs that relate to the storage of waste, SNF, and the prevention of radioactive release will still be applicable to the facility. This includes supporting GDCs that relate to quality standards and fire protection.

Compliance with the 1967 GDCs and commitments to GDCs issued later (UFSAR Section 3.1), and compliance with the intent of the 1971 GDCs (UFSAR Appendix 3.1A) were reviewed as applied to the permanently shutdown and defueled condition and the limitations imposed by 10 CFR 50.82(a)(2) upon docketing the certifications required by 10 CFR 50.82(a)(1).

This review determined that compliance could be expressed in four categories[2] as follows:

A. No Longer Applies – Compliance with the GDC is no longer applicable to DCPP Units 1 and 2 since the intent and scope are based on conditions that do not

[2] When applying the categories to the 1971 GDCs that are only included in Table 2.1.3, these categories are associated with complying with the intent of the GDC as described in UFSAR, Appendix 3.1A.

apply to the facility in a permanently shutdown and defueled condition. The DBAs that evaluate conditions applicable to operation of the reactor no longer apply. The accidents that are applicable to DCPP Units 1 and 2 in a permanently shutdown and defueled condition do not credit active safety systems for accident mitigation.

B. Unchanged – Compliance with the GDC continues to apply to DCPP Units 1 and 2 as described in UFSAR Section 3.1 and Appendix 3.1A. The scope and intent of the GDC is not impacted by the transition from operating status to permanently shutdown and defueled status.

C. Minor Change – Compliance with the GDC is still required for DCPP Units 1 and 2; however, the scope can be reduced based on the transition from operating status to permanently shutdown and defueled status.

D. Major Change – Compliance with the GDC is still required for DCPP Units 1 and 2; however, the intent and scope are impacted or the scope is significantly reduced by the transition from operating status to permanently shutdown and defueled status.

These criteria are discussed in further detail below, reflecting the proposed changes to the DCPP Units 1 and 2 licensing basis. A list of the GDCs as described in UFSAR Section 3.1 and Appendix 3.1A, and the applicability to DCPP Units 1 and 2 in a permanently shutdown and defueled condition, are provided in Tables 2.1.2 and 2.1.3, respectively. GDCs included in UFSAR Section 3.1 and Appendix 3.1A that are not part of the current licensing basis for DCPP Units 1 and 2 are not included below in Tables 2.1.2 and 2.1.3.

Table 2.1.2 – Evaluation of Conformance to GDCs included in UFSAR Section 3.1

UFSAR Section 3.1 GDC	Applicability to Permanently Defueled DCPP Units 1 and 2
Criterion 1, 1967 - Quality Standards (Category A)	D. Major Change
Criterion 2, 1967 - Performance Standards (Category A)	D. Major Change
Criterion 3, 1971 - Fire Protection	D. Major Change
Criterion 4, 1967 - Sharing of Systems (Category A)	D. Major Change
Criterion 5, 1967 - Records Requirements (Category A)	B. Unchanged
Criterion 6, 1967 - Reactor Core Design (Category A)	A. No Longer Applies
Criterion 10, 1971 - Reactor Design	A. No Longer Applies
Criterion 12, 1971 - Suppression of Reactor Power Oscillations	A. No Longer Applies
Criterion 11, 1971 - Reactor Inherent Protection	A. No Longer Applies
Criterion 9, 1967 - Reactor Coolant Pressure Boundary (Category A)	A. No Longer Applies
Criterion 10, 1967 - Containment (Category A)	A. No Longer Applies
Criterion 11, 1967 - Control Room (Category B)	A. No Longer Applies

UFSAR Section 3.1 GDC	Applicability to Permanently Defueled DCPP Units 1 and 2
Criterion 19, 1999 – Control Room	A. No Longer Applies
Criterion 12, 1967 - Instrumentation and Control Systems (Category B)	D. Major Change
Criterion 13, 1967 - Fission Process Monitors and Controls (Category B)	A. No Longer Applies
Criterion 14, 1967 - Core Protection Systems (Category B)	A. No Longer Applies
Criterion 15, 1967 - Engineered Safety Features Protection Systems (Category B)	A. No Longer Applies
Criterion 16, 1967 - Monitoring Reactor Coolant Pressure Boundary (Category B)	A. No Longer Applies
Criterion 17, 1967 - Monitoring Radioactivity Releases (Category B)	C. Minor Change
Criterion 18, 1967 - Monitoring Fuel and Waste Storage (Category B)	B. Unchanged
Criterion 19, 1967 - Protection Systems Reliability (Category B)	A. No Longer Applies
Criterion 20, 1967 - Protection Systems Redundancy and Independence (Category B)	A. No Longer Applies
Criterion 29, 1971 – Protection Against Anticipated Operational Occurrences	A. No Longer Applies
Criterion 21, 1967 - Single Failure Definition (Category B)	A. No Longer Applies
Criterion 22, 1967 - Separation of Protection and Control Instrumentation Systems (Category B)	A. No Longer Applies
Criterion 23, 1967 - Protection Against Multiple Disability of Protection Systems (Category B)	A. No Longer Applies
Criterion 24, 1967 - Emergency Power for Protection Systems (Category B)	A. No Longer Applies
Criterion 25, 1967 - Demonstration of Functional Operability of Protection Systems (Category B)	A. No Longer Applies
Criterion 26, 1967 - Protection Systems Fail-Safe Design (Category B)	A. No Longer Applies
Criterion 26, 1971 - Reactivity Control System Redundancy and Capability	A. No Longer Applies
Criterion 28, 1967 - Reactivity Hot Shutdown Capability (Category A)	A. No Longer Applies
Criterion 29, 1967 - Reactivity Shutdown Capability (Category A)	A. No Longer Applies
Criterion 30, 1967 - Reactivity Holddown Capability (Category B)	A. No Longer Applies
Criterion 31, 1967 - Reactivity Control Systems Malfunction (Category B)	A. No Longer Applies
Criterion 25, 1971 - Protection System Requirements for Reactivity Control Malfunctions	A. No Longer Applies
Criterion 32, 1967 - Maximum Reactivity Worth of Control Rods (Category A)	A. No Longer Applies
Criterion 28, 1971 - Reactivity Limits	A. No Longer Applies
Criterion 33, 1967 - Reactor Coolant Pressure Boundary Capability (Category A)	A. No Longer Applies

UFSAR Section 3.1 GDC	Applicability to Permanently Defueled DCPP Units 1 and 2
Criterion 34, 1967 - Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A)	A. No Longer Applies
Criterion 35, 1967 - Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Category A)	A. No Longer Applies
Criterion 36, 1967 - Reactor Coolant Pressure Boundary Surveillance (Category A)	A. No Longer Applies
Criterion 37, 1967 - Engineered Safety Features Basis for Design (Category A)	A. No Longer Applies
Criterion 38, 1967 - Reliability and Testability of Engineered Safety Features (Category A)	A. No Longer Applies
Criterion 17, 1971 - Electric Power Systems	A. No Longer Applies
Criterion 18, 1971 - Inspection and Testing of Electric Power Systems	A. No Longer Applies
Criterion 40, 1967 - Missile Protection (Category A)	A. No Longer Applies
Criterion 4, 1987 - Environmental and Dynamic Effects Design Bases	A. No Longer Applies
Criterion 41, 1967 - Engineered Safety Features Performance Capability (Category A)	A. No Longer Applies
Criterion 42, 1967 - Engineered Safety Features Components Capability (Category A)	A. No Longer Applies
Criterion 43, 1967 - Accident Aggravation Prevention (Category A)	A. No Longer Applies
Criterion 44, 1967 - Emergency Core Cooling Systems Capability (Category A)	A. No Longer Applies
Criterion 45, 1967 - Inspection of Emergency Core Cooling Systems (Category A)	A. No Longer Applies
Criterion 46, 1967 - Testing of Emergency Core Cooling System Components (Category A)	A. No Longer Applies
Criterion 47, 1967 - Testing of Emergency Core Cooling Systems (Category A)	A. No Longer Applies
Criterion 48, 1967 - Testing of Operational Sequence of Emergency Core Cooling Systems (Category A)	A. No Longer Applies
Criterion 49, 1967 - Containment Design Basis (Category A)	A. No Longer Applies
Criterion 50, 1967 - NDT Requirement for Containment Material (Category A)	A. No Longer Applies
Criterion 51, 1967 - Reactor Coolant Pressure Boundary Outside Containment (Category A)	A. No Longer Applies
Criterion 52, 1967 - Containment Heat Removal Systems (Category A)	A. No Longer Applies
Criterion 53, 1967 - Containment Isolation Valves (Category A)	A. No Longer Applies
Criterion 54, 1971 - Piping Systems Penetrating Containment	A. No Longer Applies
Criterion 55, 1971 - Reactor Coolant Pressure Boundary Penetrating Containment	A. No Longer Applies
Criterion 56, 1971 - Primary Containment Isolation	A. No Longer Applies

UFSAR Section 3.1 GDC	Applicability to Permanently Defueled DCPP Units 1 and 2
Criterion 57, 1971 - Closed System Isolation Valves	A. No Longer Applies
Criterion 54, 1967 - Containment Leakage Rate Testing (Category A)	A. No Longer Applies
Criterion 55, 1967 - Containment Periodic Leakage Rate Testing (Category A)	A. No Longer Applies
Criterion 56, 1967 - Provisions for Testing of Penetrations (Category A)	A. No Longer Applies
Criterion 57, 1967 - Provisions for Testing of Isolation Valves (Category A)	A. No Longer Applies
Criterion 58, 1967 - Inspection of Containment Pressure-Reducing Systems (Category A)	A. No Longer Applies
Criterion 59, 1967 - Testing of Containment Pressure-Reducing Systems Components (Category A)	A. No Longer Applies
Criterion 60, 1967 - Testing of Containment Spray Systems (Category A)	A. No Longer Applies
Criterion 61, 1967 - Testing of Operational Sequence of Containment Pressure-Reducing Systems (Category A)	A. No Longer Applies
Criterion 62, 1967 - Inspection of Air Cleanup Systems (Category A)	A. No Longer Applies
Criterion 63, 1967- Testing of Air Cleanup Systems Components (Category A)	A. No Longer Applies
Criterion 64, 1967 - Testing Air Cleanup Systems (Category A)	A. No Longer Applies
Criterion 65, 1967 - Testing of Operational Sequence of Air Cleanup Systems (Category A)	A. No Longer Applies
Criterion 66, 1967 - Prevention of Fuel Storage Criticality (Category B)	C. Minor Change
Criterion 67, 1967 - Fuel and Waste Storage Decay Heat (Category B)	B. Unchanged
Criterion 68, 1967 - Fuel and Waste Storage Radiation Shielding (Category B)	B. Unchanged
Criterion 69, 1967 - Protection Against Radioactivity Release from Spent Fuel and Waste Storage (Category B)	B. Unchanged
Criterion 70, 1967 - Control of Releases of Radioactivity to the Environment (Category B)	C. Minor Change

Table 2.1.3 – Evaluation of Conformance to the intent of the GDCs included in UFSAR Appendix 3.1A

UFSAR Appendix 3.1A GDCs	Applicability to Permanently Defueled DCPP Units 1 and 2
Criterion 1, 1971 - Quality Standards and Records	D. Major Change
Criterion 2, 1971 - Design Basis for Protection Against Natural Phenomena	D. Major Change

UFSAR Appendix 3.1A GDCs	Applicability to Permanently Defueled DCPP Units 1 and 2
Criterion 3, 1971 - Fire Protection	See Table 2.1.2
Criterion 4, 1971 - Environmental and Missile Design Bases	A. No Longer Applies
Criterion 5, 1971 - Sharing of Structures, Systems and Components	D. Major Change
Criterion 10, 1971 - Reactor Design	See Table 2.1.2
Criterion 11, 1971 - Reactor Inherent Protection	See Table 2.1.2
Criterion 12, 1971 - Suppression of Reactor Power Oscillations	See Table 2.1.2
Criterion 13, 1971 - Instrumentation and Control	D. Major Change
Criterion 14, 1971 - Reactor Coolant Pressure Boundary	A. No Longer Applies
Criterion 15, 1971 - Reactor Coolant System Design	A. No Longer Applies
Criterion 16, 1971 - Containment Design	A. No Longer Applies
Criterion 17, 1971 - Electric Power Systems	See Table 2.1.2
Criterion 18, 1971 - Inspection and Testing of Electric Power Systems	See Table 2.1.2
Criterion 19, 1971 - Control Room	A. No Longer Applies
Criterion 20, 1971 - Protection System Functions	A. No Longer Applies
Criterion 21, 1971 - Protection System Reliability and Testability	A. No Longer Applies
Criterion 22, 1971 - Protection System Independence	A. No Longer Applies
Criterion 23, 1971 - Protective System Failure Modes	A. No Longer Applies
Criterion 24, 1971 - Separation of Protection and Control Systems	A. No Longer Applies
Criterion 25, 1971 - Protection System Requirements for Reactivity Control Malfunctions	See Table 2.1.2
Criterion 26, 1971 - Reactivity Control System Redundancy and Capability	See Table 2.1.2
Criterion 27, 1971 - Combined Reactivity Control Systems Capability	A. No Longer Applies
Criterion 28, 1971 - Reactivity Limits	See Table 2.1.2
Criterion 29, 1971 - Protection Against Anticipated Operational Occurrences	See Table 2.1.2
Criterion 30, 1971 - Quality of Reactor Coolant Pressure Boundary	A. No Longer Applies
Criterion 31, 1971 - Fracture Prevention of Reactor Coolant Pressure Boundary	A. No Longer Applies
Criterion 32, 1971 - Inspection of Reactor Coolant Pressure Boundary	A. No Longer Applies
Criterion 33, 1971 - Reactor Coolant Makeup	A. No Longer Applies
Criterion 34, 1971 - Residual Heat Removal	A. No Longer Applies
Criterion 35, 1971 - Emergency Core Cooling	A. No Longer Applies
Criterion 36, 1971 - Inspection of Emergency Core Cooling System	A. No Longer Applies
Criterion 37, 1971 - Testing of Emergency Core Cooling System	A. No Longer Applies
Criterion 38, 1971 - Containment Heat Removal	A. No Longer Applies

UFSAR Appendix 3.1A GDCs	Applicability to Permanently Defueled DCPP Units 1 and 2
Criterion 39, 1971 - Inspection of Containment Heat Removal System	A. No Longer Applies
Criterion 40, 1971 - Testing of Containment Heat Removal System	A. No Longer Applies
Criterion 41, 1971 - Containment Atmosphere Cleanup	A. No Longer Applies
Criterion 42, 1971 - Inspection of Containment Atmosphere Cleanup Systems	A. No Longer Applies
Criterion 43, 1971 - Testing of Containment Atmosphere Cleanup Systems	A. No Longer Applies
Criterion 44, 1971 - Cooling Water	A. No Longer Applies
Criterion 45, 1971 - Inspection of Cooling Water System	A. No Longer Applies
Criterion 46, 1971 - Testing of Cooling Water System	A. No Longer Applies
Criterion 50, 1971 - Containment Design Basis	A. No Longer Applies
Criterion 51, 1971 - Fracture Prevention of Containment Pressure Boundary	A. No Longer Applies
Criterion 52, 1971 - Capability for Containment Leakage Rate Testing	A. No Longer Applies
Criterion 53, 1971 - Provisions for Containment Testing and Inspection	A. No Longer Applies
Criterion 54, 1971 - Piping Systems Penetrating Containment	See Table 2.1.2
Criterion 55, 1971 - Reactor Coolant Pressure Boundary Penetrating Containment	See Table 2.1.2
Criterion 56, 1971 - Primary Containment Isolation	See Table 2.1.2
Criterion 57, 1971 - Closed System Isolation Valves	See Table 2.1.2
Criterion 60, 1971 - Control of Releases of Radioactive Materials to the Environment	C. Minor Change
Criterion 61, 1971 - Fuel Storage and Handling and Radioactivity Control	B. Unchanged
Criterion 62, 1971 - Prevention of Criticality in Fuel Storage and Handling	B. Unchanged
Criterion 63, 1971 - Monitoring Fuel and Waste Storage	B. Unchanged
Criterion 64, 1971 - Monitoring Radioactivity Releases	C. Minor Change

GDCs described above that are categorized as a major change are discussed further below.

Quality Standards

<u>GDC 1, 1967 – Quality Standards (Category A)</u>

Those systems and components of reactor facilities that are essential to the prevention of accidents which could affect the public health and safety, or mitigation of their consequences, shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to ensure a quality product in keeping with the safety functions, they shall be supplemented or modified as necessary. Quality assurance (QA) programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, QA programs, test procedures, and inspection acceptance levels used is required.

<u>GDC 1, 1971 – Quality Standards and Records</u>

SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A QA program shall be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety function. Appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Discussion:

The scope of systems and components that are essential to the prevention of accidents and/or classified as important to safety will be significantly reduced during decommissioning. These GDCs will still be applicable to systems and components in the permanently defueled condition. The specific requirements and scope of components will be defined as part of the Decommissioning QA Plan.

Based on the information above the intent of these GDCs are not impacted by the transition from operating to permanently shutdown and defueled. However, the scope of systems and components to which they apply will be significantly reduced. Therefore, these GDCs are categorized as a major change.

Performance Standards

<u>GDC 2, 1967 – Performance Standards (Category A)</u>

Those systems and components of reactor facilities that are essential to the prevention of accidents which could affect the public health and safety, or to mitigation of their consequences, shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects.

The design bases so established shall reflect (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area, and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

<u>GDC 2, 1971- Design Basis for Protection Against Natural Phenomena</u>

SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect:

(1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated;

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and

(3) the importance of the safety functions to be performed.

Discussion:

The scope of systems and components that are categorized as important to safety and/or essential to the prevention and mitigation of accidents will be significantly reduced during decommissioning. These GDCs will still be applicable to systems and components in the permanently defueled condition.

Based on the information above the intent of these GDCs is not impacted by the transition from operating to permanently shutdown and defueled. However, the scope of systems and components to which they apply will be significantly reduced. Therefore, these GDCs are categorized as a major change.

Fire Protection

<u>Criterion 3, 1971 – Fire Protection</u>

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials are used wherever practical throughout the unit, particularly in locations such as the containment, and CR. Fire-detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Discussion:

RG 1.191, "Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown," describes the changes to the fire protection program related to the operating unit as required by GDC 3 transitioning to a permanently shutdown condition.

> The primary objectives of the fire protection program for operating reactors are to minimize fire damage to structures, systems, and components (SSCs) important to safety; to ensure the capability to safely shut down the reactor; and to maintain it in a safe shutdown condition. For an initial period following shutdown, accidents that can challenge the 10 CFR Part 100 limits remain credible. The fire protection program should continue to provide protection against these events. The primary fire protection concern for permanently shutdown plants is protecting the integrity of the spent fuel and preventing or minimizing the release of radioactive materials resulting from fires involving contaminated plant SSCs or radioactive wastes. The radiation dose limits specified in 10 CFR Part 20, "Standards for Protection Against Radiation," apply to plant personnel and members of the public for fire incidents at permanently shutdown nuclear power plants. Licensees should make every effort to maintain exposures to radiation resulting from a fire as low as reasonably achievable.

The fire protection program for a decommissioned unit is governed by the requirements of 10 CFR 50.48(f).

Based on the information above both the scope and intent of GDC 3, 1971 is impacted by the transition from operating to permanently shutdown and defueled. Therefore, this GDC is categorized as a major change.

Sharing of Systems

<u>GDC 4, 1967 – Sharing of Systems (Category A)</u>

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

Discussion:

DCPP Units 1 and 2 share multiple systems, components and facilities between units. In the permanently shutdown and defueled condition, GDC 4, 1967 will still be applicable to certain shared components and facilities.

Based on the information above the intent of this GDC is not impacted by the transition from operating to permanently shutdown and defueled. However, the scope of systems and components to which it applies will be significantly reduced. Therefore, this GDC is categorized as a major change.

GDC 5, 1971 – Sharing of Structures, Systems and Components

SSCs important to safety shall not be shared between nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Discussion:

The scope of systems and components classified as important to safety will be significantly reduced during decommissioning. However, DCPP Units 1 and 2 will still share certain SSCs. The portion of the GDC associated with an orderly shutdown and cooldown will no longer be applicable.

Based on the information above the intent of the GDC is impacted and the scope of systems and components to which it applies will be significantly reduced. Therefore, this GDC is categorized as a major change.

Instrumentation and Control

GDC 12, 1967 – Instrumentation and Control Systems (Category B)

Instrumentation and controls (I&C) shall be provided as required to monitor and maintain variables within prescribed operating ranges.

GDC 13, 1971 – Instrumentation and Control

Instrumentation and Control - Instrumentation shall be provided to monitor variables and systems over their anticipated range for normal operation, for anticipated operational occurrences (AOOs), and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the RCPB, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Discussion:

In the permanently shut down and defueled condition, there will be no I&C systems that are required to mitigate an accident. In addition, there are no operability requirements for I&C systems in the proposed PDTS.

In accordance with the DCPP UFSAR, GDC 12, 1967 will still be applicable to the storage and handling of spent fuel. Instrumentation is provided to give an alarm in the CR when the water level in the SFP reaches either the high- or low-level alarm setpoint. In addition, local instrumentation is provided to measure the temperature of the water in the SFP and provide local indication as well as annunciation in the CR when normal temperatures or rates of temperature change are exceeded.

Instrumentation is also provided for the SFP cooling and cleanup system. Local instrumentation is provided for the following:

- indication of the temperature of the SFP water as it leaves the SFP heat exchanger;
- measuring and providing indication of pressures across the SFP pumps;
- measuring pressure differential on the SFP demineralizer and SFP resin trap filter; and
- measuring and providing indication of flows in the outlet line of the SFP filter and the inlet line to the SFP demineralizer.

Based on the information above the intent of GDC 12, 1967 is not impacted by the transition from operating to permanently shutdown and defueled. However, the scope of systems and components to which it applies will be significantly reduced. Therefore, this GDC is categorized as a major change.

Regarding compliance with the intent of GDC 13, 1971, the scope of the GDC associated with assuring adequate safety related to variables and systems that can affect the fission process, the integrity of the reactor core, the RCPB, and the containment and its associated systems, will no longer be applicable. Therefore, the intent of this GDC is impacted and as described above the scope of systems and components to which it applies will be significantly reduced. Therefore, this GDC is also categorized as a major change.

Detailed Discussion of Proposed Changes to the FOL, Additional Conditions, and TS

The following tables identify each License Condition, Additional Condition (Appendix D), and TS (Appendix A) for DCPP Units 1 and 2 that is being changed, the proposed change, and the basis for each change. Proposed revisions to the current conditions and TS are shown with additions in *underlined italics* and deletions using ~~strikethrough~~. Upon approval of this amendment, changes to the TS Bases will be incorporated in accordance with TS 5.5.14, “Technical Specification Bases Control Program,” which is retained in its entirety without change.

Attachments 1 through 3 provide the marked-up version of the DCPP Units 1 and 2 FOL, Additional Conditions, and TS. TS 3.1 through 3.9 that are deleted in their entirety are identified as such below and shown in the table of contents, but the associated deleted pages are not included in Attachment 3. The TS Table of Contents is revised to reflect the remaining applicable sections and new page numbering. Attachment 4 provides a markup of the TS Bases, with Bases that are completely deleted only shown in the table of contents. Attachments 5 through 7 provide the clean versions of Attachments 1, 3, and 4. In addition, the following administrative changes are included in the Attachments as appropriate:

- reformatting (margins, font, tabs, line spacing, etc.) content to create a continuous electronic file;
- renumbering of pages, where appropriate, to condense and reduce the number of pages; and
- historical amendment numbers in the footer are deleted.

10 CFR 50.36, "Technical specifications," promulgates the regulatory requirements related to the content of TS. As detailed in Section 3.1 of this proposed amendment, this regulation lists four criteria to define the scope of equipment and parameters that must be included in TS. In a permanently defueled condition, the scope of equipment and parameters that must be included in the DCPP Unit 1 and 2 TS is limited to those needed to address the remaining applicable accidents so that the consequences of the accidents are maintained within acceptable limits.

2.2 Facility Operating Licenses, Additional Conditions, and Technical Specifications

The proposed changes described below are applicable to DCPP Units 1 and 2. Based on PG&Es current plans, DCPP Unit 1 will shutdown prior to DCPP Unit 2; however, the changes described below to DCPP Units 1 and 2 FOL, including Appendix D (Additional Conditions), and Appendix A (TS), will become effective at the same time, after the following conditions have been met:

- certifications required by 10 CFR 50.82(a)(1)(i and ii) have been submitted for both Units,
- both Units have been shutdown for at least 45 days, and
- a Certified Fuel Handler Training and Retraining Program has been implemented in accordance with 10 CFR 50.2.

2.2.1 DCPP Unit 1 Facility Operating License - Proposed Changes

License Finding 1.B	
Current License Finding 1.B. Construction of the Diablo Canyon Nuclear Power Plant, Unit 1 (the facility), has been substantially completed in	Proposed License Finding 1.B *Deleted per Amendment No. ###.*

conformity with Provisional Construction Permit No. CPPR-39 and the application, as amended, the provisions of the Act, and the regulations of the Commission;	
Basis	
This license finding is proposed for deletion in its entirety. Decommissioning of DCPP Unit 1 is not dependent on the regulations that governed the construction of the facility.	

License Finding 1.C	
Current License Finding 1.C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission, except as exempted from compliance in Section 2.D below;	Proposed License Finding 1.C The facility will ~~operate~~*be maintained* in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission~~, except as exempted from compliance in Section 2.D below~~;
Basis	
In accordance with 10 CFR 50.82(a)(2), once PG&E has permanently ceased operation of DCPP Unit 1 and certified that fuel has been permanently removed from the reactor, the license will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel. The removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with a decommissioning plant. Reference to the exemption in Section 2.D is proposed for deletion because Section 2.D is proposed for deletion in its entirety as described below.	

License Finding 1.D	
Current License Finding 1.D. There is reasonable assurance (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter I, except as exempted from compliance in Section 2.D below;	Proposed License Finding 1.D There is reasonable assurance (i) that the activities authorized by this~~ operating~~ license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter I~~, except as exempted from compliance in Section 2.D below~~;

Basis
In accordance with 10 CFR 50.82(a)(2), once PG&E has permanently ceased operation of DCPP Unit 1 and certified that fuel has been permanently removed, the license will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel. The removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with a decommissioning plant. Reference to the exemption in Section 2.D is proposed for deletion because Section 2.D is proposed for deletion in its entirety as described below.

License Finding 1.E	
<u>Current License Finding 1.E</u> The Pacific Gas and Electric Company is technically qualified to engage in the activities authorized by this operating license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;	<u>Proposed License Finding 1.E</u> The Pacific Gas and Electric Company is technically qualified to engage in the activities authorized by this ~~operating~~ license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;
Basis	
Once PG&E has permanently ceased operation of DCPP Unit 1 and certified that fuel has been permanently removed, the license will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel. The removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with a decommissioning plant.	

License Finding 1.H	
<u>Current License Finding 1.H</u> After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. DPR-80, subject to the conditions for protection of the environment set forth herein, is in accordance with applicable Commission regulations governing environmental reviews (10 CFR Part 50, Appendix D and 10 CFR Part 51) and all applicable requirements have been satisfied; and	<u>Proposed License Finding 1.H</u> After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility ~~Operating~~ License No. DPR-80, subject to the conditions for protection of the environment set forth herein, is in accordance with applicable Commission regulations governing environmental reviews (10 CFR Part 50, Appendix D and 10 CFR Part 51) and all applicable requirements have been satisfied; and

Basis
Once PG&E has permanently ceased operation of DCPP Unit 1 and certified that fuel has been permanently removed, the license will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel. The removal of the term operating provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with a decommissioning plant.

License Condition 2	
<u>Current License Condition 2</u> Pursuant to Commission's Memorandum and Order CLI-84-13, dated August 10, 1984, Facility Operating License No. DPR-76 issued September 22, 1981, as subsequently amended, is superseded by Facility Operating License No. DPR-80, hereby issued to Pacific Gas and Electric Company to read as follows:	<u>Proposed License Condition 2</u> Pursuant to Commission's Memorandum and Order CLI-84-13, dated August 10, 1984, Facility Operating License No. DPR-76 issued September 22, 1981, as subsequently amended, is superseded by Facility ~~Operating~~ License No. DPR-80, hereby issued to Pacific Gas and Electric Company to read as follows:
Basis	
Once PG&E has permanently ceased operation of DCPP Unit 1 and certified that fuel has been permanently removed, the license will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel. The removal of the term operating provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with a decommissioning plant.	

License Condition 2.A	
<u>Current License Condition 2.A</u> This License applies to the Diablo Canyon Nuclear Power Plant, Unit 1, a pressurized water nuclear reactor and associated equipment (the facility), owned by the Pacific Gas and Electric Company (PG&E). The facility is located in San Luis Obispo County, California, and is described in PG&E's Final Safety Analysis Report as supplemented and amended, and the Environmental Report as supplemented and amended.	<u>Proposed License Condition 2.A</u> This License applies to the Diablo Canyon Nuclear Power Plant, Unit 1, a pressurized water nuclear reactor and associated equipment (the facility), owned by the Pacific Gas and Electric Company (PG&E). The facility is located in San Luis Obispo County, California, and is described in PG&E's ~~Final~~<u>*Defueled*</u> Safety Analysis Report as supplemented and amended, and the Environmental Report as supplemented and amended.
Basis	
License Condition 2.A is revised to reflect the conversion of the "Final Safety Analysis Report" to the "Defueled Safety Analysis Report" upon implementation of this LAR. The proposed terminology is consistent with a decommissioning plant.	

License Condition 2.B.(1)

Current License Condition 2.B.(1)	Proposed License Condition 2.B.(1)
Pursuant to Section 104(b) of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess, use, and operate the facility at the designated location in San Luis Obispo County, California, in accordance with the procedures and limitations set forth in this license;	Pursuant to Section 104(b) of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess~~,~~ *and* use~~, and operate~~ the facility at the designated location in San Luis Obispo County, California, in accordance with the procedures and limitations set forth in this license;

Basis

In accordance with 10 CFR 50.82(a)(2), once PG&E has permanently ceased operation of DCPP Unit 1 and certified that fuel has been permanently removed, the license will no longer authorize operation of the reactor. As such, the facility would remain authorized to possess the existing spent fuel and use the systems required to support safe fuel storage during the decommissioning period, in accordance with the specified limitations for storage.

License Condition 2.B.(2)

Current License Condition 2.B.(2)	Proposed License Condition 2.B.(2)
Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;	Pursuant to the Act and 10 CFR Part 70, to ~~receive,~~ possess~~, and use~~ at any time special nuclear material *that was used* as reactor fuel, in accordance with the limitations for storage ~~and amounts required for reactor operation~~, as described in the ~~Final~~ *Defueled* Safety Analysis Report, as supplemented and amended;

Basis

The proposed change to this license condition removes the authorization for receipt and use of special nuclear material (SNM) as reactor fuel. It eliminates the references to use of SNM for reactor operations and limits the possession of SNM to SNM "that was used" as reactor fuel at DCPP Unit 1. In accordance with 10 CFR 50.82(a)(2), once DCPP Unit 1 has permanently ceased operation and PG&E has certified that fuel has been permanently removed from the reactor, the 10 CFR Part 50 license will no longer authorize operation of the reactor. As such, PG&E has no need to receive SNM in the form of reactor fuel for DCPP Unit 1 and cannot use SNM as reactor fuel for reactor operations. The continued authorization to possess SNM "that was used" as reactor fuel for DCPP Unit 1 is necessary, as PG&E possesses the reactor fuel that was used for the past operations of the reactor.

The license condition is also revised to reflect the conversion of the "Final Safety Analysis Report" to the "Defueled Safety Analysis Report" upon implementation of this LAR. The proposed terminology is consistent with a decommissioning plant.

License Condition 2.B.(3)	
Current License Condition 2.B.(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;	Proposed License Condition 2.B.(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources *that were used* for reactor startup, sealed sources *that were used* for *calibration of* reactor instrumentation and *are used in the calibration of* radiation monitoring equipment ~~calibration~~, and as fission detectors in amounts as required;
Basis	
The proposed changes to this license condition remove the authorization for receipt and use of byproduct, source, and SNM as sealed neutron sources for reactor startup. The deletion of the authorization to receive and use sources for reactor startup is consistent with the fact that DCPP Unit 1 will no longer be authorized to operate. The authorization to possess such sources previously used for reactor startup is retained. The continued authorization to possess neutron sources that were used for reactor startup is consistent with the safe storage of byproduct, source, and SNM. The use of sources for radiation monitoring will continue to be required. After the certifications required by 10 CFR 50.82(a)(1) are docketed, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). These changes are consistent with the permanently defueled condition.	

License Condition 2.B.(5)	
Current License Condition 2.B.(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.	Proposed License Condition 2.B.(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials ~~as may be~~ *that were* produced by the operation of the facility.

Basis
This license condition is proposed for revision to allow possession of byproduct and SNM that were produced during operation of the reactor, but not allow the separation of material that was produced by operations of the reactor. In accordance with 10 CFR 50.82(a)(2), once DCPP Unit 1 has permanently ceased operation and PG&E has certified that fuel has been permanently removed from the reactor, the 10 CFR Part 50 license will no longer authorize operation of the reactor. This proposed license condition is consistent with the requirements associated with a decommissioning plant.

<table>
<tr><th colspan="2">License Condition 2.C.(1)</th></tr>
<tr><td><u>Current License Condition 2.C.(1)</u>

<u>Maximum Power Level</u>
The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.</td><td><u>Proposed License Condition 2.C.(1)</u>

<u>Deleted per Amendment No. ###.</u></td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This license condition is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once PG&E has permanently ceased operation of DCPP Unit 1 and certified that fuel has been permanently removed from the reactor, reference to operation of the facility would be inconsistent with the provisions of 10 CFR 50.82(a)(2).</td></tr>
</table>

<table>
<tr><th colspan="2">License Condition 2.C.(2)</th></tr>
<tr><td><u>Current License Condition 2.C.(2)</u>

<u>Technical Specifications</u>
The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 237 are hereby incorporated in the license. Pacific Gas and Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.</td><td><u>Proposed License Condition 2.C.(2)</u>

<u>Permanently Defueled Technical Specifications</u>
The <u>Permanently Defueled</u> Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. ~~237~~<u>XXX</u>, are hereby incorporated in the license. Pacific Gas and Electric Company shall ~~operate~~ <u>maintain</u> the facility in accordance with the <u>Permanently Defueled</u> Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.</td></tr>
</table>

Basis
This license condition is proposed for revision to reflect the nomenclature change to PDTS to more accurately describe the document. The designation from operating to maintaining the facility was also changed. Once PG&E has permanently ceased operation of DCPP Unit 1 and certified that fuel has been permanently removed from the reactor, reference to operation of the facility would be inconsistent with the provisions of 10 CFR 50.82(a)(2).

License Condition 2.C.(3)	
Current License Condition 2.C.(3) Initial Test Program The Pacific Gas and Electric Company shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Pacific Gas and Electric Company's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as: a. Elimination of any test identified in Section 14 of PG&E's Final Safety Analysis Report as amended as being essential; b. Modification of test objectives, methods, or acceptance criteria for any test identified in section 14 of PG&E's Final Safety Analysis Report, as amended, as being essential; c. Performance of any test at a power level different from that described in the program; and d. Failure to complete any test included in the described program (planned or scheduled for power levels up to the authorized power level).	Proposed License Condition 2.C.(3) *Deleted per Amendment No. ###.*
Basis	
This license condition reflects the initial startup test program and is historical. The license condition is proposed for deletion in its entirety as it will no longer be applicable during decommissioning.	

<table>
<tr><th colspan="2">License Condition 2.C.(4)</th></tr>
<tr><td><u>Current License Condition 2.C.(4)</u>

<u>Special Tests</u>

PG&E is authorized to perform steam generator moisture carryover studies and turbine performance tests at the Diablo Canyon Nuclear Power Plant, Unit 1. These studies involve the use of an aqueous tracer solution of three (3) curies of sodium-24. PG&E's personnel shall be in charge of conducting these studies and be knowledgeable in the procedures. PG&E shall impose personnel exposure limits, posting, and survey requirements in conformance with those in 10 CFR Part 20 to minimize personnel exposure and contamination during the studies. Radiological controls shall be established in the areas of the chemical feed, feedwater, steam, condensate and sampling systems where the presence of the radioactive tracer is expected to warrant such controls. PG&E shall take special precautions to minimize radiation exposure and contamination during both the handling of the radioactive tracer prior to injection and the taking of system samples following injection of the tracer. PG&E shall ensure that all regulatory requirements for liquid discharge are met during disposal of all sampling effluents and when re-establishing continuous blowdown from the steam generators after completion of the studies.</td><td><u>Proposed License Condition 2.C.(4)</u>

<u>Deleted per Amendment No. ###.</u></td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This license condition is proposed for deletion in its entirety. This license condition provides PG&E the authorization to perform SG moisture carryover studies and turbine performance tests for DCPP, Unit 1. In accordance with 10 CFR 50.82(a)(2), once DCPP Unit 1 has permanently ceased operation and PG&E has certified that fuel has been permanently removed from the reactor, the 10 CFR Part 50 license will no longer authorize operation of the reactor. Therefore, during decommissioning activities, the performance of SG moisture carryover studies and turbine performance tests for DCPP Unit 1 are not necessary and the license condition can be deleted.</td></tr>
</table>

<table>
<tr><th colspan="2">License Condition 2.C.(6)</th></tr>
<tr><td>
<u>Current License Condition 2.C.(6)</u>

<u>NUREG-0737 Conditions</u>

Each of the following conditions shall be completed to the satisfaction of the NRC as indicated below. Each of the following conditions references the appropriate Section in SER Supplements No. 10 and/or No. 12.

a. <u>Shift Technical Advisor (Section I.A.1.1)</u>
PG&E shall provide a fully-trained, on-shift technical advisor to the Shift Foreman.

b. <u>Shift Staffing (Section I.A.1.3)</u>
Until the plant has completed its startup test program, licensed personnel who are not regularly assigned members of the shift staff, including but not limited to the Operations Supervisor, shall not be assigned shift duties to satisfy the minimum staffing requirements for operation in Modes 1, 2, 3 and 4 except for cases of emergencies such as unexpected illness. Exceptions to this requirement may be made only after prior consultation with and approval by the NRC.

c. <u>Management of Operations (Section I.B.1)</u>
The Pacific Gas and Electric Company shall augment the plant staff to provide on each shift an individual experienced in comparable size pressurized water reactor operation. These individuals shall have at least one year of experience in operation of large pressurized water reactors or shall have participated in the startup of at least three pressurized water reactors.
</td><td>
<u>Proposed License Condition 2.C.(6)</u>

<u>Deleted per Amendment No. ###.</u>
</td></tr>
</table>

At least one such experienced individual shall be on duty on each shift through the startup test program whenever the reactor is not in a cold shutdown condition for at least the first year of operation or until the plant has attained a nominal 100% power level, whichever occurs first. d. <u>Procedures for Verifying Correct Performance of Operating Activities (Section I.C.6)</u> Procedures shall be available to verify the adequacy of the operating activities. e. Deleted. f. <u>Relief and Safety Valve Test Requirements (Section II.D.1)</u> PG&E shall implement the results of the EPRI test program. g. <u>Containment Isolation Dependability (Section II.E.4.2)</u> PG&E shall limit the 12-inch vacuum/overpressure relief valve opening to less than or equal to 50 degrees. h. <u>Calculations for Small-Break LOCAs (Sections II.K.3.30 and II.K.3.31)</u> PG&E is participating in the Westinghouse Owners Group effort for this item and shall conform to the results of this effort. Within one year of staff approval of the Westinghouse generic methodology for calculating small break LOCAs (II.K.3.30), PG&E shall submit a plant specific calculation (II.K.3.31) for staff review and approval. i. <u>Long-Term Emergency Preparedness (Section III.A.2)</u>	

(1) PG&E shall submit a detailed control room design review summary report by December 31, 1984. (2) PG&E shall complete operator training on the Safety Parameter Display System and emergency operating procedures by March 28, 1985. (3) PG&E shall implement emergency operating procedures based upon Westinghouse Owners Group guidelines by March 28, 1985.	
Basis	
This license condition is proposed for deletion in its entirety. NUREG-0737 (Reference 19) and NUREG-0737, Supplement 1 (Reference 20) implemented programmatic changes to the way reactor operators are trained, instrumentation information is presented, and procedures are structured using human factors and a function-oriented approach to address operating events and accidents. These accidents, and the associated emergency operating procedures to detect, respond to, and mitigate such accidents, concerned malfunctions of the reactor and its supporting systems and are not relevant to a permanently shut-down and defueled reactor. In accordance with 10 CFR 50.82(a)(2), once DCPP Unit 1 has permanently ceased operation and PG&E has certified that fuel has been permanently removed from the reactor, the 10 CFR Part 50 license will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel. Therefore, this license condition is no longer applicable and can be deleted.	

License Condition 2.C.(8)	
<u>Current License Condition 2.C.(8)</u> <u>Control of Heavy Loads (SSER 27, Section IV. 6)</u> Prior to startup following the first refueling outage, the licensee shall submit commitments necessary to implement changes and modifications as required to satisfy the guidelines of Section 5.1.2 through 5.1.6 of NUREG-0612 (Phase II: 9-month responses to the NRC Generic Letter dated December 22, 1980).	<u>Proposed License Condition 2.C.(8)</u> *<u>Deleted per Amendment No. ###.</u>*
Basis	
This license condition is proposed for deletion in its entirety. The NRC previously determined that this license condition for DCPP Unit 1 was no longer necessary and could be removed from the license via submittal of a LAR (Reference 25). Therefore,	

this license condition is historical and not applicable to DCPP Unit 1 during decommissioning.

License Condition 2.C.(9)	
<u>Current License Condition 2.C.(9) Emergency Preparedness (SSER 27, Section IV.3)</u> In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.	<u>Proposed License Condition 2.C.(9)</u> *<u>Deleted per Amendment No. ###.</u>*
Basis	
This license condition is proposed for deletion in its entirety. The above license condition was associated with the initial completion of procedures in the Federal Emergency Management Agency's final rule. In addition, 10 CFR 50.54(s)(2) states that if the NRC finds that the state of emergency preparedness does not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency and if the deficiencies are not corrected within four months of that finding, the Commission will determine whether the reactor shall be shut down until such deficiencies are remedied or whether other enforcement action is appropriate. This license condition is historical and 10 CFR 50.54(s)(2) will continue to apply regardless of the license condition and is therefore being proposed for deletion.	

License Condition 2.C.(10)	
<u>Current License Condition 2.C.(10)</u> <u>Masonry Walls (SSER-27, Section IV.4; Safety Evaluation of November 2, 1984)</u> Prior to start-up following the first refueling outage, the licensee shall (1) evaluate the differences in margins between the staff criteria as set forth in the Standard Review Plan and the criteria used by the licensee, and (2) provide justification acceptable to the staff for those cases where differences exist between the staff's and the licensee's criteria.	<u>Proposed License Condition 2.C.(10)</u> *<u>Deleted per Amendment No. ###.</u>*

Basis
This license condition is proposed for deletion in its entirety. The NRC previously determined that this license condition for DCPP Unit 1 was satisfied (Reference 26). Therefore, this license condition is historical and not applicable to decommissioning of the facility.

<table>
<tr><th colspan="2">License Condition 2.C.(11)</th></tr>
<tr><td><u>Current License Condition 2.C.(11)</u>

<u>Spent Fuel Pool Modification</u>

The licensee is authorized to modify the spent fuel pool as described in the application dated October 30, 1985 (LAR 85-13) as supplemented. Amendment No. 8 issued on May 30, 1986 and stayed by the U.S. Court of Appeals for the Ninth Circuit pending completion of NRC hearings is hereby reinstated.

Prior to final conversion to the modified rack design, fuel may be stored, as needed, in either the modified storage racks described in Technical Specification 5.6.1.1 or in the unmodified storage racks (or both) which are designed and shall be maintained with a nominal 21-inch center-to-center distance between fuel assemblies placed in the storage racks.</td><td><u>Proposed License Condition 2.C.(11)</u>

<u>Deleted per Amendment No. ###.</u></td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This license condition is proposed for deletion in its entirety. The current requirements for fuel storage racks are included in TS 4.3, Fuel Storage and TS 3.7.17, Spent Fuel Assembly Storage. Therefore, this license condition is historical and not applicable to decommissioning of the facility.</td></tr>
</table>

<table>
<tr><th colspan="2">License Condition 2.C.(12)</th></tr>
<tr><td><u>Current License Condition 2.C.(12)</u>

<u>Additional Conditions</u>

The Additional Conditions contained in Appendix D, as revised through Amendment No. 230, are hereby incorporated into this license. Pacific Gas and Electric Company shall operate the</td><td><u>Proposed License Condition 2.C.(12)</u>

<u>Deleted per Amendment No. ###.</u></td></tr>
</table>

facility in accordance with the Additional Conditions.	
Basis	
Appendix D as revised through Amendment No. 230, is an Appendix to the Operating License which includes additional conditions incorporated into the license through License Amendment 230 for DCPP Unit 1. The additional conditions are associated with implementation of specific license amendments for DCPP Unit 1. Reference to Appendix D, Additional Conditions, is being removed because, as discussed below in Section 2.2.2, this appendix is proposed for deletion in its entirety.	

New License Condition 2.C.(13)
Proposed New License Condition 2.C.(13) *Aging Management Program* *If all spent fuel has not been removed from the Unit 1 spent fuel pool prior to November 2, 2028, an aging management program shall be submitted prior to this date for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Defueled Safety Analysis Report and shall remain in effect for Unit 1 until such time that all spent fuel has been removed from the Unit 1 spent fuel pool.*
Basis
The initial operating license for Unit 1 is scheduled to expire on November 2, 2024. The Irradiated Fuel Management Plan (Reference 18) currently projects that all SNF from the Unit 1 SFP will be in dry storage in 2031. However, PG&E has issued a request for proposal to implement a modified or new dry cask storage design to potentially reduce the required SFP cooling time to allow safe transfer to the Diablo Canyon ISFSI as soon as possible and not to exceed 4 years after the expiration of the Unit 1 operating license (November 2, 2028). Therefore, the period of extended operations for structures and components associated with wet storage of spent fuel is anticipated to be approximately 10 percent of the initial licensed operating period. This minimal increase in service time for SSCs associated with the SFP, firewater system, and radiation monitoring system does not pose a nuclear safety concern for the reasons discussed below. Spent Fuel Pool In the permanently shutdown and defueled condition, safely storing SNF relies principally on maintaining: 1. storage geometry; 2. water level; and, 3. sub-critical characteristics of the spent fuel storage racks.

In accordance with 10 CFR 50.65(a), Requirements for monitoring the effectiveness of maintenance at nuclear power plants, nuclear power plants for which the licensee has submitted the certifications specified in 10 CFR 50.82(a)(1), must apply the requirements of this section to the extent that the licensee shall monitor the performance or condition of all SSCs associated with the storage, control, and maintenance of SNF in a safe condition, in a manner sufficient to provide reasonable assurance that these SSCs are capable of fulfilling their intended functions. PG&E will continue to implement the portion of the Maintenance Rule Program that is applicable to SSCs associated with the storage, control, and maintenance of SNF. DCPP Procedures will continue to be implemented to ensure SNF storage structures and components maintain structural integrity. This program includes the SFP liner and the SNF storage racks.

In addition to ensuring the structural integrity of the SFP and components, the design of the SFP protects against the possibility of a complete loss of water in the SFP. The SFP cooling suction connection enters near the normal water level so that the SFP cannot be gravity-drained. The cooling water return line contains an anti-siphon hole to prevent the possibility of gravity draining the SFP. The design ensures greater than ten feet of water exists over the top of the fuel assemblies should inadvertent drainage occur.

Criticality analyses have been performed for the permanent storage racks, which demonstrate that the multiplication factor, k_{eff}, of the fuel assemblies in the spent fuel storage racks is less than or equal to 0.95. In order to maintain k_{eff} less than or equal to 0.95, the presence of soluble boron is credited in the SFP criticality analyses. These criticality analyses were conservatively performed in the region of the SFP that does not contain Boraflex panels and therefore does not take credit for these panels. TS 3.7.16, “Spent Fuel Pool Boron Concentration,” requires that the SFP boron concentration be greater than or equal to 2000 parts per million (PPM) when fuel assemblies are stored in the SFP. In addition, TS 3.7.17, “Spent Fuel Assembly Storage,” ensures the configuration of fuel assemblies in the SFP will maintain the k_{eff} of the pool is less than or equal to 0.95, under analyzed accident scenarios.

PG&E concludes that the actions described above are sufficient to ensure safe storage of spent fuel in the SFP.

Fire Water System

As part of the transition to decommissioning, PG&E will implement a Fire Protection Program that complies with 10 CFR 50.48(f). The components and maintenance of the fire protection system will meet National Fire Protection Association codes, as applicable, and any necessary exceptions or deviations from these codes and standards will be approved by the NRC. The firewater system is a diverse system that provides multiple flow paths to move water throughout the plant which uses an underground loop for plant area water supply. From an emergency planning perspective, firewater can be used as a means to provide makeup water to the SFP.

The firewater system water supply is normally obtained from the raw water reservoir, which is gravity fed to the fire suppression header and has a minimum usable volume of 360,000 gallons. The firewater system can also be supplied from the Firewater Storage Tank 0-1, which has a minimum contained volume of 270,000 gallons, through Fire Pump 0-1 or 0-2. The water supply to the firewater system is either (1) filtered, sterilized and desalinated sea water, or (2) filtered well water. The Maintenance Rule Structural Monitoring Program includes inspection of the firewater storage tank and the raw water reservoir. Fire Pumps 0-1 and 0-2 will continue to be run on a quarterly basis with pump performance testing (suction and discharge pressures, and flow rates), pump and motor vibration readings, motor voltages and currents, performed on a nominal 18-month frequency.

Visual checks of the position and condition of firewater system valves in the flow path from the water source to the end use system, hose reel or hydrant is performed every 31 days (except valves that are located inside containment and are locked, sealed, or otherwise secured in position). Each testable valve in the flow path is cycled through at least one complete cycle of full travel every 12 months. In addition, the indoor hoses are hydrostatically tested every 3 years. Within the firewater system there are multiple potential flow paths allowing for valve alignment to sectionalize any potentially damaged or out of service portions of the system while still providing acceptable flow paths.

PG&E concludes that the actions described above to maintain and test the firewater system are sufficient to ensure continued functionality.

Radiation Monitoring

For indication and monitoring of spent fuel conditions, during continued wet storage of the fuel up to and beyond the original 40-year life of the plant, DCPP Unit 1 will maintain portions of the radiation monitoring system. This will consist of two SFP Area Monitors (RM-58 and RM-59). RM-58 is located near the SFP and RM-59 is located near the new fuel storage area. These monitors provide continuous monitoring and indication with alert and high radiation level alarms in the main CR. Local audible and visual indicators are also provided. During decommissioning, these local area radiation monitors will be tested quarterly and maintained to ensure continued functionality. Contingency plans such as manual radiation surveys can be implemented should a monitor become non-functional. Other surveillance activities, such as observation of the pool water level is performed weekly to assure that abnormal conditions in the pool are not developing.

In the permanently shutdown and defueled condition these radiation monitors will continue to be maintained as Equipment Important to Emergency Response. RM-58 and RM-59 will continue to be included in DCPP procedures associated with Equipment Important to Emergency Response with specific activities identified as compensatory measures. The monitors will continue to be included in DCPP

procedures as equipment important to emergency response after conversion to the Post Shutdown and Permanently Defueled Emergency Plans as long as fuel remains in the SFP. Components associated with maintaining the spent fuel in a safe configuration are important to safety, while the radiation monitoring system is used to alert operators of an unusual or catastrophic event. Failure of any of the radiation monitoring components or devices will have no impact on the performance of the important to safety functions. If the radiation monitors in the SFP area fail, compensatory actions, such as installing portable radiation monitors or manual local area surveys or grab samples, will be instituted to assure that the CR staff remain cognizant of the radiation areas around the spent fuel. PG&E concludes that the actions described above to maintain the applicable radiation monitors are sufficient to ensure continued functionality. The above proposed license condition is similar to the NRC language included in Reference 10 for Crystal River, Unit 3.

License Condition 2.D	
<u>Current License Condition 2.D</u> <u>Exemption</u> Exemption from certain requirements of Appendix J to 10 CFR Part 50 is described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 9. This exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. Therefore, this exemption, previously granted in Facility Operating License No. DPR-76, is hereby reaffirmed. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.	<u>Proposed License Condition 2.D</u> <u>*Deleted per Amendment No. ###.*</u>
Basis	
In accordance with 10 CFR Part 50.54(o), “Condition of Licenses,” “Primary reactor containments for water cooled power reactors, other than facilities for which the certifications required under §§ 50.82(a)(1) or 52.110(a)(1) of this chapter have been submitted, shall be subject to the requirements set forth in appendix J to this part.” Once DCPP Unit 1 has permanently ceased operation and PG&E has certified that	

fuel has been permanently removed from the reactor, 10 CFR 50, Appendix J is no longer applicable and therefore the associated exemptions are not necessary. Therefore, this license condition is not applicable during decommissioning and the license condition can be deleted.

License Condition 2.J	
Current License Condition 2.J Term of License This License is effective as of the date of issuance and shall expire at midnight on November 2, 2024.	Proposed License Condition 2.J Term of License This License is effective as of the date of issuance and ~~shall expire at midnight on November 2, 2024~~*is effective until the Commission notifies the licensee in writing that the license is terminated*.
Basis	
This license condition is revised to conform to 10 CFR 50.51, "Continuation of license," in that the license authorizes ownership and possession of the facility until the Commission notifies the licensee in writing that the license is terminated.	

Attachments	
Current List of Attachments Attachments: 1. Appendix A – Technical Specifications 2. Appendix B – Environmental Protection Plan 3. Appendix C – Deleted 4. Appendix D – Additional Conditions	Proposed List of Attachments Attachments: 1. Appendix A – *Permanently Defueled* Technical Specifications 2. Appendix B – Environmental Protection Plan 3. Appendix C – Deleted 4. Appendix D – ~~Additional Conditions~~ *Deleted*
Basis	
This list of attachments is proposed for revision to reflect the changes described in this LAR.	

2.2.2 DCPP Unit 1 Facility Operating License, Appendix D, Additional Conditions - Proposed Changes

Appendix D of FOL Number DPR-80 includes additional conditions with specific schedules associated with previous license amendments that PG&E was required to comply with. As described below Appendix D, is proposed for deletion in its entirety.

Amendment Number	Additional Condition	Implementation Date
120	The licensee is authorized to relocate certain technical specifications requirements to the equipment control guidelines (ECGs) as referenced in the Updated Final Safety Analysis Report. Implementation of these amendments shall include relocation of these technical specification requirements to the ECGs as described in the licensee's application dated October 4, 1995, as supplemented by letters dated July 17, 1996, August 20, 1996, and June 2, 1997, and evaluated in the staff's safety evaluation dated February 3, 1998.	The amendment shall be implemented within 90 days of its issuance.
Basis for Deletion		
In Reference 27, PG&E notified the NRC that the above license amendment had been implemented in accordance with the license condition. Therefore, this additional condition is historical and is proposed for deletion.		

Amendment Number	Additional Condition	Implementation Date
135	This amendment authorizes the relocation of certain Technical Specification requirements to licensee controlled documents. Implementation of this amendment shall include the relocation of these Technical Specification requirements to the appropriate documents, as described in Table LG of Details Relocated from Current Technical Specifications, Table R of Relocated Current Technical Specifications, Table LS of Less Restrictive Changes to Current Technical Specifications, and Table A of Administrative Changes to Current Technical Specifications that are attached to the NRC staff's Safety Evaluation enclosed with this amendment.	The amendment shall be implemented by June 30, 2000.
Basis for Deletion		
In Reference 28, PG&E notified the NRC that the above license amendment had been implemented. Therefore, this additional condition is historical and is proposed for deletion.		

Amendment Number	Additional Condition	Implementation Date
135	The schedule for the performance of new and revised Surveillance Requirements (SR) shall be as follows: For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment. For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment. For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment. For SRs that existed prior to this amendment whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to implementation of this amendment.	The amendment shall be implemented by June 30, 2000.
Basis for Deletion		
In Reference 28, PG&E notified the NRC that the above license amendment had been implemented. Therefore, this additional condition is historical and is proposed for deletion.		

Amendment Number	Additional Condition	Implementation Date
201	Determination of CRE unfiltered air inleakage as required by surveillance	The amendment is effective as of the date of its issuance and the

	requirement (SR) 3.7.10.5, in accordance with TS 5.5.19.c.(i). The assessment of CRE habitability as required by TS 5.5.19.c.(ii). The measurement of CRE pressure as required by TS 5.5.19.d. Following implementation, this condition will be performed as stated in the condition: The first performance of SR 3.7.10.5, in accordance with Specification 5.5.19.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful tracer gas test, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years. The first performance of the periodic assessment of CRE habitability, Specification 5.5.19.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful tracer gas test, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years. The first performance of the periodic measurement of CRE pressure, Specification 5.5.19.d, shall be within 24 months, plus the 182 days allowed by SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful pressure	condition shall be implemented within 180 days of its issuance.

	measurement test, or within 182 days if not performed previously.	
Basis for Deletion		
PG&E completed implementation of License Amendment 201 as required in the above additional condition. Therefore, this additional condition is no longer required and is proposed for deletion.		

Amendment Number	Additional Condition	Implementation Date
230	Implementation of the amendment adopting the alternative source term shall include the following plant modifications: Install shielding material, equivalent to that provided by the Control Room outer walls, at the external concrete west wall of the Control Room briefing room. Install a high efficiency particulate air filter in the Technical Support Center normal ventilation system. Re-classify a portion of the 40-inch Containment Penetration Area (GE/GW) Ventilation line from PG&E Design Class II to PG&E Design Class I and upgrade the damper actuators, pressure switches, and the damper solenoid valves to PG&E Design Class I. Update setpoints for the redundant safety related gamma sensitive area radiation monitors (1-RE 25/26, 2-RE 25/26).	The amendment is effective as of the date of its issuance and the condition shall be implemented within 365 days of its issuance
Basis for Deletion		
PG&E completed implementation of License Amendment 230 as required in the above additional condition. Therefore, this additional condition is no longer required and is proposed for deletion.		

2.2.3 DCPP Unit 2 Facility Operating License - Proposed Changes

License Finding 1.B	
Current License Condition 1.B Construction of the Diablo Canyon Nuclear Power Plant, Unit 2 (the facility), has been substantially completed in conformity with Provisional Construction Permit No. CPPR-69 and the application, as amended, the provisions of the Act, and the regulations of the Commission;	Proposed License Condition 1.B *Deleted per Amendment No. ###.*
Basis	
This license finding is proposed for deletion in its entirety. Decommissioning of DCPP Unit 2 is not dependent on the regulations that governed the construction of the facility.	

License Finding 1.C	
Current License Finding 1.C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission, except as exempted from compliance in Section 2.D below;	Proposed License Finding 1.C The facility will ~~operate~~*be maintained* in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission~~, except as exempted from compliance in Section 2.D below~~;
Basis	
In accordance with 10 CFR 50.82(a)(2), once PG&E has permanently ceased operation of DCPP Unit 2 and certified that fuel has been permanently removed from the reactor, the license will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel. The removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with a decommissioning plant. Reference to the exemption in Section 2.D is proposed for deletion because Section 2.D is proposed for deletion in its entirety as described below.	

License Finding 1.D	
Current License Finding 1.D. There is reasonable assurance (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the	Proposed License Finding 1.D There is reasonable assurance (i) that the activities authorized by this ~~operating~~ license can be conducted without endangering the health and safety of the

public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter I, except as exempted from compliance in Section 2.D below;	public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter I~~, except as exempted from compliance in Section 2.D below~~;
Basis	
In accordance with 10 CFR 50.82(a)(2), once PG&E has permanently ceased operation of DCPP Unit 2 and certified that fuel has been permanently removed, the license will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel. The removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with a decommissioning plant. Reference to the exemption in Section 2.D is proposed for deletion because Section 2.D is proposed for deletion in its entirety as described below.	

License Finding 1.E	
Current License Finding 1.E The Pacific Gas and Electric Company is technically qualified to engage in the activities authorized by this operating license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;	Proposed License Finding 1.E The Pacific Gas and Electric Company is technically qualified to engage in the activities authorized by this~~ operating~~ license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;
Basis	
Once PG&E has permanently ceased operation of DCPP Unit 2 and certified that fuel has been permanently removed, the license will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel. The removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with a decommissioning plant.	

License Finding 1.H	
Current License Finding 1.H After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. DPR-82, subject to the conditions for protection of the environment set forth herein, is in accordance with applicable Commission	Proposed License Finding 1.H After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility ~~Operating~~ License No. DPR-82, subject to the conditions for protection of the environment set forth herein, is in accordance with applicable Commission

regulations governing environmental reviews (10 CFR Part 50, Appendix D and 10 CFR Part 51) and all applicable requirements have been satisfied; and	regulations governing environmental reviews (10 CFR Part 50, Appendix D and 10 CFR Part 51) and all applicable requirements have been satisfied; and
Basis	
Once PG&E has permanently ceased operation of DCPP Unit 2 and certified that fuel has been permanently removed, the license will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel. The removal of the term operating provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with a decommissioning plant.	

License Condition 2	
Current License Condition 2 Pursuant to approval by the Nuclear Regulatory Commission in its Memorandum and Order (CLI-85-14) dated August 1, 1985, the license for fuel loading and low power testing, Facility Operating License No. DPR-81, issued on April 26, 1985, is superseded by Facility Operating License No. DPR-82, hereby issued to Pacific Gas and Electric Company to read as follows:	Proposed License Condition 2 Pursuant to approval by the Nuclear Regulatory Commission in its Memorandum and Order (CLI-85-14) dated August 1, 1985, the license for fuel loading and low power testing, Facility Operating License No. DPR-81, issued on April 26, 1985, is superseded by Facility ~~Operating~~ License No. DPR-82, hereby issued to Pacific Gas and Electric Company to read as follows:
Basis	
Once PG&E has permanently ceased operation of DCPP Unit 2 and certified that fuel has been permanently removed, the license will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel. The removal of the term operating provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with a decommissioning plant.	

License Condition 2.A	
Current License Condition 2.A This License applies to the Diablo Canyon Nuclear Power Plant, Unit 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by PG&E. The facility is located in San Luis Obispo County, California, and is described in PG&E's Final Safety Analysis Report as supplemented and	Proposed License Condition 2.A This License applies to the Diablo Canyon Nuclear Power Plant, Unit 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by PG&E. The facility is located in San Luis Obispo County, California, and is described in PG&E's ~~Final~~*Defueled* Safety Analysis Report as supplemented

amended, and the Environmental Report as supplemented and amended.	and amended, and the Environmental Report as supplemented and amended.
Basis	
License Condition 2.A is revised to reflect the conversion of the "Final Safety Analysis Report" to the "Defueled Safety Analysis Report" upon implementation of this LAR. The proposed terminology is consistent with a decommissioning plant.	

License Condition 2.B.(1)	
Current License Condition 2.B.(1) Pursuant to Section 104(b) of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess, use, and operate the facility at the designated location in San Luis Obispo County, California, in accordance with the procedures and limitations set forth in this license;	Proposed License Condition 2.B.(1) Pursuant to Section 104(b) of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess~~,~~ *and* use~~, and operate~~ the facility at the designated location in San Luis Obispo County, California, in accordance with the procedures and limitations set forth in this license;
Basis	
In accordance with 10 CFR 50.82(a)(2), once PG&E has permanently ceased operation of DCPP Unit 2 and certified that fuel has been permanently removed, the license will no longer authorize operation of the reactor. As such, the facility will remain authorized to possess the existing SNF and use the systems required to support safe fuel storage during the decommissioning period, in accordance with the specified limitations for storage.	

License Condition 2.B.(2)	
Current License Condition 2.B.(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;	Proposed License Condition 2.B.(2) Pursuant to the Act and 10 CFR Part 70, to~~ receive,~~ possess~~, and use~~ at any time special nuclear material *that was used* as reactor fuel, in accordance with the limitations for storage~~ and amounts required for reactor operation~~, as described in the ~~Final~~ *Defueled* Safety Analysis Report, as supplemented and amended;
Basis	
The proposed change to this license condition removes the authorization for receipt and use of SNM as reactor fuel. It eliminates the references to use of SNM for reactor operations and limits the possession of SNM to SNM "that was used" as reactor fuel at DCPP Unit 2. In accordance with 10 CFR 50.82(a)(2), once DCPP Unit 2 has permanently ceased operation and PG&E has certified that fuel has been permanently removed from the reactor, the 10 CFR Part 50 license will no longer	

authorize operation of the reactor. As such, PG&E has no need to receive SNM in the form of reactor fuel for DCPP Unit 2 and cannot use SNM as reactor fuel for reactor operations. The continued authorization to possess SNM "that was used" as reactor fuel for DCPP Unit 2 is necessary, as PG&E possesses the reactor fuel that was used for the past operations of the reactor. The license condition is also revised to reflect the conversion of the "Final Safety Analysis Report" to the "Defueled Safety Analysis Report" upon implementation of this LAR. The proposed terminology is consistent with a decommissioning plant.

<table>
<tr><th colspan="2">License Condition 2.B.(3)</th></tr>
<tr><td>Current License Condition 2.B.(3)

Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;</td><td>Proposed License Condition 2.B.(3)

Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in the calibration of radiation monitoring equipment <s>calibration</s>, and as fission detectors in amounts as required;</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">The proposed changes to this license condition remove the authorization for receipt and use of byproduct, source, and SNM as sealed neutron sources for reactor startup. The deletion of the authorization to receive and use sources for reactor startup is consistent with the fact that DCPP Unit 2 will no longer be authorized to operate.

The authorization to possess such sources previously used for reactor startup is retained. The continued authorization to possess neutron sources that were used for reactor startup is consistent with the safe storage of byproduct, source, and SNM. The use of sources for radiation monitoring will continue to be required.

After the certifications required by 10 CFR 50.82(a)(1) are docketed, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). These changes are consistent with the permanently defueled condition.</td></tr>
</table>

License Condition 2.B.(5)	
Current License Condition 2.B.(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.	Proposed License Condition 2.B.(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials ~~as may be~~ *that were* produced by the operation of the facility.
Basis	
This license condition is proposed for revision to allow possession of byproduct and SNM that were produced during operation of the reactor, but not allow the separation of material that was produced by operations of the reactor. In accordance with 10 CFR 50.82(a)(2), once DCPP Unit 2 has permanently ceased operation and PG&E has certified that fuel has been permanently removed from the reactor, the 10 CFR Part 50 license no longer authorizes operation of the reactor. This proposed license condition is consistent with the requirements associated with a decommissioning plant.	

License Condition 2.C.(1)	
Current License Condition 2.C.(1) Maximum Power Level The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.	Proposed License Condition 2.C.(1) *Deleted per Amendment No. ###.*
Basis	
This license condition is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once PG&E has permanently ceased operation of DCPP Unit 2 and certified that fuel has been permanently removed from the reactor, reference to operation of the facility would be inconsistent with the provisions of 10 CFR 50.82(a)(2).	

License Condition 2.C.(2)	
Current License Condition 2.C.(2) Technical Specifications The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. *239*, are hereby incorporated in the license. Pacific Gas and Electric Company shall	Proposed License Condition 2.C.(2) *Permanently Defueled* Technical Specifications The *Permanently Defueled* Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No.~~239~~*XXX*, are

operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.	hereby incorporated in the license. Pacific Gas and Electric Company shall ~~operate~~ *maintain* the facility in accordance with the *Permanently Defueled* Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.
Basis	
This license condition is proposed for revision to reflect the nomenclature change to PDTS to more accurately describe the document. The designation from operating to maintaining the facility was also changed. Once PG&E has permanently ceased operation of DCPP Unit 2 and certified that fuel has been permanently removed from the reactor, reference to operation of the facility would be inconsistent with the provisions of 10 CFR 50.82(a)(2).	

License Condition 2.C.(3)	
Current License Condition 2.C.(3) Initial Test Program (SSER 31, Section 4.4.1) Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.	Proposed License Condition 2.C.(3) *Deleted per Amendment No. ###.*
Basis	
This license condition is applicable to the initial startup test program and is historical. The license condition is proposed for deletion in its entirety as it is no longer applicable during decommissioning.	

License Condition 2.C.(5)	
Current License Condition 2.C.(5) NUREG-0737 Items Each of the following conditions shall be completed to the satisfaction of the NRC as indicated below. Each condition	Proposed License Condition 2.C.(5) *Deleted per Amendment No. ###.*

references the appropriate Section in SER Supplements. a. <u>I.D.1 Detailed Control Room Design Review (SSER 31, Section 4.13)</u> PG&E shall comply with the requirements of Supplement 1 to NUREG-0737 for the conduct of a Detailed Control Room Design Review (DCRDR) in accordance with a schedule acceptable to the NRC staff. b. <u>II.E.4.2 Containment Isolation Dependability (SSER 31, Section 4.21)</u> PG&E shall limit the 12-inch vacuum/overpressure relief valve opening to less than or equal to 50 degrees.	
Basis	
This license condition is proposed for deletion in its entirety. NUREG-0737 (Reference 19) and NUREG-0737, Supplement 1 (Reference 20) implemented programmatic changes to the way reactor operators are trained, instrumentation information is presented, and procedures are structured using human factors and a function-oriented approach to address operating events and accidents. These accidents, and the associated emergency operating procedures to detect, respond to, and mitigate such accidents, concerned malfunctions of the reactor and its supporting systems and are not relevant to a permanently shutdown and defueled reactor. In accordance with 10 CFR 50.82(a)(2), once DCPP Unit 2 has permanently ceased operation and PG&E has certified that fuel has been permanently removed from the reactor, the 10 CFR Part 50 license will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel. Therefore, this license condition is no longer applicable and can be deleted.	

License Condition 2.C.(6)	
<u>Current License Condition 2.C.(6)</u> <u>Emergency Preparedness (SSER 31, Section 4.23.2 and SSER 32, Section 7)</u> In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR	<u>Proposed License Condition 2.C.(6)</u> *<u>Deleted per Amendment No. ###.</u>*

Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.	
Basis	
The above license condition was associated with the initial completion of procedures in the Federal Emergency Management Agency's final rule. In addition, 10 CFR 50.54(s)(2) states that if the NRC finds that the state of emergency preparedness does not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency and if the deficiencies are not corrected within four months of that finding, the Commission will determine whether the reactor shall be shut down until such deficiencies are remedied or whether other enforcement action is appropriate. This license condition is historical and 10 CFR 50.54(s)(2) will continue to apply regardless of the license condition and is therefore being proposed for deletion.	

License Condition 2.C.(7)	
Current License Condition 2.C.(7) Masonry Walls (SSER 31, Section 4.7) Prior to start-up following the first refueling outage, PG&E shall (1) evaluate the differences in margins between the staff criteria as set forth in the Standard Review Plan and the criteria used by the licensee, and (2) provide justification acceptable to the staff for those cases where differences exist between the staff's and PG&E's criteria.	Proposed License Condition 2.C.(7) *Deleted per Amendment No. ###.*
Basis	
This license condition is proposed for deletion in its entirety. The NRC previously determined that this license condition for DCPP Unit 2 was satisfied (Reference 26). Therefore, this license condition is historical and not applicable to decommissioning the facility.	

License Condition 2.C.(8)	
Current License Condition 2.C.(8) Reactor Trip System Reliability – Generic Letter 83-28 (SSER 31, Section 4.8)	Proposed License Condition 2.C.(8) *Deleted per Amendment No. ###.*

PG&E shall submit responses to and implement the requirements of Generic Letter 83-28 on a schedule which is consistent with that given in the PG&E letters dated January 24, and March 13, 1985.	
Basis	
This license condition is proposed for deletion in its entirety. As shown in References 29 through 45, PG&E submitted all required responses and satisfied the requirements of Generic Letter 83-28. In addition, the focus of Generic Letter 83-28 was the reactor trip system (RTS), which will no longer be applicable in the permanently defueled condition. Therefore, this license condition has been satisfied and can be deleted in the PDTS.	

License Condition 2.C.(9)	
<u>Current License Condition 2.C.(9)</u> <u>Steam Generator Tube Rupture Analysis (SSER 31, Section 4.25)</u> By April 1988, PG&E shall submit for NRC review and approval an analysis which demonstrates that the steam generator tube rupture (SGTR) analysis presented in the FSAR is the most sever case with respect to the release of fission products and calculated doses. Consistent with the analytical assumptions, PG&E shall propose all necessary changes to the Technical Specifications (Appendix A) to this license.	<u>Proposed License Condition 2.C.(9)</u> <u>*Deleted per Amendment No. ###.*</u>
Basis	
This license condition is proposed for deletion in its entirety. The NRC previously determined that this license condition for DCPP Unit 2 was satisfied (Reference 21). Therefore, this license condition has been satisfied and can be deleted.	

License Condition 2.C.(10)	
<u>Current License Condition 2.C.(10)</u> <u>Pipeway Structure DE and DDE Analysis (SSER 32, Section 4)</u>	<u>Proposed License Condition 2.C.(10)</u> <u>*Deleted per Amendment No. ###.*</u>

Prior to start-up following the first refueling outage PG&E shall complete a confirmatory analysis for the pipeway structure to further demonstrate the adequacy of the pipeway structure for load combinations that include the design earthquake (DE) and double design earthquake (DDE).	
Basis	
This license condition is proposed for deletion in its entirety. The NRC previously determined that this license condition for DCPP Unit 2 was satisfied (Reference 22). Therefore, this license condition has been satisfied and can be deleted.	

License Condition 2.C.(11)	
Current License Condition 2.C.(11) Spent Fuel Pool Modification The licensee is authorized to modify the spent fuel pool as described in the application dated October 30, 1985 (LAR 85-13) as supplemented. Amendment No. 6 issued on May 30, 1986 and stayed by the U.S. Court of Appeals for the Ninth Circuit pending completion of NRC hearings is reinstated. Prior to final conversion to the modified rack design, fuel may be stored, as needed, in either the modified storage racks described in Technical Specification 5.6.1.1 or in the unmodified storage racks (or both) which are designed and shall be maintained with a nominal 21-inch center-to-center distance between fuel assemblies placed in the storage racks.	Proposed License Condition 2.C.(11) *Deleted per Amendment No. ###.*
Basis	
This license condition is proposed for deletion in its entirety. The current requirements for fuel storage racks are included in TS 4.3, “Fuel Storage,” and TS 3.7.17, “Spent Fuel Assembly Storage.” Therefore, this license condition is historical and not applicable to decommissioning of the facility.	

<table>
<tr><th colspan="2">License Condition 2.C.(12)</th></tr>
<tr><td>Current License Condition 2.C.(12)

Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 232, are hereby incorporated into this license. Pacific Gas and Electric Company shall operate the facility in accordance with the Additional Conditions.</td><td>Proposed License Condition 2.C.(12)

Deleted per Amendment No. ###.</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">Appendix D as revised through Amendment No. 232, is an Appendix to the Operating License which includes additional conditions incorporated into the license through License Amendment 232 for DCPP Unit 2. The additional conditions are associated with implementation of specific license amendments for DCPP Unit 2. Reference to Appendix D, “Additional Conditions,” is being removed because, as discussed below in Section 2.2.4, this appendix is proposed for deletion in its entirety.</td></tr>
</table>

<table>
<tr><th>New License Condition 2.C.(13)</th></tr>
<tr><td>Proposed New License Condition 2.C.(13)

If all spent fuel has not been removed from the Unit 2 spent fuel pool prior to August 26, 2029, an aging management program shall be submitted prior to this date for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Defueled Safety Analysis Report and shall remain in effect for Unit 2 until such time that all spent fuel has been removed from the Unit 2 spent fuel pool.</td></tr>
<tr><th>Basis</th></tr>
<tr><td>The initial operating license for Unit 2 is scheduled to expire on August 26, 2025. The Irradiated Fuel Management Plan (Reference 18) currently projects that all SNF from the Unit 2 SFP will be in dry storage in 2032. However, PG&E has issued a request for proposal to implement a modified or new dry cask storage design, to potentially reduce the required SFP cooling time to allow safe transfer to the Diablo Canyon ISFSI as soon as possible and not to exceed 4 years after the expiration of the Unit 2 operating license (August 26, 2029). Therefore, the period of extended operations for structures and components associated with wet storage of SNF is anticipated to be approximately 10 percent of the initial licensed operating period. This minimal increase in service time for SSCs associated with the SFP, firewater system, and radiation monitoring system does not pose a nuclear safety concern for the reasons discussed below.</td></tr>
</table>

<u>Spent Fuel Pool</u>
In the permanently shutdown and defueled condition, safely storing SNF relies principally on maintaining:

1. storage geometry;
2. water level; and,
3. sub-critical characteristics of the spent fuel storage racks.

In accordance with 10 CFR 50.65(a), “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” nuclear power plants for which the licensee has submitted the certifications specified in 10 CFR 50.82(a)(1), must apply the requirements of this section to the extent that the licensee shall monitor the performance or condition of all SSCs associated with the storage, control, and maintenance of SNF in a safe condition, in a manner sufficient to provide reasonable assurance that these SSCs are capable of fulfilling their intended functions. PG&E will continue to implement the portion of the Maintenance Rule Program that is applicable to SSCs associated with the storage, control, and maintenance of SNF. DCPP procedures will continue to be implemented to ensure SNF storage structures and components maintain structural integrity. This program includes the SFP liner and the SNF storage racks.

In addition to ensuring the structural integrity of the SFP and components, the design of the SFP protects against the possibility of a complete loss of water in the SFP. The SFP cooling suction connection enters near the normal water level so that the SFP cannot be gravity-drained. The cooling water return line contains an anti-siphon hole to prevent the possibility of gravity draining the SFP. The design ensures greater than ten feet of water exists over the top of the fuel assemblies should inadvertent drainage occur.

Criticality analyses have been performed for the permanent storage racks, which demonstrate that the multiplication factor, k_{eff}, of the fuel assemblies in the spent fuel storage racks is less than or equal to 0.95. In order to maintain k_{eff} less than or equal to 0.95, the presence of soluble boron is credited in the SFP criticality analyses. These criticality analyses were conservatively performed in the region of the SFP that does not contain Boraflex panels and therefore does not take credit for these panels. TS 3.7.16, “Spent Fuel Pool Boron Concentration,” requires that the SFP boron concentration be greater than or equal to 2000 PPM when fuel assemblies are stored in the SFP. In addition, TS 3.7.17, “Spent Fuel Assembly Storage,” ensures the configuration of fuel assemblies in the SFP will maintain the k_{eff} of the pool less than or equal to 0.95, under analyzed accident scenarios.

PG&E concludes that the actions described above are sufficient to ensure safe storage of spent fuel in the SFP.

Fire Water System
As part of the transition to decommissioning PG&E will implement a Fire Protection Program that complies with 10 CFR 50.48(f). The components and maintenance of the fire protection system will meet National Fire Protection Association codes, as applicable, and any necessary exceptions or deviations from these codes and standards will be approved by the NRC. The firewater system is a diverse system that provides multiple flow paths to move water throughout the plant which uses an underground loop for plant area water supply. From an emergency planning perspective, firewater can be used as a means to provide makeup water to the SFP.

The firewater system water supply is normally obtained from the raw water reservoir, which is gravity fed to the fire suppression header and has a minimum usable volume of 360,000 gallons. The firewater system can also be supplied from the Firewater Storage Tank 0-1, which has a minimum contained volume of 270,000 gallons, through Fire Pump 0-1 or 0-2. The water supply to the firewater system is either (1) filtered, sterilized and desalinated sea water, or (2) filtered well water. The Maintenance Rule Structural Monitoring Program includes inspection of the firewater storage tank and the raw water reservoir. Fire Pumps 0-1 and 0-2 will continue to be run on a quarterly basis with pump performance testing (suction and discharge pressures, and flow rates), pump and motor vibration readings, motor voltages, and currents, performed on a nominal 18-month frequency.

Visual checks of the position and condition of firewater system valves in the flow path from the water source to the end use system, hose reel or hydrant is performed every 31 days (except values that are located inside containment and are locked, sealed, or otherwise secured in position). Each testable valve in the flow path is cycled through at least one complete cycle of full travel every 12 months. In addition, the indoor hoses are hydrostatically tested every 3 years. Within the firewater system there are multiple potential flow paths allowing for valve alignment to sectionalize any potentially damaged or out of service portions of the system while still providing acceptable flow paths.

PG&E concludes that the actions described above to maintain and test the firewater system are sufficient to ensure continued functionality.

Radiation Monitoring
For indication and monitoring of spent fuel conditions, during continued wet storage of the fuel up to and beyond the original 40-year life of the plant, DCPP Unit 2 will maintain portions of the radiation monitoring system. This will consist of two SFP Area Monitors (RM-58 and RM-59). RM-58 is located near the SFP and RM-59 is located near the new fuel storage area. These monitors provide continuous monitoring and indication with alert and high radiation level alarms in the main CR. Local audible and visual indicators are also provided. During decommissioning, these local area radiation monitors will be tested quarterly and maintained to ensure continued functionality. Contingency plans such as portable radiation monitors or

manual radiation surveys can be implemented should the monitors become non-functional. Other surveillance activities, such as observation of the pool water level is performed weekly to assure that abnormal conditions in the pool are not developing. In the permanently shutdown and defueled condition these radiation monitors will continue to be maintained as Equipment Important to Emergency Response. RM-58 and RM-59 will continue to be included in DCPP procedures associated with Equipment Important to Emergency Response with specific activities identified as compensatory measures. The monitors will continue to be included in DCPP procedures as Equipment Important to Emergency Response after conversion to the Post Shutdown and Permanently Defueled Emergency Plans as long as fuel remains in the SFP. Components associated with maintaining the spent fuel in a safe configuration are important to safety, while the radiation monitoring system is used to alert operators of an unusual or catastrophic event. Failure of any of the radiation monitoring components or devices will have no impact on the performance of the important to safety function. If the radiation monitors in the SFP area fail, compensatory actions, such as installing portable radiation monitors or manual local area surveys or grab samples, will be instituted to assure that the CR staff remain cognizant of the radiation areas around the spent fuel. PG&E concludes that the actions described above to maintain the applicable radiation monitors are sufficient to ensure continue functionality. The above proposed license condition is similar to the NRC language included in Reference 10 for Crystal River, Unit 3.

License Condition 2.D	
<u>Current License Condition 2.D</u> <u>Exemption (SSER 31, Section 6.2.6)</u> An exemption from certain requirements of Appendix J to 10 CFR Part 50 is described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 9. This exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. Therefore, this exemption previously granted in Facility Operating License No. DPR-81 pursuant to 10 CFR 50.12 is hereby reaffirmed. The	<u>Proposed License Condition 2.D</u> <u>*Deleted per Amendment No. ###.*</u>

facility will operate, with the exemption authorized, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.	
Basis	
In accordance with 10 CFR Part 50.54(o), “Condition of Licenses,” “Primary reactor containments for water cooled power reactors, other than facilities for which the certifications required under §§ 50.82(a)(1) or 52.110(a)(1) of this chapter have been submitted, shall be subject to the requirements set forth in appendix J to this part.” Once DCPP Unit 2 has permanently ceased operation and PG&E has certified that fuel has been permanently removed from the reactor, 10 CFR 50, Appendix J will no longer be applicable and the associated exemptions are not necessary. Therefore, this license condition is not applicable during decommissioning and can be deleted.	

License Condition 2.J	
Current License Condition 2.J Term of License This License is effective as of the date of issuance and shall expire at midnight on August 26, 2025.	Proposed License Condition 2.J Term of License This License is effective as of the date of issuance and ~~shall expire at midnight on August 26, 2025~~*is effective until the Commission notifies the licensee in writing that the license is terminated*.
Basis	
This license condition is revised to conform to 10 CFR 50.51, "Continuation of license," in that the license authorizes ownership and possession of the facility until the Commission notifies the licensee in writing that the license is terminated.	

Attachments	
Current List of Attachments Attachments: 1. Appendix A – Technical Specifications (NUREG-1151) 2. Appendix B – Environmental Protection Plan 3. Appendix C – Deleted 4. Appendix D – Additional Conditions	Proposed List of Attachments Attachments: 1. Appendix A – *Permanently Defueled* Technical Specifications ~~(NUREG-1151)~~ 2. Appendix B – Environmental Protection Plan 3. Appendix C – Deleted 4. Appendix D – ~~Additional Conditions~~ *Deleted*
Basis	
This list of attachments is proposed for revision to reflect the changes described in this LAR.	

2.2.4 DCPP Unit 2 Facility Operating License, Appendix D, Additional Conditions - Proposed Changes

Appendix D of FOL Number DPR-82 includes additional conditions with specific schedules associated with previous license amendments that PG&E was required to comply with. As described below Appendix D, is proposed for deletion in its entirety.

Amendment Number	Additional Condition	Implementation Date
118	The licensee is authorized to relocate certain technical specifications requirements to the equipment control guidelines (ECGs) as referenced in the Updated Final Safety Analysis Report. Implementation of these amendments shall include relocation of these technical specification requirements to the ECGs as described in the licensee's application dated October 4, 1995, as supplemented by letters dated July 17, 1996, August 20, 1996, and June 2, 1997, and evaluated in the staff's safety evaluation dated February 3, 1998.	The amendment shall be implemented within 90 days of its issuance.
Basis for Deletion		
In Reference 27, PG&E notified the NRC that the above license amendment had been implemented in accordance with the license condition. Therefore, this additional condition is historical and is proposed for deletion.		

Amendment Number	Additional Condition	Implementation Date
135	This amendment authorizes the relocation of certain Technical Specification requirements to licensee controlled documents. Implementation of this amendment shall include the relocation of these Technical Specification requirements to the appropriate documents, as described in Table LG of Details Relocated from Current Technical Specifications, Table R of Relocated Current Technical Specifications, Table LS of Less Restrictive Changes to Current Technical Specifications, and Table A of Administrative Changes to Current Technical Specifications that are attached	The amendment shall be implemented by June 30, 2000.

	to the NRC staff's Safety Evaluation enclosed with this amendment.	
Basis for Deletion		
In Reference 28, PG&E notified the NRC that the above license amendment had been implemented. Therefore, this additional condition is historical and is proposed for deletion.		

Amendment Number	Additional Condition	Implementation Date
135	The schedule for the performance of new and revised Surveillance Requirements (SR) shall be as follows: For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment. For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment. For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment. For SRs that existed prior to this amendment whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to implementation of this amendment.	The amendment shall be implemented by June 30, 2000.

Basis for Deletion
In Reference 28, PG&E notified the NRC that the above license amendment had been implemented. Therefore, this additional condition is historical and is proposed for deletion.

Amendment Number	Additional Condition	Implementation Date
202	Determination of CRE unfiltered air inleakage as required by surveillance requirement (SR) 3.7.10.5, in accordance with TS 5.5.19.c.(i). The assessment of CRE habitability as required by TS 5.5.19.c.(ii). The measurement of CRE pressure as required by TS 5.5.19.d. Following implementation, this condition will be performed as stated in the condition: The first performance of SR 3.7.10.5, in accordance with Specification 5.5.19.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful tracer gas test, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years. The first performance of the periodic assessment of CRE habitability, Specification 5.5.19.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful tracer gas test, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.	The amendment is effective as of the date of its issuance and the condition shall be implemented within 180 days of its issuance.

	The first performance of the periodic measurement of CRE pressure, Specification 5.5.19.d, shall be within 24 months, plus the 182 days allowed by SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful pressure measurement test, or within 182 days if not performed previously.	
	Basis for Deletion	
PG&E previously completed implementation of License Amendment 202 as required in the above additional condition. Therefore, this additional condition is no longer required and is proposed for deletion.		

Amendment Number	Additional Condition	Implementation Date
232	Implementation of the amendment adopting the alternative source term shall include the following plant modifications: Install shielding material, equivalent to that provided by the Control Room outer walls, at the external concrete west wall of the Control Room briefing room. Install a high efficiency particulate air filter in the Technical Support Center normal ventilation system. Re-classify a portion of the 40-inch Containment Penetration Area (GE/GW) Ventilation line from PG&E Design Class II to PG&E Design Class I and upgrade the damper actuators, pressure switches, and the damper solenoid valves to PG&E Design Class I. Update setpoints for the redundant safety related gamma sensitive area radiation monitors (1-RE 25/26, 2-RE 25/26).	The amendment is effective as of the date of its issuance and the condition shall be implemented within 365 days of its issuance.

Basis for Deletion
PG&E previously completed implementation of License Amendment 232 as required in the above additional condition. Therefore, this additional condition is no longer required and is proposed for deletion.

2.2.5 Technical Specifications

The following table provides a summary of the DCPP Units 1 and 2 TS that are being deleted in their entirety and which TS are being retained in the PDTS. The details and justification for the proposed changes follow in subsequent sections (arranged by TS Section).

TS Being Deleted	TS Being Retained
1.0 USE AND APPLICATION	
	1.1 Definition
	1.2 Logical Connectors
	1.3 Completion Times
	1.4 Frequency
2.0 SAFETY LIMITS (SLs)	
2.1 Safety Limits	
2.2 Safety Limit Violations	
3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY	
	LCO 3.0.1
	LCO 3.0.2
LCO 3.0.3	
LCO 3.0.4	
LCO 3.0.5	
LCO 3.0.6	
LCO 3.0.7	
LCO 3.0.8	
3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILTY	
	SR 3.0.1
	SR 3.0.2
	SR 3.0.3
	SR 3.0.4
3.1 REACTIVITY CONTROL SYSTEMS	
3.1.1 SHUTDOWN MARGIN (SDM)	
3.1.2 Core Reactivity	
3.1.3 Moderator Temperature Coefficient (MTC)	
3.1.4 Rod Group Alignment Limits	
3.1.5 Shutdown Bank Insertion Limits	

TS Being Deleted	TS Being Retained
3.1.6 Control Bank Insertion Limits	
3.1.7 Rod Position Indication	
3.1.8 PHYSICS TESTS Exceptions MODE 2	
3.2 POWER DISTRIBUTION LIMITS	
3.2.1 Heat Flux Hot Channel Factor (Fq(Z))	
3.2.2 Nuclear Enthalpy Rise Hot Channel factor $F^N_{\Delta H}$	
3.2.3 AXIAL FLUX DIFFERENCE (AFD)	
3.2.4 QUADRANT POWER TILT RATIO (QPTR)	
3.3 INSTRUMENTATION	
3.3.1 Reactor Trip System (RTS) Instrumentation	
3.3.2 Engineering Safety Feature Actuation System (ESFAS) Instrumentation	
3.3.3 Post Accident Monitoring (PAM) Instrumentation	
3.3.4 Remote Shutdown System	
3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation	
3.3.6 Containment Ventilation Isolation Instrumentation	
3.3.7 Control Room Ventilation System (CRVS) Actuation Instrumentation	
3.3.8 Fuel Building Ventilation System (FBVS) Actuation Instrumentation	
3.4 REACTOR COOLANT SYSTEM (RCS)	
3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	
3.4.2 RCS Minimum Temperature for Criticality	
3.4.3 RCS Pressure and Temperature (P/T) Limits	

TS Being Deleted	TS Being Retained
3.4.4 RCS Loops – MODES 1 and 2	
3.4.5 RCS Loops – MODE 3	
3.4.6 RCS Loops MODE 4	
3.4.7 RCS Loops – MODE 5 Loops Filled	
3.4.8 RCS Loops – MODE 5 Loops Not Filled	
3.4.9 Pressurizer	
3.4.10 Pressurizer Safety Valves	
3.4.11 Pressurizer Power Operated Relief Valves (PORVs)	
3.4.12 Low Temperature Overpressure Protection (LTOP) System	
3.4.13 RCS Operations LEAKAGE	
3.4.14 RCS Pressure Isolation Valve (PIV) Leakage	
3.4.15 RCS Leakage Detection Instrumentation	
3.4.16 RCS Specific Activity	
3.4.17 Steam Generator (SG) Tube Integrity	
3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)	
3.5.1 Accumulators	
3.5.2 ECCS – Operating	
3.5.3 ECCS – Shutdown	
3.5.4 Refueling Water Storage Tank (RWST)	
3.5.5 Seal Injection Flow	
3.6 CONTAINMENT SYSTEMS	
3.6.1 Containment	
3.6.2 Containment Air Locks	
3.6.3 Containment Isolation Valves	
3.6.4 Containment Pressure	
3.6.5 Containment Air Temperature	
3.6.6 Containment Spray System and Cooling Systems	
3.6.7 Spray Additive System	
3.6.8 Deleted	

TS Being Deleted	TS Being Retained
3.7 PLANT SYSTEMS	
3.7.1 Main Steam Safety Valves (MSSVs)	
3.7.2 Main Steam Isolation Valves (MSIVs)	
3.7.3 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs), MFRV Bypass Valves, and Main Feedwater Pump (MFWP) Turbine Stop Valves	
3.7.4 10% Atmospheric Dump Valves (ADVs)	
3.7.5 Auxiliary Feedwater (AFW) System	
3.7.6 Condensate Storage Tank (CST)	
3.7.7 Vital Component Cooling Water (CCW) System	
3.7.8 Auxiliary Saltwater System (ASW)	
3.7.9 Ultimate Heat Sink (UHS)	
3.7.10 Control Room Ventilation (CRVS)	
3.7.11 Control Room Emergency Air Temperature Control Systems (CREATCS) – Not Used	
3.7.12 Auxiliary Building Ventilation System (ABVS)	
3.7.13 Fuel Handling Building Ventilation System (FHBVS)	
3.7.14 Penetration Room Exhaust Air Cleanup System (PREACS) – Not Used	
	3.7.15 Spent Fuel Pool Water Level
	3.7.16 Spent Fuel Pool Boron Concentration
	3.7.17 Spent Fuel Assembly Storage
3.7.18 Secondary Specific Activity	

TS Being Deleted	TS Being Retained
3.8 ELECTRICAL POWER SYSTEMS	
3.8.1 AC Sources – Operating	
3.8.2 AC Sources – Shutdown	
3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air Turbocharger Air Assist	
3.8.4 DC Sources - Operating	
3.8.5 DC Sources – Shutdown	
3.8.6 Battery Cell Parameters	
3.8.7 Inverters - Operating	
3.8.8 Inverters - Shutdown	
3.8.9 Distribution Systems – Operating	
3.8.10 Distribution Systems – Shutdown	
3.9 REFUELING OPERATIONS	
3.9.1 Boron Concentration	
3.9.3 Nuclear Instrumentation	
3.9.4 Containment Penetrations	
3.9.5 Residual Heat Removal (RHR) and Coolant Circulation – High Water Level	
3.9.6 Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level	
3.9.7 Refueling Cavity Water Level	
4.0 DESIGN FEATURES	
	4.1 Site Location
4.2 Reactor Core	
	4.3 Fuel Storage
5.0 ADMINISTRATIVE CONTROLS	
	5.1 Responsibility
	5.2 Organization
	5.3 Unit Staff Qualifications
	5.4 Procedures
	5.5 Programs and Manuals
	5.6 Reporting Requirements
	5.7 High Radiation Area

The TS Table of Contents are being revised accordingly.

The corresponding TS Bases are also being either deleted or revised (as applicable) to reflect the changes described below.

<table>
<tr><th colspan="2">SECTION 1.1 - DEFINITIONS</th></tr>
<tr><td colspan="2">TS Section 1.1 "Definitions," contains defined terms that are applicable to an operating plant throughout the TS and TS Bases. Once PG&E submits the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). The revisions to the definitions identified below align with a permanently shut down and defueled condition. Many of the definitions have been proposed for deletion since they are relevant to an operating reactor and are not used in the PDTS. The standard convention of indicating the defined term in ALL CAPITAL LETTERS throughout the TS has been adopted in the PDTS.</td></tr>
<tr><th>Proposed Definitions - Added</th><th>Basis for Addition</th></tr>
<tr><td><u>CERTIFIED FUEL HANDLER</u>

A CERTIFIED FUEL HANDLER is an individual who complies with the provisions of the CERTIFIED FUEL HANDLER Training and Retraining Program required by Specification 5.3.2.</td><td>PG&E proposes to add a definition for Certified Fuel Handler. This ensures that these positions are consistently utilized throughout the TS. These terms are used in the proposed TS Section 5.0, “Administrative Controls.” PG&E will implement a Certified Fuel Handler Training and Retraining Program prior to implementing the PDTS.</td></tr>
<tr><td><u>NON-CERTIFIED OPERATOR</u>

A NON-CERTIFIED OPERATOR is an operator who complies with the qualification requirements of Specification 5.3.1, but is not a CERTIFIED FUEL HANDLER.</td><td>PG&E proposes to add a definition for Non-Certified Operator. This ensures that these positions are consistently utilized throughout the TS. These terms are used in the proposed TS Section 5.0, “Administrative Controls.” As described above, PG&E will implement a Certified Fuel Handler Training and Retraining Program prior to implementing the PDTS.</td></tr>
<tr><th>Proposed Definitions - Deleted</th><th>Basis for Deletion</th></tr>
<tr><td>ACTUATION LOGIC TEST

An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.</td><td>The actuation logic test is not applicable in the permanently defueled condition and is not used in any PDTS. Therefore, the definition is proposed for deletion.</td></tr>
</table>

AXIAL FLUX DIFFERENCE (AFD) AFD shall be the difference in normalized flux signals between the top and bottom halves of an excore neutron detector.	This term is not applicable to a permanently defueled reactor and is not used in any PDTS. Therefore, the definition is proposed for deletion.
CHANNEL CALIBRATION A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY. Calibration of instrument channels with resistance temperature detectors (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping or total channel steps.	Channel calibrations are not applicable in the permanently defueled condition and the term is not used in any PDTS. Therefore, the definition is proposed for deletion.
CHANNEL CHECK A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.	Channel checks are not applicable in the permanently defueled condition and the term is not used in any PDTS. Therefore, the definition is proposed for deletion.
CHANNEL FUNCTIONAL TEST (CFT) A CFT shall be: a. Analog channels - the injection of a simulated or actual signal into the channel as close to the sensor as practical to verify OPERABILITY of all devices in the channel required for	The channel functional test is not applicable in the permanently defueled condition and the term is not used in any PDTS. Therefore, the definition is proposed for deletion.

channel OPERABILITY, or b. Bistable channels - the injection of a simulated or actual signal into the sensor to verify OPERABILITY of all devices in the channel required for channel OPERABILITY, or c. Digital channels - the injection of a simulated or actual signal into the channel as close to the sensor input to the process racks as practical to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CFT may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.	
CHANNEL OPERATIONAL TEST (COT) A COT shall be: a. Analog, bistable, and Eagle 21 process protection system digital channels - the injection of a simulated or actual signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. b. Tricon/Advanced Logic System process protection system digital channels - the use of diagnostic programs to test digital hardware, manual verification that the setpoints and tunable parameters are correct, and the injection of simulated process data into the channel as close to the sensor input to the process racks as practical to verify channel OPERABILITY of all devices in the channel required for OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm,	The channel operational test is not applicable in the permanently defueled condition and the term is not used in any PDTS. Therefore, the definition is proposed for deletion.

interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping or total channel steps.	
CORE ALTERATION CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.	The term core alteration is not applicable in the permanently defueled condition and the term is not used in any PDTS. Therefore, the definition is proposed for deletion.
CORE OPERATING LIMITS REPORT (COLR) The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.	The COLR is not applicable in the permanently defueled condition and the term is not used in any PDTS. TS 5.6.5 is also proposed for deletion. Therefore, this definition is proposed for deletion.
DOSE EQUIVALENT I-131 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using the committed thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, “Limiting Values of Radionuclide Intake and Air Concentration and Dose	This term is used in the current TS to express the DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133, specific activity limits for the RCS. These specific activity limits will no longer apply after DCPP Units 1 and 2 are permanently defueled, and this term is not used in any PDTS. Therefore, this definition is proposed for deletion.

Conversion Factors for Inhalation, Submersion, and Ingestion."	
DOSE EQUIVALENT XE-133 DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."	This term is used in the current TS to express the DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133, specific activity limits for the RCS. These specific activity limits will no longer apply after DCPP Units 1 and 2 are permanently defueled, and this term is not used in any PDTS. Therefore, this definition is proposed for deletion.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.	In the permanently defueled condition, there are no ESF systems that are credited in the analysis of the FHA in the FHB (remaining credible accident) and the term is not used in any PDTS. Therefore, this definition is proposed for deletion.

LEAKAGE LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE). b. <u>Unidentified LEAKAGE</u> All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE. c. <u>Pressure Boundary LEAKAGE</u> LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.	This definition refers to leakage from the primary system of an operating plant. It is not applicable during decommissioning and the term is not used in any PDTS. Therefore, this definition is proposed for deletion.
MASTER RELAY TEST A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The	The master relay test is applicable to the surveillances performed in TS Section 3.3, “Instrumentation,” which is proposed for deletion. This term is not applicable in the permanently defueled condition and is not used in any PDTS. Therefore, this definition is proposed for deletion.

MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.	
MODE A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.	These modes are defined for operating and refueling conditions and do not apply to a facility in the permanently defueled condition. This term is not used in any PDTS. Therefore, this definition is proposed for deletion.
OPERABLE-OPERABILITY A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).	In the permanently defueled condition, there are no systems, subsystems, trains, components, or devices included in the PDTS that are required to perform a specified safety function and the term is not used in any PDTS. Therefore, this definition is proposed for deletion.
PHYSICS TESTS PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14 of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.	Physics tests are only applicable to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition and is not used in any PDTS. Therefore, this definition is proposed for deletion.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup	This report does not apply to a facility in the permanently defueled condition and is not used in any PDTS. TS 5.6.6 is also proposed for deletion from the PDTS. Therefore, this definition is proposed for deletion.

and cooldown rates, and the power operated relief valve (PORV) lift settings and arming temperature associated with the Low Temperature Overpressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6.	
QUADRANT POWER TILT RATIO (QPTR) QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.	This term is only applicable to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition and is not used in any PDTS. Therefore, this definition is proposed for deletion.
RATED THERMAL POWER (RTP) RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt for each unit.	This term is only applicable to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition and is not used in any PDTS. Therefore, this definition is proposed for deletion.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.	The RTS is only applicable to a reactor authorized to operate at power. It does not apply to facility in the permanently defueled condition and is not used in any PDTS. Therefore, this definition is proposed for deletion.

SHUTDOWN MARGIN (SDM) SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.	This term is only applicable to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition and is not used in any PDTS. Therefore, this definition is proposed for deletion.
SLAVE RELAY TEST A SLAVE RELAY TEST shall consist of energizing all slave relays and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.	The slave relay test is not applicable in the permanently defueled condition and the term is not used in any PDTS. Therefore, the definition is proposed for deletion.
THERMAL POWER THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.	This term is only applicable to a reactor authorized to contain fuel and operate at power. It does not apply to a facility in the permanently defueled condition and is not used in any PDTS. Therefore, this definition is proposed for deletion.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the	The trip functions associated with TADOT are not credited in the analysis of the FHA in the FHB (remaining credible accident). This term is not applicable in the permanently defueled condition and is not used in any PDTS. Therefore, this definition is proposed for deletion.

required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping or total channel steps.	
Table 1.1-1, MODES	The modes defined in Table 1.1-1 are defined for operating or refueling conditions. The modes do not apply to a facility in a permanently defueled condition and the term is not used in any PDTS. Therefore, this table is proposed for deletion.

TS SECTION 1.3 – COMPLETION TIMES

The purpose of TS Section 1.3 is to establish the Completion Time convention and to provide guidance for its use.

Current Subsection	Proposed Subsection
BACKGROUND Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).	BACKGROUND Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe ~~operation of the unit~~ <u>*handling and storage of nuclear fuel*</u>. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply	DESCRIPTION The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., ~~inoperable equipment or~~ variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the ~~unit~~ <u>*facility*</u> is in a ~~MODE or~~ specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the

until the Condition no longer exists or the unit is not within the LCO Applicability. If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition. Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition However, when a <u>subsequent</u> train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability: a. Must exist concurrent with the <u>first</u> inoperability; and b. Must remain inoperable or not within limits after the first inoperability is resolved. The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either: a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24	~~unit~~ <u>*facility*</u> is not within the LCO Applicability. ~~If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.~~ ~~Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition~~ ~~However, when a <u>subsequent</u> train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:~~ ~~a. Must exist concurrent with the <u>first</u> inoperability; and~~ ~~b. Must remain inoperable or not within limits after the first inoperability is resolved.~~ ~~The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:~~ ~~a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24~~

hours; or b. The stated Completion Time as measured from discovery of the subsequent inoperability. The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications. The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ."	~~hours; or~~ ~~b. The stated Completion Time as measured from discovery of the subsequent inoperability.~~ ~~The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.~~ ~~The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ."~~
Basis	
BACKGROUND – This subsection is proposed for revision to remove reference to "operation of the unit" and replace it with reference to "handling and storage of nuclear fuel." This change is administrative and more appropriately represents the permanently shutdown and defueled condition. DESCRIPTION – This subsection is proposed for revision to remove reference to inoperable equipment because the PDTS do not have LCOs for equipment to be operable or in operation. In addition, the proposed revisions replace "unit" with "facility" to more appropriately represent the permanently shutdown and defueled condition. The discussion of Modes is removed because the term is not applicable to a permanently defueled facility. Discussion is also removed for entries into more than one condition, or alternating between conditions, as each of the three remaining PDTS that contain conditions only has a single condition. The discussion related to Completion Time extensions is no longer applicable and is removed because the remaining TS have a Completion Time of "Immediately."	
Current Subsection	**Basis for Deletion**
Examples 1.3-1, 1.3-2, 1.3-3, 1.3-4, 1.3-5, 1.3-6, and 1.3-7	The purpose of the examples included in Section 1.3 is to illustrate the use of Completion Times with different types of Conditions and changing conditions.

	These examples are no longer necessary because they describe examples that do not remain in the PDTS. Therefore, the examples are proposed for deletion.

TS SECTION 1.4 – FREQUENCY The purpose of TS Section 1.4 is to define the proper use and application of Frequency requirements.	
Current Subsection	**Proposed Subsection**
EXAMPLES The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.	EXAMPLES The following examples ~~illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.~~ *illustrate the type of frequency statements that appear in the Permanently Defueled Technical Specifications (PDTS).*
EXAMPLE 1.4-1 SURVEILLANCE REQUIREMENTS SURVEILLANCE | FREQUENCY Perform CHANNEL CHECK. | 12 hours	EXAMPLE 1.4-1 SURVEILLANCE REQUIREMENTS SURVEILLANCE | FREQUENCY ~~Perform CHANNEL CHECK~~*Verify level is within limits*. | 12 hours
Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a	Example 1.4-1 contains the type of SR most often encountered in the ~~Technical Specifications (TS)~~*PDTS*. The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for ~~operational~~ flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when ~~the equipment is~~

<table>
<tr>
<td>variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.</td>
<td>~~inoperable,~~ a variable is outside specified limits, or the ~~unit~~ *facility* is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the ~~unit~~ *facility* is in a ~~MODE or other~~ specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified ~~(refer to Example 1.4-3)~~, then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the ~~unit~~ *facility* is not in a ~~MODE or other~~ specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the ~~MODE or other~~ specified condition. Failure to do so would result in a violation of SR 3.0.4.</td>
</tr>
<tr>
<td>EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS
<table><tr><th>SURVEILLANCE</th><th>FREQUENCY</th></tr><tr><td>Verify flow is within limits.</td><td>Once within 12 hours after ≥ 25% RTP

AND

24 hours thereafter</td></tr></table>
Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other</td>
<td>EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS
<table><tr><th>SURVEILLANCE</th><th>FREQUENCY</th></tr><tr><td>Verify ~~flow is within limits~~…</td><td>~~Once within 12 hours after ≥ 25% RTP~~

~~AND~~

~~24 hours thereafter~~
Prior to each fuel assembly move…</td></tr></table>
~~Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.~~

~~The use of "once" indicates a single~~</td>
</tr>
</table>

Frequencies are connected by "<u>AND</u>"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.	~~performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.~~ <u>*Example 1.4-2 illustrates a one time performance Frequency.*</u> <u>*This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2.*</u>

Basis

EXAMPLE – This subsection is proposed for revision to remove reference to Modes because the term is not applicable to a permanently defueled facility. In addition, the introduction to the examples is updated to reference the PDTS.

EXAMPLE 1.4-1 – The example surveillance is updated to align with the surveillances in the PDTS. The discussion associated with this example is proposed for revision to replace reference to the TS with reference to the PDTS. In addition, other terms such as "Mode," "operational" and "unit" are either being deleted or updated to a term more appropriate for a permanently shutdown and defueled facility. Reference to equipment being inoperable is proposed for deletion because there is no equipment required to be operable in the PDTS.

Reference to Example 1.4-3 is also deleted because the Example is proposed for deletion as described below.

EXAMPLE 1.4-2 – This example is revised to be applicable to the permanent defueled condition and the use of a one time performance Frequency in the proposed TS.

Current Subsection	**Basis for Deletion**
Example 1.4-3.	This example is proposed for deletion because it is not applicable in the PDTS.

TS SECTION 2.0 – SAFETY LIMITS (SLs)

TS Section 2.0, "Safety Limits," contains SLs to establish limits on important process variables to assure the integrity of the fuel cladding and the RCS in all Modes of operation.

Pursuant to 10 CFR 50.36(c)(1), SLs are limiting parameters necessary to protect the physical barriers that guard against uncontrolled release of radioactivity from a nuclear reactor. The SLs established in TS 2.1.1 and 2.1.2 protect the integrity of the fuel cladding and RCS barriers, respectively. TS 2.2.1 and 2.2.2 describe actions to take if TS 2.1.1 or 2.1.2 is violated.

This section is proposed for deletion in its entirety, because the SLs do not apply to a reactor that is in a permanently defueled condition. Once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). These specifications do not apply to the safe storage and handling of spent fuel in the SFPs.

Current DCPP TS	Basis for Deletion
TS 2.1.1, Reactor Core SLs	TS 2.1.1 provides SLs that prevent overheating of the fuel and cladding, as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. This specification is applicable in MODES 1 and 2. Pursuant to 10 CFR 50.82(a)(2), once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), the licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels. Since the SLs apply to an operating reactor, they have no function in the permanently defueled condition. Therefore, this SL is proposed for deletion.
TS 2.1.2, RCS Pressure SL	TS 2.1.2 provides SLs that prevent potential damage to the RCPB from overpressure that could result in the uncontrolled release of fission products to the containment atmosphere. This specification is applicable in MODES 1, 2, 3, 4 and 5. Pursuant to 10 CFR 50.82(a)(2), once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), the licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels. Since the SLs apply to an operating reactor, they have no function in the permanently defueled condition. Therefore, this SL is proposed for deletion.

TS 2.2, SL Violations	TS 2.2.1 and 2.2.2 describe actions to take if TS 2.1.1 or 2.1.2 is violated. Because TS 2.1.1 and 2.1.2 are being proposed for deletion, TS 2.2.1 and 2.2.2 are no longer required and are proposed for deletion.
Figure 2.1.1-1, Reactor Core Safety Limit	Figure 2.1.1-1 provides the acceptable values for thermal power, RCS highest loop average temperature, and pressurizer pressure to meet the requirements of TS 2.1.1. The limits provided in Figure 2.1.1-1 prevent overheating of the fuel and cladding, as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. These limits are applicable is MODES 1 and 2. Pursuant to 10 CFR 50.82(a)(2), once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), the licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels. Since the SLs apply to an operating reactor, they have no function in the permanently defueled condition. Therefore, this figure is proposed for deletion.

TS SECTION 3 – LIMITING CONDITION FOR OPERATION

TS Section 3 of the current DCPP TS contain the LCO. In accordance with 10 CFR 50.36(c)(2), LCO specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The LCO typically place restrictions on availability of safety equipment needed to prevent or mitigate a postulated DBA, or on process variables necessary to preserve the initial conditions assumed in the safety analyses of postulated DBAs. 10 CFR 50.36(c)(2)(ii) defines four criteria for establishing LCO. Associated SRs help to ensure that specified equipment and parameters are maintained within the limits specified in the LCO.

As discussed previously, with the reactor in a permanently defueled state, the only postulated DBA that remains applicable is an FHA in the FHB. As a result, most of the LCO and accompanying SR contained in the DCPP Units 1 and 2, TS are proposed for deletion.

Each subsection of TS Section 3 is discussed in more detail in the tables below.

TS SECTION 3.0 – LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

TS Section 3.0, "Limiting Conditions for Operation (LCO) Applicability," contains the general requirements applicable to all LCO and applies at all times unless otherwise

stated in TS. Due to the limited number of LCO in the proposed PDTS, a number of the DCPP TS provisions in this section are no longer necessary or applicable to DCPP Units 1 and 2 as described below.	
Current DCPP TS	**Proposed TS**
LCO 3.0.1 LCO shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.8.	LCO 3.0.1 LCO shall be met during the ~~MODES or other~~ specified conditions in the Applicability, except as provided in LCO 3.0.2~~, LCO 3.0.7, and LCO 3.0.8~~.
LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.	LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met~~, except as provided in LCO 3.0.5 and LCO 3.0.6~~. ~~If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Require Action(s) is not required, unless otherwise stated~~.
Basis	
LCO 3.0.1 – The term "MODES" is removed because it is not applicable in a permanently defueled condition. TS 3.0.7 and 3.0.8 are being deleted from LCO 3.0.1 because they are proposed for deletion as described below. LCO 3.0.2 – The statement "…except as provided in LCO 3.0.5 and LCO 3.0.6." will be removed. LCO 3.0.5 and LCO 3.0.6 are proposed for deletion as part of the PDTS. In addition, the discussion related to meeting the LCO prior to the expiration of the specified completion time, will be deleted. The only remaining completion time in the PDTS is "immediately" and therefore this discussion is no longer applicable.	

Current DCPP TS	Basis for Deletion
LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 3 within 7 hours;	LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met. LCO 3.0.3 is only applicable in Modes 1, 2, 3, and 4. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), LCO 3.0.3 is no longer applicable. Therefore, LCO 3.0.3 is proposed for deletion.

Current DCPP TS	Basis for Deletion
b. MODE 4 within 13 hours; and c. MODE 5 within 37 hours. Exceptions to this Specification are stated in the individual Specifications. Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.	
<u>LCO 3.0.4</u> When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made: a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or c. When an allowance is stated in the individual value, parameter, or other Specification. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that	LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c. Pursuant to 10 CFR 50.82(a)(2), upon docketing the certifications required by 10 CFR 50.82(a)(1), the facility licenses for DCPP Units 1 and 2, will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels. References to operating MODES are no longer relevant. Therefore, LCO 3.0.4 is no longer applicable in the permanently defueled condition and is proposed for deletion.

Current DCPP TS	Basis for Deletion
are part of a shutdown of the unit.	
LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.	LCO 3.0.5 allows equipment removed from service or declared inoperable to be returned to service for the purpose of testing. The remaining PDTS ACTIONS do not include requirements to declare equipment inoperable or to remove it from service. Therefore LCO 3.0.5 is proposed for deletion.
LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.15, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.	LCO 3.0.6 addresses the actions required for a supported system when the support system LCO is not met. The PDTSs do not include any LCOs for equipment to be operable or in operation in the PDTS. In addition, TS 5.5.15 is proposed for deletion as described below. Therefore, LCO 3.0.6 is no longer applicable and is proposed for deletion.
LCO 3.0.7 Test Exception LCO 3.1.8, allows	LCO 3.0.7 is associated with test exceptions in LCO 3.1.8, which is proposed for deletion as described below.

Current DCPP TS	Basis for Deletion
specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCO is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.	Special tests and operations are not applicable to a permanently defueled facility. Therefore, LCO 3.0.7 is not required and is proposed for deletion.
<u>LCO 3.0.8</u> When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and: a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours. At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported	LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), all TS systems associated with snubbers are no longer required to be operable. Therefore, the allowance provided by LCO 3.0.8 is no longer needed and is proposed for deletion.

Current DCPP TS	Basis for Deletion
system LCO(s) shall be declared not met.	

TS SECTION 3.0 – SURVEILLANCE REQUIREMENT (SR) APPLICABILITY TS Section 3.0, "Surveillance Requirement (SR) Applicability," establishes the general requirements applicable to all TS and apply at all times, unless otherwise stated.	
Current DCPP TS	**Proposed TS**
SR 3.0.1 SR shall be met during the MODES or other specified conditions in the Applicability for individual LCO, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.	SR 3.0.1 SR shall be met during the ~~MODES or other~~ specified conditions in the Applicability for individual LCO, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on ~~inoperable equipment or~~ variables outside specified limits.
SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per ..." basis, the above Frequency extension applies to each performance after the initial performance.	SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. ~~For Frequencies specified as "once," the above interval extension does not apply.~~ ~~If a Completion Time requires periodic performance on a "once per ..." basis, the above Frequency extension applies to each performance after the initial performance.~~

Exceptions to this Specification are stated in the individual Specifications.	~~Exceptions to this Specification are stated in the individual Specifications.~~
<u>SR 3.0.4</u> Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.	<u>SR 3.0.4</u> Entry into a ~~MODE or other~~ specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. ~~When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.~~ This provision shall not prevent entry into ~~MODES or other~~ specified conditions in the Applicability that are required to comply with ACTIONS ~~or that are part of a shutdown of the unit~~.

Basis

SR 3.0.1 – The term "MODES" will be removed because it is not applicable in a permanently defueled condition. In the PDTS, there are no LCOs for equipment to be operable or in operation and therefore references to inoperable equipment are proposed for deletion.

SR 3.0.2 – SR 3.0.2 is proposed for revision to remove conditions for frequencies that do not exist in PDTS LCOs. The discussion related to Completion Times that require periodic performance is proposed for deletion because periodic Completion Times are not included in the PDTS. Reference to exceptions to this SR is also proposed for deletion because the PDTS do not include any exceptions.

SR 3.0.4 – The term "MODES" and reference to shutdown of the unit will be removed because it is not applicable in a permanently defueled condition. In addition, the discussion pertaining to LCO 3.0.4 is proposed for deletion because LCO 3.0.4 is proposed for deletion as described above.

<u>Summary:</u>
Once PG&E dockets the certifications for DCPP Units 1 and 2, required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore, the proposed revisions to SR 3.0.1, 3.0.2, and 3.0.4 are acceptable and will not impact continued safe maintenance of the facility.

TS SECTION 3.1 – REACTIVITY CONTROL TS Section 3.1 "Reactivity Control," contains LCO related to reactivity control capability and applies to core reactivity and the reactivity control systems to protect the integrity of the fission product barrier. The table below describes the specifications in this section. This section is proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFPs. Once PG&E dockets the certifications for DCPP Units 1 and 2, required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore, the reactivity control functions addressed in TS Section 3.1 will not be required and these LCO (and associated SR) will not apply in a permanently defueled condition.	
Current DCPP TS	**Basis for Deletion**
TS 3.1.1, SHUTDOWN MARGIN (SDM)	This specification requires the SDM to be within the limits provided in the COLR, and is applicable in Mode 2 with k_{eff} less than 1.0, MODES 3, 4, and 5. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and AOOs. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all shutdown and control rods assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), reactivity limitations associated with fuel in the reactor will no longer be applicable. Therefore, TS 3.1.1 is proposed for deletion.
TS 3.1.2, Core Reactivity	This specification requires the measured core reactivity to be within ± 1% Δ k/k of predicted values, and is applicable in MODES 1 and 2. The reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. This periodic confirmation of core reactivity is necessary to ensure that DBA and transient safety analyses remain valid. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor

	vessels, pursuant to 10 CFR 50.82(a)(2), confirmation of core reactivity will no longer be applicable. Therefore, TS 3.1.2 is proposed for deletion.
TS 3.1.3, Moderator Temperature Coefficient (MTC)	This specification requires the MTC to be maintained within the limits specified in the COLR. In addition, the maximum upper limit is specified in Figure 3.1.3-1. This specification is applicable in MODE 1 and MODE 2 with k_{eff} greater than or equal to 1.0 for the upper MTC limit, MODES 1, 2, and 3 for the lower MTC limit. The MTC relates a change in core reactivity to a change in reactor coolant temperature. The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self-limiting, and stable power operation will result. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), requirements for maintaining the MTC within applicable limits in MODES 1, 2 and 3, will no longer be applicable. Therefore, TS 3.1.3 (including Figure 3.1.3-1) is proposed for deletion.
TS 3.1.4, Rod Group Alignments	This specification requires all shutdown and control rods be operable and individual indicated rod positions to be within 12 steps of their group step counter demand position. This specification is applicable during MODES 1 and 2. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and operability are related to core operation in design power peaking limits and the core design requirement of a minimum SDM. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), requirements for control rod alignment and operability which are related to core operation will no longer be applicable. Therefore, TS 3.1.4 is proposed for deletion.

TS 3.1.5, Shutdown Bank Insertion Limits	This specification requires each shutdown bank to be within insertion limits specified in the COLR and is not applicable to shutdown banks inserted while performing SR 3.1.4.2. This specification is applicable during MODE 1, and MODE 2 with any control bank not fully inserted. The shutdown banks affect core power and burnup distribution and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal. The insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shutdown the reactor and maintain the required SDM following a reactor trip from full power. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), insertion limits associated with the shutdown banks will no longer be applicable. Therefore, TS 3.1.5 is proposed for deletion.
TS 3.1.6, Control Bank Insertion Limits	This specification requires control banks to be within insertion, sequence, and overlap limits specified in the COLR and is not applicable to control banks inserted while performing SR 3.1.4.2. This specification is applicable during MODE 1, and MODE 2 with k_{eff} greater than or equal to 1.0. The control banks are used for precise reactivity control of the reactor. They are capable of adding reactivity very quickly. These limits prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by RTS trip function. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), insertion limits associated with the control banks will no longer be applicable. Therefore, TS 3.1.6 is proposed for deletion.
TS 3.1.7, Rod Position Indication	This specification requires the digital rod position indication (DRPI) System and the demand position indication system to be operable and is applicable during MODES 1 and 2. Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a DBA, with control or shutdown rods operating outside their limits undetected. The

	axial position of shutdown rods and control rods are determined by two separate and independent systems: the bank demand position indication system and the DRPI system. Operability of the rod position indicators is required to determine rod positions and thereby ensure compliance with the rod alignment and insertion limits. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), rod position indication will no longer be applicable. Therefore, TS 3.1.7 is proposed for deletion.
TS 3.1.8, PHYSICS TEST Exceptions – MODE 2	This specification allows for suspension of other LCO during MODE 2 physics tests provided the following: a. RCS lowest operating loop average temperature is ≥ 531° F; b. SDM is within the limits provided in the COLR; and c. THERMAL P0WER is less than or equal to 5 percent RTP. The primary purpose of the Mode 2 Physics Tests exceptions is to permit relaxations of existing LCO to allow certain Physics Tests to be performed. The Physics Tests requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), Physics Testing will no longer be required. Therefore, TS 3.1.8 is proposed for deletion.

TS SECTION 3.2 – POWER DISTRIBUTION LIMITS

TS Section 3.2 "Power Distribution Limits," contains LCO to ensure that power distribution limits are met. These LCO will not apply to a reactor that is in a permanently defueled condition. The table below describes the specifications included in this section.

This section is proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFPs. Once PG&E dockets the certifications for DCPP Units 1 and 2, required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore,

the specifications addressed in TS Section 3.2 will not be required and these LCO (and associated SR) will not apply in a permanently defueled condition.	
Current DCPP TS	**Basis for Deletion**
TS 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$)	This specification requires the Heat Flux Hot Channel Factor, to be within the limits specified within the COLR, and is applicable in MODE 1. The Heat Flux Hot Channel Factor is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, it is a measure of the peak fuel pellet power within the reactor core. The purpose of the limits on the Heat Flux Hot Channel Factor is to limit the local (i.e., pellet) peak power density. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), limiting the peak fuel pellet power within the reactor core will no longer be applicable. Therefore, TS 3.2.1 is proposed for deletion.
TS 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor	This specification requires the Nuclear Enthalpy Rise Hot Channel Factor to be within the limits specified in the COLR and is applicable during MODE 1. This LCO limits the power density at any point in the core. The design limits on local and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during normal operation, operational transients, and any transient condition arising from events of moderate frequency analyzed in the safety analyses. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), limits on power density in the core will no longer be applicable. Therefore, TS 3.2.2 is proposed for deletion.
TS 3.2.3, Axial Flux Difference (AFD)	This specification requires the AFD in percent flux difference units to be maintained within the limits specified in the COLR, and is applicable in MODE 1 with thermal power greater than or equal to 50 percent RTP.

	The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), limits on the AFD will no longer be applicable. Therefore, TS 3.2.3 is proposed for deletion.
TS 3.2.4, Quadrant Power Tilt Ratio (QPTR)	This specification requires the QPTR to be less than or equal to 1.02, and is applicable in MODE 1 with thermal power greater than 50 percent RTP. The QPTR limit ensures that the gross radial power distribution within the core remains consistent with the design values used in the safety analyses. Since PG&E will no longer be authorized to operate the DCPP Units 1 and 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), limits on the power distribution within the core will no longer be applicable. Therefore, TS 3.2.4 is proposed for deletion.

TS SECTION 3.3 – INSTRUMENTATION	
TS Section 3.3 "Instrumentation," contains LCO that provide for appropriate functional capability sensing and controlling instrumentation required for safe operation of the facility. As described further in the table below, the LCO will no longer apply in the permanently defueled condition when PG&E implements the PDTS. This section is proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFPs. Once PG&E dockets the certifications for DCPP Units 1 and 2, required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS Section 3.3 will not be required and these LCO (and associated SR) will not apply in a permanently defueled condition.	
TS 3.3.1, Reactor Trip System (RTS) Instrumentation	This specification requires RTS instrumentation for certain functions to be operable and is applicable in MODES 1, 2, 3, 4,

	and 5 (according to specific applicability requirements for each RTS function listed in TS Table 3.3.1-1). The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and RCS pressure boundary during AOOs and to assist the ESF systems in mitigating accidents. The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCO on other reactor system parameters and equipment performance. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability of the RTS instrumentation in MODES 1, 2, 3, 4, and 5 will no longer be applicable. Therefore, TS 3.3.1 is proposed for deletion.
TS 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation	This specification requires ESFAS instrumentation for certain functions to be operable and is applicable in MODES 1, 2, 3, and 4 (according to specific applicability requirements for each RTS function listed in TS Table 3.3.2-1). The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the RCS pressure boundary, and to mitigate accidents. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability of the ESFAS instrumentation in MODES 1, 2, 3, and 4, will no longer be applicable. Therefore, TS 3.3.2 is proposed for deletion.
TS 3.3.3, Post Accident Monitoring (PAM) Instrumentation	This specification requires the PAM instrumentation for certain functions to be operable and is applicable in MODES 1, 2, and 3. The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the CR operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are

	required for safety systems to accomplish their safety functions for DBAs. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability of the PAM instrumentation in MODES 1, 2, and 3 will no longer be applicable. Therefore, TS 3.3.3 is proposed for deletion.
TS 3.3.4, Remote Shutdown System	This specification requires certain remote shutdown system functions to be operable and is applicable in Modes 1, 2, and 3. The remote shutdown system provides the CR operator with sufficient I&Cs to place and maintain the unit in a safe shutdown condition from a location other than the CR. This capability is necessary to protect against the possibility that the CR becomes inaccessible. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability of the remote shutdown system functions in MODES 1, 2, and 3 will no longer be applicable. Therefore, TS 3.3.4 is proposed for deletion.
TS 3.3.5, Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation	This specification requires one channel per bus of loss of voltage DG start function; and two channels per bus of degraded voltage function to be operable in MODES 1, 2, 3, and 4 and when the associated DG is required to be operable by LCO 3.8.2, "AC Sources-Shutdown." The DGs provide a source of emergency power when offsite power is either unavailable or is degraded below a point that would allow safe unit operation. Undervoltage protection will generate a LOP start if a loss of voltage or degraded voltage condition occurs on the 4.16-kV Class 1E bus. There are three LOP start signals, one for each 4.16-kV Class 1E bus. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability of the LOP DG Start Instrumentation in Modes 1, 2, 3, and 4 will no longer be applicable. In addition, as described below, TS 3.8.2, "AC Sources-Shutdown" is proposed for deletion. Therefore, TS 3.3.5 is proposed for deletion.

TS 3.3.6, Containment Ventilation Isolation Instrumentation	This specification requires containment ventilation isolation instrumentation for certain functions to be operable in MODES 1, 2, 3, 4, and during movement of recently irradiated fuel assemblies within containment (according to specific applicability requirements for each function listed in TS Table 3.3.6-1). Containment ventilation isolation instrumentation closes the containment purge supply, exhaust, and the vacuum/pressure relief valves. It also closes the containment atmosphere sample valves. This action in conjunction with a Phase A signal isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability of containment ventilation system isolation instrumentation functions in MODES 1, 2, 3, and 4 will no longer be applicable. In addition, the mode of applicability associated with moving recently irradiated fuel in containment will no longer be applicable. Therefore, TS 3.3.6 is proposed for deletion.
TS 3.3.7, Control Room Ventilation System (CRVS) Actuation Instrumentation	This specification requires CRVS actuation instrumentation for specific functions to be operable in MODES 1, 2, 3, 4, 5, and 6 and during movement of recently irradiated fuel assemblies (according to specific applicability requirements for each CRVS function listed in TS Table 3.3.7-1). The CR must be kept habitable for the operators stationed there during accident recovery and post-accident operations. The CRVS acts to terminate the supply of unfiltered outside air to the CR, initiate filtration, and pressurize the CR. These actions are necessary to ensure the CR is kept habitable for the operators stationed there during accident recovery and post-accident operations by minimizing the radiation exposure of CR personnel. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), requiring CRVS actuation instrumentation for specific functions to be operable in Modes 1 through 6 and during movement of recently irradiated fuel (i.e., fuel that has occupied part of a critical

	reactor core within the previous 100 hours), will no longer be applicable. In addition, the revised FHA calculation (Reference 12) does not credit operability of the CRVS. The results of the analysis demonstrate that the dose consequences for the CR, the EAB and the LPZ remain below the acceptance criteria, without relying on active components remaining functional for accident mitigation during and following the event. As discussed in Section 2.1, PG&E will implement the revised FHA and the PDTS, 45 days after both Units have shutdown. Therefore, TS 3.3.7 is proposed for deletion.
TS 3.3.8, Fuel Building Ventilation System (FBVS) Actuation Instrumentation	This specification requires the FBVS actuation instrumentation associated with specific functions to be operable during movement of recently irradiated fuel assemblies in the FHB (according to specific applicability requirements for each FBVS function listed in TS Table 3.3.8-1). The FBVS ensures that radioactive materials in the fuel building atmosphere following a FHA involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) are filtered and adsorbed prior to exhausting to the environment. The system initiates filtered ventilation of the fuel building automatically following receipt of a high radiation signal from the SFP monitor or from the new fuel storage vault monitor. Initiation may also be performed manually as needed from the main CR or FHB. High radiation, from either of the two monitors, provides FBVS initiation. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the fuel building. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), requiring the FBVS actuation instrumentation associated with specific functions to be operable during movement of recently irradiated fuel assemblies in the FHB will no longer be applicable. In addition, the revised FHA calculation (Reference 12) does not credit operability of the FBVS. The results of the analysis demonstrate that the dose consequences for the CR, the EAB, and the LPZ remain below the acceptance criteria, without relying on active components remaining functional for accident mitigation during and following the event. As discussed in Section 2.1, PG&E will implement the revised FHA and the

	PDTS, 45 days after both units have shutdown. Therefore, TS 3.3.8 is proposed for deletion.
TS 3.3.9, Boron Dilution Protection System	This specification is not used in the current TS and therefore is proposed for deletion.

TS SECTION 3.4 – REACTOR COOLANT SYSTEM

TS Section 3.4 "Reactor Coolant System," contains LCO that provide assurance of the RCPB integrity and safe operation of the RCS. The protection and monitoring functions of the RCS have been designed to ensure safe operation of the reactor required to protect the integrity of a fission product barrier. The RCS is a primary barrier against the release of fission products to the environs.

10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation," stipulates that reactor facilities which have submitted the certifications required under 10 CFR 50.82(a)(1), no longer need to meet the fracture toughness and material surveillance program requirements for the RCPB set forth in Appendices G and H. The table below describes the specifications in this section.

This section is proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFPs. The majority of these TS are applicable in Modes which are not applicable in the permanently defueled condition. Once PG&E dockets the certifications for DCPP Units 1 and 2, required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore, the RCS specifications addressed in TS Section 3.4 will not be required and these LCO (and associated SR) will not apply in a permanently defueled condition.

Current DCPP TS	Basis for Deletion
TS 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	This specification provides limits for RCS DNB parameters for pressurizer pressure, RCS average temperature and RCS flow rate. The LCO is applicable during MODE 1 except the pressurizer pressure limit does not apply during: a. THERMAL POWER ramp greater than 5 percent RTP per minute; or b. THERMAL POWER step greater than 10 percent RTP. In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure the DNB ratio criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other

	MODES, the power level is low enough that DNB is not a concern. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), limits for RCS DNB parameters for pressurizer pressure, RCS average temperature and RCS flow rate in MODE 1, will no longer be applicable. Therefore, TS 3.4.1 is proposed for deletion.
TS 3.4.2, RCS Minimum Temperature for Criticality	This specification requires each operating RCS loop average temperature (Tavg) to be greater than or equal to 541°F and is applicable in Mode 1 and MODE 2 with k_{eff} greater than or equal to 1.0. This LCO ensures that the reactor will not be made or maintained critical (k_{eff} greater than or equal to 1.0) with an operating loop temperature less than a small band below the hot zero power temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), limits on RCS loop average temperature in MODES 1 and 2, will no longer be applicable. Therefore, TS 3.4.2 is proposed for deletion.
TS 3.4.3, RCS Pressure and Temperature (P/T) Limits	This specification requires RCS pressure, RCS temperature, and RCS heatup and cooldown rates to be maintained within the limits specified in the PTLR and is applicable at all times. All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), cyclic loads associated with startup and shutdown will no longer be applicable and therefore limits on RCS pressure and

	temperature are not required. Therefore, TS 3.4.3 is proposed for deletion.
TS 3.4.4, RCS Loops – Modes 1 and 2	This specification requires four RCS loops to be operable and in operation, and is applicable in MODES 1 and 2. An Operable RCS loop consists of one operable RCP for heat transport and the associated operable SG, with a water level within the limits specified in SR 3.4.5.2, except for operational transients. The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), core heat removal will not be applicable and therefore operability of the RCS loops is not required. Therefore, TS 3.4.4 is proposed for deletion.
TS 3.4.5, RCS Loops – Mode 3	This specification requires two RCS loops to be operable and either: a. two RCS loops shall be in operation when the rod control system is capable of rod withdrawal; or b. one RCS loop shall be in operation when the rod control system is not capable of rod withdrawal. The LCO is applicable in MODE 3. The purpose of this LCO is to require that at least two RCS loops be operable. In MODE 3 with the rod control system capable of rod withdrawal, two RCS loops must be in operation. Two RCS loops are required to be in operation in MODE 3 with the rod control system capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the SL criteria will be met for all of the postulated accidents. When the rod control system is not capable of rod withdrawal, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be operable to ensure that redundancy for heat removal is maintained. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), core heat removal

	will not be applicable and therefore operability of the RCS loops in MODE 3 is not required. Therefore, TS 3.4.5 is proposed for deletion.
TS 3.4.6 RCS Loops - MODE 4	This specification requires two loops consisting of any combination of RCS loops and RHR loops to be operable, and one loop shall be in operation. This specification is applicable in MODE 4. The purpose of this LCO is to require that at least two loops be operable in Mode 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be operable to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be operable to provide redundancy for heat removal. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), core heat removal will not be applicable and therefore operability of the RCS and RHR loops in MODE 4 is not required. Therefore, TS 3.4.6 is proposed for deletion.
TS 3.4.7, RCS Loops – MODE 5, Loops Filled	This specification requires, one RHR loop to be operable and in operation, and either: a. one additional RHR loop shall be OPERABLE; or b. the secondary side water level of at least two SGs shall be greater than or equal to 15 percent. This specification is applicable in MODE 5 with the RCS loops filled. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), decay heat removal will not be applicable and therefore operability of an RHR loop in MODE 5 is not required. Therefore, TS 3.4.7 is proposed for deletion.

TS 3.4.8, RCS Loops – Mode 5, Loops Not Filled	This specification requires two RHR loops to be operable and one RHR loop to be in operation and is applicable in MODE 5 with RCS loops not filled. In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the CCW via the RHR heat exchangers. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid. The flow provided by one RHR loop is adequate for heat removal and for boron mixing. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), heat removal and boron mixing will not be applicable and therefore operability and operation of an RHR loop in MODE 5 is not required. Therefore, TS 3.4.8 is proposed for deletion.
TS 3.4.9, Pressurizer	This specification requires the pressurizer to be operable with water level less than or equal to 90 percent and two groups of pressurizer heaters operable with the capacity of each group greater than or equal to 150 kW and capable of being powered from an emergency power supply. This specification is applicable during MODES 1, 2, and 3. The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. This LCO ensures the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), pressure control in the RCS will not be applicable and therefore operability of the pressurizer in MODES 1, 2, and 3 is not required. Therefore, TS 3.4.9 is proposed for deletion.
TS 3.4.10 Pressurizer Safety Valves	This specification requires three pressurizer safety valves to be operable with lift settings greater than or equal to 2460 psig and less than or equal to 2510 psig. The specification is applicable in MODES 1, 2, 3, and MODE 4 with all RCS cold leg temperatures greater than LTOP arming temperature specified in the PTLR.

	The pressurizer safety valves provide, in conjunction with the reactor protection system, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self-actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system SL, 2735 psig, which is 110 percent of the design pressure. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), overpressure protection for the RCS will not be applicable and therefore operability of the pressurizer safety valves in MODES 1, 2, 3, and 4 is not required. Therefore, TS 3.4.10 is proposed for deletion.
TS 3.4.11, Pressurizer Power Operated Relief Valves (PORVs)	This specification requires each PORV and associated block valve to be operable, and is applicable in MODES 1, 2, and 3. The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are air operated valves that are controlled to open when the pressurizer pressure increases above their actuation setpoint and to close when the pressurizer pressure decreases. In MODES 1, 2, and 3, the PORVs are required to be operable to mitigate a SGTR and spurious operation of the safety injection system at power event, and the main feedwater line break event, and the block valves are required to be operable to limit the potential for a small break LOCA through the flow path. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), overpressure protection for the RCS will not be applicable and therefore operability of the PORVs and associated block valves in MODES 1, 2, and 3 is not required. Therefore, TS 3.4.11 is proposed for deletion.
TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System	This specification requires the LTOP system to be operable during certain conditions. This specification is applicable in the following modes: • MODE 4, when any RCS cold leg temperature is less than or equal to LTOP arming temperature specified in the PTLR,

	• MODE 5, • MODE 6, when the reactor vessel head is on and the vessel head closure bolts are not fully de-tensioned. The LTOP system controls RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the pressure and temperature limits of 10 CFR 50, Appendix G. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), controls for RCS pressure will no longer be required. Therefore, TS 3.4.12 is proposed for deletion.
TS 3.4.13, RCS Operational Leakage	This specification provides limits on RCS operational LEAKAGE and is applicable during MODES 1, 2, 3, and 4. The purpose of this specification is to limit system operation in the presence of LEAKAGE from the RCS to amounts that do not compromise safety. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), limits on RCS LEAKAGE in MODES 1, 2, 3, and 4 will no longer be applicable. Therefore, TS 3.4.13 is proposed for deletion.
TS 3.4.14, RCS Pressure Isolation Valve Leakage	This specification limits leakage from each RCS PIV and is applicable during MODES 1, 2, 3, and 4 (with exceptions). RCS PIVs are defined as any two normally closed valves in series within the RCPB, which separates the high-pressure RCS from an attached low-pressure system. The purpose of this specification is to prevent overpressure failure of the low-pressure portions of systems that connect to the RCS. Exceeding the leakage limit may indicate the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low-pressure piping or components. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), limits on leakage from RCS PIVs in MODES 1, 2, 3, and 4 will no longer be applicable. Therefore, TS 3.4.14 is proposed for deletion.

TS 3.4.15, RCS Leakage Detection Instrumentation	This specification specifies which RCS leakage detection instrumentation shall be operable and is applicable in MODES 1, 2, 3, and 4. The leakage detection systems must have the capability to detect significant RCPB degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. The RCS leakage detection instrumentation provides an early indication or warning signal to permit proper evaluation of all unidentified leakage. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), RCS leakage detection instrumentation in MODES 1, 2, 3, and 4 will no longer be required. Therefore, TS 3.4.15 is proposed for deletion.
TS 3.4.16 RCS Specific Activity	This specification limits RCS dose equivalent I-131 and dose equivalent XE-133 specific activity and is applicable in MODES 1, 2, 3, and 4. The limits on the specific activity of the reactor coolant ensures that the resulting offsite and CR doses meet the appropriate acceptance criteria following a steam line break or a SGTR accident. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), limits on RCS specific activity in MODES 1, 2, 3, and 4 will no longer be applicable. Therefore, TS 3.4.16 is proposed for deletion.
TS 3.4.17 Steam Generator Tube Integrity	This specification requires that SG tube integrity be maintained and tubes satisfying the tube repair criteria be plugged. This specification is applicable in MODES 1, 2, 3, and 4. Maintaining SG tube integrity ensures that the tubes are capable of performing the intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements. The SGTR accident is the limiting DBA for SG tubes and therefore avoiding this accident is the basis for this specification. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), SG tube integrity and

	the SGTR accident will no longer be applicable. In addition, TS 5.5.9, “Steam Generator (SG) Tube Inspection Program,” is proposed for deletion. Therefore, TS 3.4.17 is proposed for deletion.

TS Section 3.5 – EMERGENCY CORE COOLING SYSTEMS	
TS Section 3.5, "Emergency Core Cooling Systems," contains LCO to assure the operability of the ECCS to provide core cooling and negative reactivity to ensure the reactor core is protected during certain accidents. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," specifies that light-water nuclear power reactors must have ECCS designed to meet the cooling performance requirements following postulated LOCAs. 10 CFR 50.46(a)(1)(i) states "This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted." This section is proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of SNF in the SFPs. The TS are related to assuring the appropriate functional capability of the ECCS required for mitigation of DBAs only when the reactor is in Modes 1 through 4. Once PG&E dockets the certifications for DCPP Units 1 and 2, required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore, the core cooling specifications addressed in TS Section 3.5 will not be required and these LCO (and associated SR) will not apply in a permanently defueled condition.	
Current DCPP TS	**Basis for Deletion**
TS 3.5.1, Accumulators	This specification requires four ECCS accumulators to be operable in MODES 1 and 2, and Mode 3 with RCS pressure greater than 1000 psig. This specification establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability of the accumulators in MODES 1, 2, and 3 will no longer be applicable. Therefore, TS 3.5.1 is proposed for deletion.
TS 3.5.2, ECCS - Operating	This specification requires two ECCS trains to be operable in MODES 1, 2, and 3. The purpose of this specification is to help ensure that the acceptance criteria for the ECCS, established by 10 CFR

	50.46, will be met following a LOCA. In addition, it limits the potential for a post-trip return to power following a main steam line break (MSLB) event and ensures that containment temperature limits are met. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability of the ECCS in MODES 1, 2, and 3 will no longer be applicable. Therefore, TS 3.5.2 is proposed for deletion.
TS 3.5.3, ECCS - Shutdown	This specification requires one ECCS train to be operable in MODE 4. The purpose of this specification is similar to TS 3.5.2. However, in MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS high head and low head train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability of the ECCS in MODE 4 will no longer be applicable. Therefore, TS 3.5.3 is proposed for deletion.
TS 3.5.4, Refueling Water Storage Tank (RWST)	This specification requires the RWST to be operable in MODES 1, 2, 3, and 4. The purpose of this specification is to ensure that an adequate supply of borated water is available to cool and depressurize the containment in the event of a DBA, to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment recirculation sump to support ECCS pump operation in the recirculation mode. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability of the RWST in MODES 1, 2, 3, and 4 will no longer be applicable. Therefore, TS 3.5.4 is proposed for deletion.

TS 3.5.5, Seal Injection Flow	This specification requires RCP seal injection flow resistance to be greater than or equal to 0.2117 ft/gpm^2, and is applicable in MODES 1, 2, and 3. The intent of this specification is to ensure that the seal injection flow resistance remains within limit. This in turn assures that flow through the RCP seal injection line during an accident is restricted. The seal injection flow is restricted by the injection line hydraulic flow resistance which is adjusted through positioning of the manual seal injection throttle valves. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), the RCPs are no longer needed and therefore limits on RCP seal injection flow resistance in MODES 1, 2, and 3 will no longer be applicable. Therefore, TS 3.5.5 is proposed for deletion.

TS 3.6 – CONTAINMENT SYSTEMS

TS Section 3.6 "Containment Systems," contains LCO that assure the integrity of the containment systems. The containment provides a barrier against uncontrolled release of fission products to the environs and provides shielding from the fission products that may be present in the containment during an accident.

This section is proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFPs. All TS in Section 3.6 are related to assuring the appropriate functional capability of plant equipment associated with containment systems required for safe operation of the facility and accident mitigation only when the reactor is in Modes 1 through 4. Once PG&E dockets the certifications for DCPP Units 1 and 2, required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications for the containment systems addressed in TS Section 3.6 will not be required and these LCO (and associated SR) will not apply in a permanently defueled condition.

Current DCPP TS	Basis for Deletion
TS 3.6.1, Containment	This specification requires containment to be operable in MODES 1, 2, 3, and 4. Containment consists of the concrete reactor building, its steel liner, and the penetrations through the structure. Operability of the structure ensures containment can perform the design function of containing radioactive material that may be released from the reactor core following a Design Basis LOCA. Additionally, operability ensures the structure is capable of

	providing shielding from the fission products that may be present in the containment atmosphere following accident conditions. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), the containment design function of containing radioactive material and providing shielding when the reactor is in MODES 1 through 4, will no longer be applicable. Therefore, TS 3.6.1 is proposed for deletion.
TS 3.6.2, Containment Air Locks	This specification requires two containment air locks to be operable in MODES 1, 2, 3, and 4. The containment air locks form part of the containment pressure boundary, and therefore the safety function is related to control of containment leakage in the event of a DBA that could cause a release of radioactive material to containment. The purpose of this specification is to ensure the structural integrity and leak tightness of the air locks in order to maintain the pressure boundary and mitigate such DBAs. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), the air locks which support the containment design function of containing radioactive material and providing shielding in MODES 1 through 4, will no longer be applicable. Therefore, TS 3.6.2 is proposed for deletion.
TS 3.6.3, Containment Isolation Valves	This specification requires each containment isolation valve to be operable in MODES 1, 2, 3, and 4. This specification provides assurance that the containment isolation valves and the containment purge supply and exhaust, and containment pressure/vacuum relief valves will perform their designed safety function to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability of the containment isolation valves in MODES 1 through 4, will no longer be applicable. Therefore, TS 3.6.3 is proposed for deletion.

TS 3.6.4, Containment Pressure	This specification requires containment pressure to be greater than or equal to -1.0 psig and less than or equal to +1.2 psig in MODES 1, 2, 3, and 4. Maintaining containment pressure at less than or equal to the specification upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the containment spray system. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining containment pressure within specific limits in MODES 1 through 4, will no longer be applicable. Therefore, TS 3.6.4 is proposed for deletion.
TS 3.6.5, Containment Air Temperature	This specification requires containment average air temperature to be less than or equal to 120°F in MODES 1, 2, 3, and 4. This specification ensures that the temperature profile resulting from an accident will be maintained below the containment design temperature and that the required PG&E Design Class I equipment within containment will continue to perform its function. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining containment air temperature within the specified limits in MODES 1 through 4, will no longer be applicable. Therefore, TS 3.6.5 is proposed for deletion.
TS 3.6.6, Containment Spray and Cooling Systems	This specification requires the containment fan cooling unit (CFCU) system and two containment spray trains to be operable in MODES 1, 2, 3, and 4. Containment spray and cooling systems provide containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the reduction of fission products in the containment atmosphere reduces the release

	of fission product radioactivity from containment to the environment, in the event of a DBA, to within limits. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining CFCUs and two containment spray trains operable in MODES 1 through 4, will no longer be applicable. Therefore, TS 3.6.6 is proposed for deletion.
TS 3.6.7, Spray Additive System	This specification requires the spray additive system to be operable in MODES 1, 2, 3, and 4. The spray additive system is a subsystem of the containment spray system that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a design basis accident loss of coolant accident (DBA LOCA). Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining the Spray Additive System operable in MODES 1 through 4, will no longer be applicable. In addition, a DBA LOCA is no longer possible in the permanently defueled condition. Therefore, TS 3.6.7 is proposed for deletion.

TS Section 3.7 – PLANT SYSTEMS

TS 3.7, "Plant Systems," contains LCO that provide for appropriate functional capability of plant equipment required for safe operation of the facility, including the plant being in a permanently defueled condition.

<u>TS 3.7.15, 3.7.16, 3.7.17</u>

The above listed sections of TS 3.7 are applicable to the safe storage and handling of spent fuel in the SFPs and therefore, are still applicable in the permanently defueled condition. These specifications are being proposed for inclusion in the PDTS with the proposed changes described below.

<u>TS 3.7.1, 3.7.2, 3.7.3, 3.7.4, 3.7.5, 3.7.6, 3.7.7, 3.7.8, 3.7.9, 3.7.12, and 3.7.18</u>

As described further below, the above listed sections of TS 3.7 are proposed for deletion. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. The above listed TS in Section 3.7 are related to assuring the appropriate capability of plant systems and components required for safe operation of the facility and accident mitigation only when the reactor is in Modes 1 through 4. Once PG&E dockets the certifications for DCPP Units 1 and 2, required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications for the plant systems and components listed above within TS Section 3.7 will not be required and these LCO (and associated SR) will not apply in a permanently defueled condition.

<u>TS 3.7.10</u>
As described further below, the above listed section of TS 3.7 is proposed for deletion. This TS is only applicable when the reactor is in Modes 1 through 6 and during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours). Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2) and the PDTS will not be effective until 45 days after both units have shutdown, this TS will no longer be applicable.

<u>TS 3.7.11, and 3.7.14</u>
As described below, the above listed sections of TS 3.7 are not used within the current TS and therefore are proposed for deletion.

<u>TS 3.7.13</u>
As described further below, the above listed section of TS 3.7 is proposed for deletion. This TS is only applicable during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours). Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2) and the PDTS will not be effective until 45 days after both units have shutdown, this TS will no longer be applicable.

<table>
<tr><th>Current DCPP TS</th><th>Proposed TS</th></tr>
<tr>
<td><u>TS 3.7.15, Spent Fuel Pool Water Level</u>
LCO 3.7.15:
The spent fuel pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

Applicability:
During movement of irradiated fuel assemblies in the spent fuel pool.

Actions:
<table>
<tr><th>CONDITION</th><th>REQUIRED ACTION</th><th>COMPLETION TIME</th></tr>
<tr><td>A. Spent fuel pool water level not within limit.</td><td>A.1
-----NOTE-----
LCO 3.0.3 is not applicable.

Suspend movement of irradiated fuel assemblies in the spent fuel pool.</td><td>Immediately</td></tr>
</table></td>
<td><u>TS 3.7.15, Spent Fuel Pool Water Level</u>
LCO 3.7.15
The spent fuel pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

Applicability:
During movement of irradiated fuel assemblies in the spent fuel pool.

Actions:
<table>
<tr><th>CONDITION</th><th>REQUIRED ACTION</th><th>COMPLETION TIME</th></tr>
<tr><td>A. Spent fuel pool water level not within limit.</td><td>A.1
~~-----NOTE-----~~
~~LCO 3.0.3 is not applicable.~~

Suspend movement of irradiated fuel assemblies in the spent fuel pool.</td><td>Immediately</td></tr>
</table></td>
</tr>
</table>

Surveillance Requirements:

SURVEILLANCE	FREQUENCY
SR 3.7.15.1 Verify the spent fuel pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	In accordance with the Surveillance Frequency Control Program

TS 3.7.16, Spent Fuel Pool Boron Concentration

LCO 3.7.16:
The spent fuel pool boron concentration shall be ≥ 2000 ppm.

Applicability:
When fuel assemblies are stored in the spent fuel pool.

Actions:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	-----NOTE----- LCO 3.0.3 is not applicable. -------------------	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

Surveillance Requirements:

SURVEILLANCE	FREQUENCY
SR 3.7.16.1 Verify the spent fuel pool boron concentration is within limit.	In accordance with the Surveillance Frequency Control Program

Surveillance Requirements:

SURVEILLANCE	FREQUENCY
SR 3.7.15.1 Verify the spent fuel pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	~~In accordance with the Surveillance Frequency Control Program~~ *7 days*

TS 3.7.16, Spent Fuel Pool Boron Concentration

LCO 3.7.16:
The spent fuel pool boron concentration shall be ≥ 2000 ppm.

Applicability:
When fuel assemblies are stored in the spent fuel pool.

Actions:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	~~-----NOTE----- - LCO 3.0.3 is not applicable. ------------------ -~~	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

Surveillance Requirements:

SURVEILLANCE	FREQUENCY
SR 3.7.16.1	~~In accordance with the~~

<table>
<tr><td></td><td><table><tr><td>Verify the spent fuel pool boron concentration is within limit.</td><td>~~Surveillance Frequency Control Program~~
7 days</td></tr></table></td></tr>
<tr><td>TS 3.7.17, Spent Fuel Assembly Storage
LCO 3.7.17:
Fuel assembly storage in the spent fuel pool shall be maintained such that:
a. In the permanent spent fuel storage racks any four cells shall be in a configuration as shown in Figure 3.7.17-1, and
b. In the cask pit storage rack, for Cycles 14 – 16, the fuel assemblies shall have:
1. An initial enrichment ≤ 4.1 wt% U-235;
2. A discharge burnup in the “acceptable” region of Figure 3.7.17-4; and
3. A minimum decay time of 10 years since being discharged from the reactor.
c. The total combined spent fuel pool capacity in the permanent and cask pit storage racks, for Cycles 14 – 16, is limited to no more than 1433 irradiated fuel assemblies. This limit does not apply for an emergency core offload.

Applicability:
Whenever any fuel assembly is stored in the spent fuel pool.

Actions:<table><tr><th>CONDITION</th><th>REQUIRED ACTION</th><th>COMPLETION TIME</th></tr><tr><td>A. Requirements of the LCO not met</td><td>A.1
-----NOTE-----
LCO 3.0.3 is not applicable.

Initiate action to move the noncomplying fuel assembly into an acceptable storage location.</td><td>Immediately</td></tr></table></td><td>TS 3.7.17, Spent Fuel Assembly Storage
LCO 3.7.17:
Fuel assembly storage in the spent fuel pool shall be maintained such that:
a. In the permanent spent fuel storage racks any four cells shall be in a configuration as shown in Figure 3.7.17-1.~~, and~~
~~b. In the cask pit storage rack, for Cycles 14 – 16, the fuel assemblies shall have:~~
~~1. An initial enrichment ≤ 4.1 wt% U-235;~~
~~2. A discharge burnup in the “acceptable” region of Figure 3.7.17-4; and~~
~~3. A minimum decay time of 10 years since being discharged from the reactor.~~
~~c. The total combined spent fuel pool capacity in the permanent and cask pit storage racks, for Cycles 14 – 16, is limited to no more than 1433 irradiated fuel assemblies. This limit does not apply for an emergency core offload.~~

Applicability:
Whenever any fuel assembly is stored in the spent fuel pool.

Actions:<table><tr><th>CONDITION</th><th>REQUIRED ACTION</th><th>COMPLETION TIME</th></tr><tr><td>A. Requirements of the LCO not met</td><td>A.1
~~-----NOTE-----~~
~~LCO 3.0.3 is not applicable.~~
~~-------------------~~
Initiate action to move the noncomplying fuel assembly into an acceptable storage location.</td><td>Immediately</td></tr></table></td></tr>
<tr><td colspan="2">**Basis**</td></tr>
<tr><td colspan="2">TS 3.7.15 – This TS ensures the minimum water level in the SFP meets the assumptions of iodine decontamination factors following a FHA. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water</td></tr>
</table>

also provides shielding during the movement of spent fuel. This TS is applicable during the movement of irradiated fuel assemblies in the SFP and will be retained in the PDTS with the following change. The Note in Required Action A.1 (LCO 3.0.3 is not applicable) will be deleted to conform with the deletion of LCO 3.0.3 described in TS Section 3.0 above. In addition, SR 3.7.15.1 will be revised to remove reference to the Surveillance Frequency Control Program. As discussed below, TS 5.5.18, “Surveillance Frequency Control Program,” is proposed for deletion. The surveillance frequency is revised to reflect 7 days, which is consistent with the current frequency requirements included in the Surveillance Frequency Control Program. Retaining this TS with the proposed changes, continues to ensure appropriate requirements for SFP water level.

TS 3.7.16 – This TS ensures the concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios. The specified boron concentration of 2000 ppm ensures that the SFP k_{eff} will remain less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level, for a postulated reactivity insertion accident or boron dilution event. This TS is applicable when fuel assemblies are stored in the SFP and is retained in the PDTS with the following change. The Note in Required Action A.1 (LCO 3.0.3 is not applicable) will be deleted to conform with the deletion of LCO 3.0.3 described in TS Section 3.0 above. In addition, SR 3.7.16.1 will be revised to remove reference to the Surveillance Frequency Control Program. As discussed below, TS 5.5.18, “Surveillance Frequency Control Program,” is proposed for deletion. The surveillance frequency will be revised to reflect 7 days, which is consistent with the current frequency requirements included in the Surveillance Frequency Control Program. Retaining this TS with the proposed changes, continues to ensure appropriate requirements for storing fuel assemblies in the SFP.

TS 3.7.17 – This TS places restrictions on the placement of fuel assemblies within the SFP, to ensure the k_{eff} of the spent fuel storage pool will always remain less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level, for a postulated reactivity insertion accident or a boron dilution event. This TS is applicable when any fuel assembly is stored in the SFP and will be retained in the PDTS (including Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3) with the following changes. TS 3.7.17.b, 3.7.17.c, and the associated Figure 3.7.17-4 are being proposed for deletion. TS 3.7.17.b, 3.7.17.c, and the associated Figure 3.7.17-4 were added to the DCPP TS via License Amendments 183 and 185 approved by the NRC on November 21, 2005 (Reference 23). These amendments allowed the installation and use of a temporary cask pit spent fuel storage rack for Units 1 and 2 during Cycles 14-16. This TS LCO is historical in nature and will not be applicable during decommissioning. Therefore, TS 3.7.17.b, 3.7.17.c, and the associated Figure 3.7.17-4 are proposed for deletion.

Current DCPP TS	**Basis for Deletion**
TS 3.7.1, Main Steam Safety Valves (MSSVs)	This specification requires five MSSVs per SG to be operable in MODES 1, 2, and 3. This specification ensures five MSSVs are operable to provide overpressure protection for design basis transients. This provides assurance that the MSSVs will perform their designed safety

	functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or main steam system integrity. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining MSSVs operable in MODES 1 through 3, will no longer be applicable. Therefore, TS 3.7.1 is proposed for deletion.
TS 3.7.2, Main Steam Isolation Valves (MSIVs)	This specification requires four MSIVs to be operable in MODE 1 and MODES 2 and 3 except when all MSIVs are closed and deactivated. The MSIVs isolate steam flow from the secondary side of the SGs following a high energy line break. MSIV closure terminates flow from the unaffected (intact) SGs. This specification provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to applicable limits. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining MSIVs operable in MODES 1 through 3, will no longer be applicable. Therefore, TS 3.7.2 is proposed for deletion.
TS 3.7.3, Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs), MFRV Bypass Valves, and Main Feedwater Pump (MFWP) Turbine Stop Valves	This specification requires four MFIVs, four MFRVs, four MFRV bypass valves and four MFWP turbine stop valves to be operable in MODES 1, 2, and 3 (except under specific configurations as described in the TS). This specification ensures that the MFIVs, MFRVs and MFRV bypass valves, and tripping of the MFWPs, will isolate main feedwater flow to the SGs, following a feedwater line break or MSLB, or an excessive feedwater event. The MFIVs will also isolate the non-PG&E Design Class I portions from the PG&E Design Class I portions of the system. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining operability of the MFIVs, MFRVs, MFRV bypass valves, and MFWP turbine stop valves operable in MODES 1 through 3, will no longer be applicable. Therefore, TS 3.7.3 is proposed for deletion.
TS 3.7.4, 10% Atmospheric Dump Valves (ADVs)	This specification requires four ADV lines to be operable in MODES 1, 2, and 3 and Mode 4 when SG is relied upon for heat removal.

	This specification ensures the ability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the steam bypass system. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining four ADV lines operable in MODES 1 through 4 will no longer be applicable. Therefore, TS 3.7.4 is proposed for deletion.
TS 3.7.5, Auxiliary Feedwater (AFW) System	This specification requires three AFW trains to be operable in MODES 1, 2 and 3 and MODE 4 when SG is relied upon for heat removal. This specification provides assurance that the AFW system will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the RCPB. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining AFW trains operable in MODES 1 through 4 will no longer be applicable. Therefore, TS 3.7.5 is proposed for deletion.
3.7.6, Condensate Storage Tank (CST)	This specification requires the CST to be operable in MODES 1, 2, and 3 and Mode 4 when SG is relied upon for heat removal. This specification ensures the minimum required tank volume is available to provide a source of water to the SGs for removing decay and sensible heat from the RCS. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining the CST operable in MODES 1 through 4 will no longer be applicable. Therefore, TS 3.7.6 is proposed for deletion.
3.7.7, Vital Component Cooling Water (CCW) System	This specification requires two vital CCW loops to be operable in MODES 1, 2, 3, and 4. The CCW system provides a heat sink for the removal of process and operating heat from PG&E Design Class I components during a DBA or transient. This specification ensures the minimum heat removal capability assumed in the safety analysis is provided for the systems to which the CCW supplies cooling water. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels,

	pursuant to 10 CFR 50.82(a)(2), maintaining two CCW loops operable in MODES 1 through 4 will no longer be applicable. Therefore, TS 3.7.7 is proposed for deletion.
3.7.8, Auxiliary Saltwater (ASW) System	This specification requires two ASW trains to be operable in MODES 1, 2, 3, and 4. The ASW system provides a heat sink from the Pacific Ocean for the removal of process and operating heat from the CCW system. This specification ensures the required ASW trains are available to remove post-accident heat loads. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining two ASW trains operable in MODES 1 through 4 will no longer be applicable. Therefore, TS 3.7.8 is proposed for deletion.
3.7.9, Ultimate Heat Sink (UHS)	This specification requires the UHS to be operable in MODES 1, 2, 3, and 4. The UHS provides a heat sink for transferring heat from PG&E Design Class I components during a transient or accident, as well as PG&E Design Class I and non-PG&E Design Class I heat loads during normal operation. This is done by utilizing the Pacific Ocean, the ASW and the CCW system. This specification ensures the UHS is at or below the maximum temperature that would allow the ASW to operate for at least 30 days following the DBA without exceeding the maximum design temperature of the CCW system. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining the UHS operable in MODES 1 through 4 will no longer be applicable. Therefore, TS 3.7.9 is proposed for deletion.
TS 3.7.10, Control Room Ventilation System (CRVS)	This specification requires two CRVS trains to be operable in MODES 1, 2, 3, 4, 5, and 6, and during movement of recently irradiated fuel assemblies. In MODES 1, 2, 3, 4, 5, and 6, and during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours), the CRVS must be operable to ensure that the CR envelope will remain habitable during and following a DBA or the release from the rupture of an outside waste gas tank.

	During movement of recently irradiated fuel assemblies, the CRVS must be operable to cope with the release from a FHA involving handling recently irradiated fuel. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), requiring two CRVS trains to be operable in MODES 1 through 6 and during movement of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours), will no longer be applicable. Therefore, TS 3.7.10 is proposed for deletion. In addition, the revised FHA calculation (Reference 12) does not credit operability of the CRVS trains. The results of the analysis demonstrate that the dose consequences for the CR, and the EAB remain below the acceptance criteria, without relying on active components remaining functional for accident mitigation during and following the event. As discussed in Section 2.1, PG&E will implement the revised FHA and the PDTS, 45 days after both Units have shutdown.
TS 3.7.11, Control Room Emergency Air Temperature Control System (CREATCS)	This specification is not used in the current TS and therefore is proposed for deletion.
TS 3.7.12, Auxiliary Building Ventilation System (ABVS)	This specification requires two trains of the ABVS to be operable in MODES 1, 2, 3, and 4. The ABVS filters air from the area of the active ECCS components during the recirculation phase of a LOCA. The ABVS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the ECCS pump room area, if one of the pumps is operating, and the auxiliary building. This specification ensures the atmospheric release from the ECCS pump room do not exceed 10 CFR 50.67 limits in the event of a DBA. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining the ABVS operable in MODES 1 through 4 will no longer be applicable. Therefore, TS 3.7.12 is proposed for deletion.
TS 3.7.13, Fuel Handling Building Ventilation System (FHBVS)	This specification requires two FHBVS trains to be operable during movement of recently irradiated fuel assemblies in the FHB. During movement of recently irradiated fuel in the FHB, the FHBVS is required to be operable to alleviate the consequences of a FHA.

	Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), requiring two FHBVS trains to be operable during movement of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours), will no longer be applicable. Therefore, TS 3.7.13 is proposed for deletion. In addition, the revised FHA calculation (Reference 12) does not credit operability of the FHBVS trains. The results of the analysis demonstrate that the dose consequences for the CR, and the EAB remain below the acceptance criteria, without relying on active components remaining functional for accident mitigation during and following the event. As discussed in Section 2.1, PG&E will implement the revised FHA and the PDTS, 45 days after both Units have shutdown.
TS 3.7.14, Penetration Room Exhaust Air Cleanup System (PREACS)	This specification is not used in the current TS and therefore is proposed for deletion.
TS 3.7.18, Secondary Specific Activity	This specification requires the specific activity of the secondary coolant to be less than or equal to 0.10 μCi/gm DOSE EQUIVALENT I-131 in MODES 1, 2, 3, and 4. Activity in the secondary coolant results from SG tube leakage from the RCS. This specification limits secondary coolant specific activity during power operation to minimize releases to the environment because of normal operation, AOOs, and accidents. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), limits on specific activity of the secondary coolant in MODES 1 through 4 will no longer be applicable. Therefore, TS 3.7.18 is proposed for deletion.

TS Section 3.8 – ELECTRICAL POWER SYSTEMS
TS Section 3.8, "Electrical Power Systems," contains LCO related to the operability of alternating current (AC) and direct current (DC) electrical systems. This section establishes the requirements for appropriate functional capability of plant electrical equipment required for safe operation of the facility. This section specifies requirements to ensure that the station safety-related electrical buses and distribution system, offsite power sources, and the onsite standby power sources (emergency diesel generators (EDGs)), provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESFs systems so that the

fuel, RCS, and containment design limits are not exceeded. The requirements for the EDG fuel oil storage are included for each EDG. This section also includes the requirements for DC power. It specifies requirements to ensure that the DC electrical power systems are operable. The DBAs and transients analyzed in UFSAR Chapter 15 will no longer be applicable in the permanently defueled condition, with the exception of the FHA in the FHB. The revised FHA calculation (Reference 12) shows the dose consequences are acceptable without relying on any active SSCs to remain functional during and following the event. As indicated in Section 2.1, PG&E will not implement the revised FHA and the changes included in this LAR until both Units have been shut down for at least 45 days. Because the FHA does not rely on normal or emergency power for accident mitigation (including any need for providing airborne radiological protection), the AC sources are not required during movement of irradiated fuel assemblies in the SFP for mitigation of a potential FHA. Therefore, during movement of irradiated fuel assemblies in the SFP, there are no active systems credited as part of the initial conditions of the analysis or as part of the primary success path for mitigation of the FHA with DCPP Units 1 and 2 permanently defueled. Therefore, the requirements for AC and DC sources are being deleted because there are no design basis events that rely on these sources for mitigation. This section is proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once PG&E dockets the certifications for DCPP Units 1 and 2, required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 licenses will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS Section 3.8 will not be required and these LCO (and associated SR) will not apply in a permanently defueled condition.	
Current DCPP TS	**Basis for Deletion**
TS 3.8.1, AC Sources – Operating	This specification identifies the AC electrical power sources that shall be operable in MODES 1, 2, 3, and 4. Operability of these systems ensures that acceptable fuel design limits and RCPB limits are not exceeded as a result of AOOs or abnormal transients. In addition, operability of these systems ensures that adequate core cooling is provided and containment operability and other PG&E Design Class I functions are maintained in the event of a postulated DBA. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2),

	maintaining the AC electrical power sources operable in MODES 1, 2, 3, and 4, will no longer be applicable. Therefore, TS 3.8.1 is proposed for deletion.
TS 3.8.2, AC Sources - Shutdown	This specification identifies the AC electrical power sources that shall be operable in MODES 5 and 6, and during movement of recently irradiated fuel assemblies. The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that: a. the unit can be maintained in the shutdown or refueling condition for extended periods; b. sufficient I&C capability is available for monitoring and maintaining the unit status; and c. adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a FHA involving handling recently irradiated fuel. Due to radioactive decay, AC electrical power is only required to mitigate FHAs involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours). Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), this specification for assuring the appropriate functional capability of the AC sources for safe operation of the facility when the reactor is in MODES 5 and 6, will no longer be applicable. The only DBA remaining is the FHA in the FHB which does not credit any AC sources for mitigation of the accident. There are no active systems credited as part of the initial conditions of the analysis or as part of the primary success path for mitigation of the FHA in the FHB with DCPP Units 1 and 2 in the permanently shutdown and defueled condition. The only electrically powered active system important for the storage of irradiated fuel is the SFP cooling, makeup and support systems. The SFP cooling system did not meet the criteria in 10 CFR 50.36 for

	inclusion in the DCPP TS even when the reactor was authorized to operate. In addition, the mode of applicability associated with moving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours), will no longer be applicable. The PDTS will become effective after the certifications of permanent fuel removal have been submitted and both units have been shut down for 45 days, which is greater than 100 hours. Therefore, this specification will no longer be needed for assuring the appropriate functional capability of the AC sources for safe operation of the facility when moving recently irradiated fuel assemblies and is proposed for deletion.
TS 3.8.3, Diesel Fuel Oil, Lube Oil, Starting Air, and Turbocharger Air Assist	This specification ensures specific limits are met for the stored diesel fuel oil, lube oil, starting air, and turbocharger air assist subsystems for each required EDG. The specification is applicable when the associated EDG is required to be OPERABLE. For proper operation of the EDGs, it is necessary to ensure sufficient quantity and proper quality of the fuel oil as well as sufficient quantity of lube oil. Stored diesel fuel oil is required to have sufficient supply to support the operation of the EDGs to power the minimum ESF systems required to mitigate a DBA LOCA in one unit and those minimum required systems for a concurrent non-LOCA safe shutdown in the remaining unit for a period of seven days. Each EDG has two redundant 100 percent capacity air start systems and a turbocharger air assist system with adequate capacity for three successive start attempts each on the EDG without recharging the air start receivers or the turbocharger air assist air receiver. The AC sources (TS 3.8.1 and 3.8.2) are required to ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO, abnormal transient or a postulated DBA. Since stored fuel oil, lube oil, starting air and turbocharger air assist support TS 3.8.1 and 3.8.2, the limits included in

	TS 3.8.3 must be met when the associated EDG is required to be operable. TS 3.8.3 is required to support the EDG requirements of TS 3.8.1 and 3.8.2. With the deletion of TS 3.8.1 and 3.8.2, the requirements of TS 3.8.3 are no longer applicable and are proposed for deletion. As discussed in the justification for deleting TS 3.8.2 above, the requirement for EDGs will be deleted from the TS because there are no DBAs or transients applicable to the facility in a permanently defueled condition that rely on the EDGs for mitigation. Since TS 3.8.3 exists solely to support the EDG requirements of TS 3.8.1 and TS 3.8.2, the elimination of the need for EDGs also obviates the need for the support systems. Therefore, TS 3.8.3 is proposed for deletion.
TS 3.8.4, DC Sources - Operating	This specification requires three Class 1E DC electrical power subsystems to be operable in MODES 1, 2, 3, and 4. The Class 1E DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected PG&E Design Class I equipment and backup 120-Vac Class 1E bus power (via inverters). The Class 1E DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. Since PG&E will no longer be authorized to operate the DCPP Units 1 and 2 reactors or place or retain of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), assuring operability of the Class 1E DC electrical power subsystems when the reactor is in MODES 1, 2, 3, and 4, will no longer be applicable. Therefore, TS 3.8.4 is proposed for deletion.
TS 3.8.5, DC Sources - Shutdown	This specification requires the Class 1E DC electrical power subsystem to be operable to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems-Shutdown," and is

	applicable in MODES 5 and 6, and during movement of recently irradiated fuel assemblies. The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that: a. the unit can be maintained in the shutdown or refueling condition for extended periods; b. sufficient I&C capability is available for monitoring and maintaining the unit status; and c. adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a FHA involving handling recently irradiated fuel. Due to radioactive decay, DC electrical power is only required to mitigate FHAs involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours). Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), assuring the appropriate functional capability of the DC sources for safe operation of the facility when the reactor is in MODES 5 and 6, will no longer be applicable. The only DBA remaining is the FHA in the FHB which does not credit any DC sources for mitigation of the accident. DC sources are therefore not required during movement of irradiated fuel assemblies in the SFP for mitigation of a potential FHA. There are no active systems credited as part of the initial conditions of the analysis or as part of the primary success path for mitigation of the FHA in the FHB with DCPP Units 1 and 2 in the permanently shutdown and defueled condition. In addition, the mode of applicability associated with moving recently irradiated fuel is no longer applicable. Therefore, this specification will no longer be needed for assuring the appropriate functional capability of the AC sources for safe operation of the facility when moving recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours), and is proposed for deletion. The PDTS will not be implemented until 45 days after both Units have shutdown.

TS 3.8.6, Battery Parameters	This specification requires the parameters for the three Class 1E batteries to be within limits, and is applicable when the associated DC electrical power subsystems are required to be operable. This specification delineates the limits on battery float current as well as electrolyte temperature, level, and float voltage for the DC electrical power subsystem batteries. In addition to the limitations of this specification, the Battery Monitoring and Maintenance Program also implements a program specified in TS 5.5.17 for monitoring various battery parameters. Battery parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA. Battery parameters are required solely for the support of the associated DC electrical power subsystems (TS 3.8.4 and TS 3.8.5). Therefore, battery parameter limits are only required (and TS 3.8.6 is only applicable) when the DC electrical power subsystems are required to be operable. TS 3.8.4 and TS 3.8.5 are proposed for deletion, and therefore TS 3.8.6 is also proposed for deletion.
TS 3.8.7, Inverters-Operating	This specification requires four Class 1E vital 120 V uninterruptible power supply (UPS) inverters to be operable in MODES 1, 2, 3, and 4. The Class 1E UPS inverters are the preferred source of power for the AC Class 1E buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the Class 1E buses. The inverters can be powered from an internal AC source/rectifier or from the station battery. The station battery provides an uninterruptible power source for the I&Cs for the reactor protective system (RPS) and the ESFAS. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), assuring the operability of the inverters in MODE 1, 2, 3, and 4, will no longer be applicable. Therefore, TS 3.8.7 is proposed for deletion.
TS 3.8.8, Inverters-Shutdown	This specification requires the Class 1E UPS inverters to be operable to support onsite Class 1E 120 VAC vital bus

	electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems-Shutdown." This specification is applicable in MODES 5 and 6 and during movement of recently irradiated fuel assemblies. The Class 1E UPS inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS I&Cs so that the fuel, RCS, and containment design limits are not exceeded. The OPERABILITY of the minimum inverters to each 120-Vac Class 1E bus during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that: a. the unit can be maintained in the shutdown or refueling condition for extended periods; b. sufficient I&C capability is available for monitoring and maintaining the unit status; and c. adequate power is available to mitigate events postulated during shutdown, such as a FHA involving handling recently irradiated fuel. Due to radioactive decay, AC and DC inverters are only required to mitigate FHAs involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours). Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), assuring the operability of the inverters for safe operation of the facility when the reactor is in MODE 5 and 6 will no longer be applicable. The only DBA remaining is the FHA in the FHB which does not rely on the inverters for accident mitigation. Therefore, the inverters are not required during movement of irradiated fuel assemblies in the SFP for mitigation of a potential FHA. There are no active systems credited as part of the initial conditions of the analysis or as part of the primary success path for mitigation of the FHA in the FHB with DCPP Units 1 and 2 in the permanently shutdown and defueled condition.

	In addition, the mode of applicability associated with moving recently irradiated fuel will no longer be applicable. Therefore, this specification will no longer be needed for assuring the operability of the inverters for safe operation of the facility when moving recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours), and is proposed for deletion. The PDTS will not be implemented until 45 days after both Units have shutdown.
TS 3.8.9, Distribution Systems - Operating	This specification requires certain Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems to be OPERABLE in MODES 1, 2, 3, and 4. The required power distribution subsystems ensure the availability of Class 1E AC, DC, and 120-Vac bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA. The Class 1E AC, DC, and 120-Vac bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. Maintaining the Class 1E AC, DC, and 120-Vac bus electrical power distribution subsystems operable ensures that the redundancy incorporated into the design of ESF is not defeated. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), assuring the operability of certain electrical power distribution subsystems in MODE 1, 2, 3, and 4, will no longer be applicable. Therefore, TS 3.8.9 is proposed for deletion.
TS 3.8.10, Distribution Systems-Shutdown	This specification requires the necessary portion of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems to be operable to support equipment that is required to be operable, and is applicable in MODE 5, 6, and during movement of recently irradiated fuel assemblies. The Class 1E AC, DC, and 120-Vac bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that

	the fuel, RCS, and containment design limits are not exceeded. The OPERABILITY of the minimum Class 1E AC, DC, and 120-Vac bus electrical power distribution subsystems during MODE 5 and 6, and during movement of recently irradiated fuel assemblies ensures that: a. the unit can be maintained in the shutdown or refueling condition for extended periods; b. sufficient I&C capability is available for monitoring and maintaining the unit status; and c. adequate power is provided to mitigate events postulated during shutdown, such as a FHA involving handling recently irradiated fuel. Due to radioactive decay, AC and DC electrical power is only required to mitigate FHAs involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours). Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), assuring the operability of necessary portions of the electrical power distribution subsystems when the reactor is in MODE 5 and 6, will no longer be applicable. The only DBA remaining is the FHA in the FHB which does not rely on electrical distribution systems for accident mitigation. Therefore, the electrical distribution systems are not required during movement of irradiated fuel assemblies in the SFP for mitigation of a potential FHA. There are no active systems credited as part of the initial conditions of the analysis or as part of the primary success path for mitigation of the FHA in the FHB with DCPP Units 1 and 2 in the permanently shutdown and defueled condition. In addition, the mode of applicability associated with moving recently irradiated fuel is no longer applicable. Therefore, this specification will no longer be needed for assuring the operability of the electrical distribution systems for safe operation of the facility when moving recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours), and is proposed for deletion. The PDTS will

	not be implemented until 45 days after both Units have shut down.

TS Section 3.9 – REFUELING OPERATIONS TS Section 3.9, "Refueling Operations," contains LCO that provide for appropriate functional capability of parameters and equipment within containment that are required for mitigation of DBAs during refueling operations (moving fuel to or from the reactor core). The specifications of this section are proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), for DCPP Units 1 and 2, the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS Section 3.9 will not be required and these LCO (and associated SR) will not apply in a permanently defueled condition.	
TS 3.9.1, Boron Concentration	This specification requires the boron concentration of all filled portions of the RCS, refueling canal, and the refueling cavity, that have direct access to the reactor vessel, to be maintained within the limit specified in the COLR, and is applicable in MODE 6. The limit on the boron concentrations of the filled portions of the RCS, the refueling canal, and the refueling cavity, that have direct access to the reactor vessel, during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling. The refueling boron concentration is sufficient to maintain SDM with the most adverse conditions of fuel assembly and control rod position allowed by plant procedures. The boron concentration that is maintained in MODE 6 is sufficient to maintain keff less than or equal to 0.95 with the most reactive rod control assembly completely removed from its fuel assembly. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), assuring the boron concentration for filled portions of certain systems is within the limits in the COLR in MODE 6, will no longer be applicable. Therefore, TS 3.9.1 is proposed for deletion.

TS 3.9.3, Nuclear Instrumentation	This specification requires two source range neutron flux monitors to be operable in MODE 6. The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), ensuring operability of two source range neutron flux monitors in MODE 6, will no longer be applicable. Therefore, TS 3.9.3 is proposed for deletion.
TS 3.9.4, Containment Penetrations	This specification requires the containment penetrations to be in a specific status and is applicable during core alterations and during movement of irradiated fuel assemblies within containment. This LCO limits the consequences of a FHA in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed or capable of being closed. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), the FHA inside containment is no longer possible and this specification for containment penetrations will no longer be applicable. Therefore, TS 3.9.4 is proposed for deletion.

TS 3.9.5, Residual Heat Removal (RHR) and Coolant Circulation – High Water Level	This specification requires one RHR loop to be operable and in operation in MODE 6 with the water level greater than or equal to 23 ft above the top of the reactor vessel flange. The purpose of the RHR System in Mode 6 is to remove decay heat and sensible heat from the RCS to provide mixing of borated coolant and to prevent boron stratification. Only one RHR loop is required for decay heat removal in MODE 6, with the water level greater than or equal to 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be operable, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability of one RHR loop in MODE 6, will no longer be applicable. Therefore, TS 3.9.5 is proposed for deletion.
TS 3.9.6, Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level	This specification requires two RHR loops to be operable and one RHR loop to be in operation in MODE 6 with the water level less than 23ft above the top of the reactor vessel flange. The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the RCS, to provide mixing of borated coolant, and to prevent boron stratification. In MODE 6, with the water level less than 23 ft above the top of the reactor vessel flange, both RHR loops must be operable. Additionally, one loop of RHR must be in operation in order to provide removal of decay heat, mixing of borated coolant to minimize the possibility of criticality, and provide an indication of reactor coolant temperature. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), operability/operation of RHR loops in MODE 6, will no longer be applicable. Therefore, TS 3.9.6 is proposed for deletion.

TS 3.9.7, Refueling Cavity Water Level	The purpose of this specification is to ensure the refueling cavity water level is greater than or equal to 23 ft above the top of the reactor vessel flange during movement of irradiated fuel assemblies within containment. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and the SFP. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a FHA inside containment. A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated FHA inside containment are within acceptable limits. Since PG&E will no longer be authorized to operate the DCPP Unit 1 and Unit 2 reactors or place or retain fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2), maintaining water level in the refueling cavity, during movement of irradiated fuel assemblies within containment, will no longer be applicable. Therefore, TS 3.9.7 is proposed for deletion.

TS Section 4.0 – DESIGN FEATURES

TS Section 4.0, "Design Features," contains descriptions and requirements for those features of the facility such as materials of construction and geometric arrangements which, if altered or modified, could have a significant effect on safety and are not covered in the previous sections of the TS.

Once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), for DCPP Units 1 and 2, the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore, the design features that do not apply in a defueled condition are being proposed for deletion. TS 4.1 and revised TS 4.3 remain applicable with the reactor permanently defueled. As such, these TS sections are retained to reflect a permanently defueled condition.

Current DCPP TS	Proposed DCPP TS
<u>TS 4.3 Fuel Storage</u> 4.3.1.1 The permanent spent fuel pool storage racks are designed and shall be maintained with: a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent; b. k_{eff} < 1.0 if fully flooded with	<u>TS 4.3 Fuel Storage</u> 4.3.1.1 The permanent spent fuel pool storage racks are designed and shall be maintained with: a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent; b. k_{eff} < 1.0 if fully flooded with

unborated water, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the FSAR; c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 806 ppm, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the FSAR;	unborated water, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the *D*~~F~~SAR; c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 806 ppm, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the *D*~~F~~SAR;
TS 4.3 Fuel Storage 4.3.3 Capacity The permanent spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1324 fuel assemblies. For cycles 14-16, the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 154 fuel assemblies. For cycles 14-16, the total combined spent fuel pool capacity in the permanent and cask pit storage racks is limited to no more than 1478 fuel assemblies.	TS 4.3 Fuel Storage 4.3.3 Capacity The permanent spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1324 fuel assemblies. ~~For cycles 14-16, the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 154 fuel assemblies. For cycles 14-16, the total combined spent fuel pool capacity in the permanent and cask pit storage racks is limited to no more than 1478 fuel assemblies.~~

Basis
TS 4.3.1.1 – TS Section 4.3.1.1.b and 4.3.1.1.c are revised to reflect the conversion of the "Final Safety Analysis Report" to the "Defueled Safety Analysis Report" upon implementation of this LAR. The proposed terminology is consistent with a decommissioning plant. TS 4.3.3 – The proposed revision removes the historical discussion related to cycles 14-16 when PG&E was approved to use the cask pit storage rack to temporarily expand the storage capacity for the SFP. This discussion is historical and is not applicable in the permanently defueled condition.

Current DCPP TS	**Basis for Deletion**
TS 4.2, Reactor Core	TS 4.2.1 specifies the number and material of the fuel assemblies. TS 4.2.2 specifies the number and material for the control rod assemblies. Once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), for DCPP Units 1 and 2, the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore, TS 4.2 is not

	applicable in a permanently defueled condition and is proposed for deletion.
TS 4.3.1.2	TS 4.3.1.2 requires the new fuel storage racks to be designed and maintained with: a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent; b. keff ≤ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1.1 of the FSAR; c. keff ≤ 0.98 if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1.1.1 of the FSAR; and d. A nominal 22 inch center to center distance between fuel assemblies placed in the storage racks. Once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), for DCPP Units 1 and 2, the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Accordingly, PG&E will not be receiving or storing new fuel in the permanently defueled condition. Therefore, TS 4.3.1.2 is not applicable in a permanently defueled condition and is proposed for deletion.
TS 4.3.1.3	TS 4.3.1.3 requires that for cycles 14-16, the cask pit storage rack is designed and maintained with: a. Fuel assemblies having a maximum U-235 enrichment of 4.1 weight percent; b. k_{eff} < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties; c. k_{eff} ≤ 0.95 if fully flooded with water borated to 800 ppm, which includes an allowance for uncertainties; d. A nominal 9 inch center to center distance between fuel assemblies placed in the cask pit fuel storage rack; e. Fuel assemblies with discharge burnup in the "acceptable" region of Figure 3.7.17-4; f. Fuel assemblies having a 10 year minimum decay time since being discharged from the reactor; and g. A neutron absorbing material (Metamic™) between the stored fuel assemblies This TS is historical and is not applicable in the permanently defueled condition. Therefore, TS 4.3.1.3 is proposed for deletion.

Proposed Changes to TS Section 5.0, Administrative Controls

The existing TS Section 5.0, “Administrative Controls,” contains provisions relating to organization and management, procedures, programs and reporting requirements necessary to ensure operation of the facility in a safe manner.

Once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), for DCPP Units 1 and 2, the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2). Therefore, the administrative controls that do not apply in a defueled condition are being proposed for deletion or revised to reflect a permanently defueled condition.

5.1 Responsibility

Current DCPP TS	Proposed DCPP TS
TS 5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence. The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.	TS 5.1.1 The plant manager shall be responsible for overall ~~unit~~ *facility* operation and shall delegate in writing the succession to this responsibility ~~during his absence~~ *when absent*. The plant manager or ~~his~~ designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect ~~nuclear safety~~ *the safe handling and storage of nuclear fuel*.
TS 5.1.2 The Shift Foreman (SFM) shall be responsible for the control room command function. During any absence of the SFM from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SFM from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.	TS 5.1.2 The *Shift Supervisor* ~~Shift Foreman (SFM)~~ shall be responsible for the ~~control room~~ *shift* command function. ~~During any absence of the SFM from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SFM from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.~~

Basis
This section identifies the responsibilities for the CR command function associated with Modes of plant operation, and is based on personnel positions and qualifications for an operating plant. It identifies the need for a delegation of authority for command in an operating plant when the principal assignee leaves the CR. TS 5.1.1 - The term "unit" is changed to "facility." This is an administrative change that reflects that Units 1 and 2 will be permanently shut down and defueled. The term "facility" is a more appropriate description of a site that is undergoing decommissioning. This change is proposed throughout this LAR. In all cases that this change is made, overall management and staff responsibilities and the description of the facility are unchanged. The term "his" is deleted. This change is proposed to remove gender specific references from the TS wording because the position is gender neutral. The term "nuclear safety" is replaced with "the safe handling and storage of nuclear fuel." The term "the safe storage and handling of nuclear fuel" is considered analogous to "nuclear safety" for a facility in the permanently defueled condition. Proposed changes to replace "nuclear safety" with this term appropriately narrows the focus of nuclear safety concerns to the nuclear fuel. TS 5.1.2 - The proposed changes eliminate the MODE dependency for this function and personnel qualifications associated with an operating plant. The proposed change establishes the shift supervisor as having command of the shift. Delegation of command is unnecessary once Units 1 and 2 are in a permanently defueled condition with fuel in the SFP. Any event involving loss of SFP cooling would evolve slowly enough that no immediate response would be required to protect the health and safety of the public.

5.2 Organization	
Current DCPP TS	**Proposed DCPP TS**
TS 5.2.1, Onsite and Offsite Organizations Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant. a. Lines of authority, responsibility, and communication shall be defined and	TS 5.2.1, Onsite and Offsite Organizations Onsite and offsite organizations shall be established for *facility staff* ~~unit operation~~ and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting *the* safe~~ty~~ *storage and handling* of ~~the nuclear power plant~~ *spent nuclear fuel. The primary role of all nuclear workers is to protect the health and safety of the public.* a. Lines of authority, responsibility, and communication shall be defined and

established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the FSAR Update;	established throughout highest management levels, intermediate levels, and all ~~operating~~ *facility* organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the *DSAR*~~FSAR Update~~;
b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;	b. The plant manager shall be responsible for overall safe operation of the ~~plant~~ *facility* and shall have control over those onsite activities necessary for ~~safe operation and maintenance of the plant~~ *safe storage and handling of the nuclear fuel*;
c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and	c. A specified corporate officer shall have corporate responsibility for ~~overall plant nuclear safety~~ *the safe storage and handling of nuclear fuel* and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the ~~plant to ensure nuclear safety~~ *facility to ensure safe storage and handling of nuclear fuel*; and
d. The individuals who train the operating staff carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have	d. The individuals who train the ~~operating staff~~ *CERTIFIED FUEL HANDLERS,* carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals

sufficient organizational freedom to ensure their independence from operating pressures.	shall have sufficient organizational freedom to ensure their ~~independence from operating pressures~~ *ability to perform their assigned functions*.
Basis	
The introduction to this section identifies that organizational positions are established that are responsible for the safety of the nuclear power plant. This is changed to require that positions be established for the safe handling and storage of nuclear fuel. This change removes the implication that Units 1 and 2 can return to operation once the certifications required by 10 CFR 50.82(a)(1) are docketed. The term "unit operation" is changed to "facility staff". In addition, the term "plant" is changed to "facility" in several locations. These are administrative changes that reflect that Units 1 and 2 will be permanently shut down and defueled. The term facility is a more appropriate description of a site that is undergoing decommissioning. This change is proposed throughout this LAR. In all cases that this change is made, overall management and staff responsibilities and the description of the facility are unchanged. The term "the safe storage and handling of spent nuclear fuel" is considered analogous to "nuclear safety" for a facility in the permanently defueled condition. Proposed changes to replace "nuclear safety" with this term appropriately narrows the focus of nuclear safety concerns to the nuclear fuel. The introduction section is also updated to add a general statement regarding the primary role of all nuclear workers, which is to protect the health and safety of the public. This change is administrative in nature. TS 5.2.1.a – The term "operating organization" is changed to "facility organization." This change is administrative and more accurately describes the shutdown and defueled condition of Units 1 and 2. The term FSAR Update is revised to DSAR to reflect the transition to the DSAR upon implementation of this LAR. TS 5.2.1.b - This section identifies the organizational position responsible for the safe operation of the plant, and for control of activities necessary for the safe operation and maintenance of the plant. To reflect the change in safety concerns from an operating plant to a permanently defueled facility, the responsibility for control of activities necessary for the safe operation and maintenance of the facility is changed to the responsibility for safe storage and handling of the nuclear fuel. The change from "plant" to "facility" is administrative.	

TS 5.2.1.c - This section identifies the organizational position responsible for overall nuclear plant safety.

To reflect the change in safety concerns from an operating plant to a permanently defueled facility, the proposed changes update the responsibility from "overall plant nuclear safety" to "safe storage and handling of nuclear fuel," and the responsibility for providing technical support to "the plant to ensure nuclear safety" is changed to "the facility to ensure safe management of nuclear fuel."

TS 5.2.1.d - This TS addresses the requirement for organizational independence of the personnel who train the operations staff, health physics personnel and QA personnel from operating pressures.

This is changed to replace "operating staff" with "CERTIFIED FUEL HANDLERS" and to replace "their independence from operating pressures" to "their ability to perform their assigned functions." These changes reflect the changed function of the previous operating staff to a focus on safe handling and storage of nuclear fuel, and to remove the implication that Units 1 and 2 can return to operation once the certifications required by 10 CFR 50.82(a)(1) are docketed.

Current DCPP TS	**Proposed DCPP TS**
TS 5.2.2, Unit Staff The unit staff organization shall include the following: a. A non-licensed operator shall be assigned to each reactor containing fuel with a total of three non-licensed operators required for both units. b. Shift crew composition may be one less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence	TS 5.2.2, ~~Unit~~ *Facility* Staff The *facility* ~~unit~~ staff organization shall include the following: a. ~~A non-licensed operator shall be assigned to each reactor containing fuel with a total of three non-licensed operators required for both units.~~ *Each on duty shift shall be composed of at least one Shift Supervisor shared between Units 1 and 2, and one NON-CERTIFIED OPERATOR per unit. The NON-CERTIFIED OPERATOR position may be filled by a CERTIFIED FUEL HANDLER.* b. *Except for the Shift Supervisor,* ~~Shift~~ *shift* crew composition may be ~~one~~ less than the minimum requirement of ~~10 CFR 50.54(m)(2)(i) and~~ 5.2.2.a ~~and 5.2.2.f~~ for a period of time not to

of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.	exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements *and all of the following conditions are met*: *1) No fuel movements are in progress* *2) No movement of loads over fuel are in progress; and* *3) No unmanned shift positions during shift turnover shall be permitted while the shift crew is less than the minimum.*
c. A health physics technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.	c. A health physics technician shall be on site ~~when fuel is in the reactor~~ *during fuel handling operations and during movement of heavy loads over the fuel storage racks*. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.
e. The operations manager shall either hold a senior reactor operator license, have at one time held a senior reactor operator license for a pressurized water reactor, or be certified to a senior reactor operator equivalent level of knowledge. If the operations manager does not hold a senior reactor operator license, the person assigned to the Operations middle manager position shall hold a senior reactor operator license.	e. ~~The operations manager shall either hold a senior reactor operator license, have at one time held a senior reactor operator license for a pressurized water reactor, or be certified to a senior reactor operator equivalent level of knowledge. If the operations manager does not hold a senior reactor operator license, the person assigned to the Operations middle manager position shall hold a senior reactor operator license.~~ *The*

	Shift Supervisor shall be a CERTIFIED FUEL HANDLER.
f. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This position shall be manned in MODES 1, 2, 3, and 4 unless an individual with a SRO license meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.	f. ~~An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This position shall be manned in MODES 1, 2, 3, and 4 unless an individual with a SRO license meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.~~ *At least one person qualified to stand watch in the control room (NON-CERTIFIED OPERATOR or CERTIFIED FUEL HANDLER) shall be present in the control room when nuclear fuel is stored in a spent fuel pool.* *g. Oversight of fuel handling operations shall be provided by a CERTIFIED FUEL HANDLER.*

Basis

As discussed above, the change from "unit" to "facility" in the title of this section is administrative.

TS 5.2.2.a - This TS stipulates the number of non-licensed operators required when fuel is in the reactor.

Upon docketing of the certifications required by 10 CFR 50.82(a)(1) the licenses for Units 1 and 2 no longer authorize fuel to be in the reactors and therefore these requirements no longer apply. The TS requirement for non-licensed operators is replaced by Certified Fuel Handers and Non-Certified Operators. The minimum crew composition is appropriate for the safe management of SNF at a permanently defueled facility. The proposed TS staffing requirements of one Certified Fuel Handler shared between Units and one Non-Certified Operator per Unit align with the requirements in 10 CFR 50.54(m) requirements for a two-unit, single control facility with neither unit operating (one SRO and two Reactor Operators).

TS 5.2.2.b - This TS addresses the conditions under which the minimum shift compliment may be reduced. It allows for shift crew composition to be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

The reference to 10 CFR 50.54(m)(2)(i) is removed, because Units 1 and 2 will not return to operation once the certifications required by 10 CFR 50.82(a)(1) are docketed, and the requirement for licensed operating personnel will no longer be required to protect the health and safety of the public. No exemption request from 10 CFR 50.54(m)(2)(i) is needed or requested to support this change, based on the NRC's response to a similar request from Vermont Yankee Nuclear Power Station in June 2014 (Reference 24).

<u>TS 5.2.2.c</u> - This TS establishes the requirement for a health physics technician to be onsite when fuel is in the reactor. This TS allows for the position to be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

PG&E proposes to revise the condition of this TS so that a health physics technician is present onsite during movement of fuel and during the movement of loads over fuel, because fuel will not be able to be placed or stored in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) are docketed.

<u>TS 5.2.2.e</u> - This TS establishes the requirement for the Operations Manager or Operations Middle Manager to hold a SRO license.

PG&E proposes to revise this TS with a requirement that the shift supervisor be a Certified Fuel Handler. Once the certifications required by 10 CFR 50.82(a)(1) have been docketed, the requirements of 10 CFR 50.54(m) will no longer be applicable because the Part 50 licenses will no longer authorize operation of the reactors or placement of fuel in the reactor vessels. These certifications remove the need for the operator's licenses specified in 10 CFR Part 55. Therefore, there is no longer a need for operations management staff to hold an SRO license. Replacing this with a requirement that the shift supervisor be a Certified Fuel Handler ensures that the senior individual on shift is appropriately trained and qualified in accordance with the Certified Fuel Handler Training and Retraining Program, to supervise shift activities. As discussed above, no exemption from 10 CFR 50.54(m) is needed or requested to support this change.

The PG&E management structure will not require positions above the shift supervisor to be a Certified Fuel Handler or attend equivalent training. Once the plant is permanently shut down and defueled, the time available to mitigate credible events is expected to be greater than that for current design basis events. As such, management oversight of the facility can be performed by individuals meeting the applicable requirements of American National Standards Institute/American Nuclear Society 3.1-1978 (as required by the QA Program) and need not be qualified as Certified Fuel Handlers.

TS 5.2.2.f - This TS establishes the requirement to have an individual provide advisory technical support during MODE 1, 2, 3, and 4 unless an individual with an SRO license meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

Once the certifications required by 10 CFR 50.82(a)(1) have been docketed, the 10 CFR Part 50 licenses no longer authorize operation of the reactors or placement of fuel in the reactor vessels. As a result, TS 5.2.2.f will not apply.

This TS is changed to reflect the requirement for having either a Certified Fuel Handler or Non-Certified Operator in the CR when fuel is stored in the SFP. This reflects the reduced requirement for CR personnel training and qualification for a plant authorized for nuclear fuel storage only. There will be a sufficient number of individuals qualified as Certified Fuel Handlers to staff the plant twenty-four hours a day, seven days a week. Additional on-shift staffing will be provided to satisfy applicable security, fire protection, and emergency preparedness requirements.

The CR will remain the physical center of the command function. However, since control of activities may be performed either remotely from the CR or locally in the facility, the location of the command center is functionally where the shift supervisor is located in accordance with TS 5.1.2.

All spent fuel handling activities are performed locally at the SFP. Indications and/or alarms are also received in the CR that would be indicative of SFP abnormalities. The shift supervisor is responsible for directing response to those abnormalities, from either the CR or local to the SFP in accordance with applicable response procedures.

For any conditions, incidents, or events that occur when the NON-CERTIFIED OPERATOR is in the CR alone and are not within the scope of qualifications that are possessed by the NON-CERTIFIED OPERATOR, the shift supervisor will be immediately contacted for direction by phone, radio, and/or the plant pager system. This philosophy is deemed acceptable because the necessity to render immediate actions to protect the health and safety of the public is not challenged.

Added TS 5.2.2.g - This TS establishes the requirement for having oversight of fuel handling operations performed by a CERTIFIED FUEL HANDLER. This new requirement ensures movement of SNF is only performed under the oversight of an individual who has been trained and qualified on the procedures, processes, requirements, and standards for safe movement of SNF. Oversight of fuel handling operations refers to the authorization from the shift supervisor/CERTIFIED FUEL HANDLER to move fuel, because proposed TS 5.2.2.e requires the shift supervisor to be a CERTIFIED FUEL HANDLER.

5.3 Unit Staff Qualifications	
Current DCPP TS	**Proposed DCPP TS**
TS 5.3, Unit Staff Qualifications 5.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications referenced for comparable positions as specified in the updated FSAR, Chapter 17, Quality Assurance.	TS 5.3, ~~Unit~~ *Facility* Staff Qualifications 5.3.1 Each member of the ~~plant~~ *facility* staff shall meet or exceed the minimum qualifications referenced for comparable positions as specified in the ~~updated FSAR, Chapter 17, Quality Assurance~~ *Quality Assurance Program*. *5.3.2 A training and retraining program for CERTIFIED FUEL HANDLERs shall be maintained.*
Basis	

As discussed above the change from "unit" to "facility" in the title of this section is administrative.

TS 5.3.1 - This TS specifies the location of the minimum qualifications for the staff. The change from "plant staff" to "facility staff" is an administrative change. The additional change deletes the specific reference to UFSAR Chapter 17, which removes the requirement to submit a LAR if the QA Plan is relocated from UFSAR Chapter 17.

TS 5.3.2 - PG&E proposes to add a new TS 5.3.2 to require that a training and retraining program for the Certified Fuel Handlers shall be maintained. The Certified Fuel Handler Training and Retraining Program ensures that the qualifications of CERTIFIED FUEL HANDLERs are commensurate with the tasks to be performed and the conditions requiring response. 10 CFR 50.120, "Training and qualification of nuclear power plant personnel," requires training programs to be derived using systems approach to training (SAT) as defined in 10 CFR 55.4. Although the requirements of 10 CFR 50.120 apply to holders of an operating license issued under 10 CFR Part 50, and the Units 1 and 2 licenses will no longer authorize operation following docketing of the certifications required by 10 CFR 50.82(a)(1), the Certified Fuel Handler Training and Retraining Program nonetheless will align with those requirements. The Certified Fuel Handler Training and Retraining Program will provide adequate confidence that appropriate SAT based training of personnel who will perform the duties of a CERTIFIED FUEL HANDLER is conducted to ensure the facility is maintained in a safe and stable condition.

As previously specified, this LAR will not be implemented until a Certified Fuel Handler Training and Retraining Program has been implemented in accordance with 10 CFR 50.2.

<table>
<tr><th colspan="2">5.4 Procedures</th></tr>
<tr><th>Current DCPP TS</th><th>Proposed DCPP TS</th></tr>
<tr><td><u>TS 5.4.1</u>
Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the applicable requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33 and response to the subject NUREGs;</td><td><u>TS 5.4.1</u>
Written procedures shall be established, implemented, and maintained covering the following activities:

a. The ~~applicable~~ procedures <u>applicable to the safe storage of spent nuclear fuel,</u> recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

~~b. The emergency operating procedures required to implement the applicable requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33 and response to the subject NUREGs;~~ <u>Not Used;</u></td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This TS provides a description and requirements regarding administration of written procedures. TS 5.4 will remain applicable with the reactor permanently defueled. As such, it is retained and revised to reflect a permanently defueled condition. Relevant procedures, drawings and instructions will continue to be controlled per 10 CFR 50, Appendix B, Criterion VI, "Document Control." Activities involving security and emergency planning and preparedness will continue to be controlled by procedure.

<u>TS 5.4.1.a</u> - PG&E proposes to revise the applicability for this TS to procedures applicable to the safe storage of SNF recommended in RG 1.33, Revision 2, Appendix A. Since operating and refueling the reactors will both be prohibited by the 10 CFR Part 50 licenses once the certifications required by 10 CFR 50.82(a)(1) have been docketed, procedures associated with these activities will no longer need to be maintained. Procedures governing fuel handling operations will provide the guidance necessary to ensure safe handling of spent fuel in the SFP and transfer from the SFP to dry fuel storage casks. Procedures governing responses to FHAs, personnel injuries, SFP events and external events provide the necessary guidance to mitigate the consequences of such events.

<u>TS 5.4.1.b</u> - This TS requires emergency operating procedures that implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33. This TS is proposed to be deleted as Generic Letter 82-33 was only addressed to licensees of operating reactors, applicants for operating licenses, and holders of construction permits, none of which will apply to DCPP Units 1 and 2 in the permanently defueled condition. As discussed above, procedures governing the site response to accidents, events, and injuries will provide the necessary guidance to mitigate</td></tr>
</table>

the consequences of such events. In addition, refer to justification for deletion of License Condition 2.C.(6) for Unit 1 and License Condition 2.C.(5) for Unit 2, NUREG-0737 Conditions. There are no changes proposed to TS 5.4.1.c, TS 5.4.1.d or TS 5.4.1.e.

TS Section 5.5 Programs and Manuals Consistent with the current format for TS Section 5.5, programs that are proposed for deletion are revised to read "not used." Refer to the markups in Attachment 3.	
Current DCPP TS	**Proposed DCPP TS**
<u>TS 5.5.4 Radioactive Effluent Control Program</u> This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:	<u>TS 5.5.4 Radioactive Effluent Control Program</u> This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:
d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;	d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from ~~each unit~~ <u>*the facility*</u> to unrestricted areas, conforming to 10 CFR 50, Appendix I;
h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;	h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from ~~each unit~~ <u>*the facility*</u> to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;
i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in	i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in

particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;	particulate form with half lives > 8 days in gaseous effluents released from ~~each unit~~ <u>*the facility*</u> to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;
Basis	
<u>TS 5.5.4.d Radioactive Effluent Control Program</u> - The term "from each unit" is changed to "from the facility." This is an administrative change that reflects that Units 1 and 2 will be permanently shut down and defueled. The term "facility" is a more appropriate description of a site that is undergoing decommissioning. <u>TS 5.5.4.h Radioactive Effluent Control Program</u> - The term "from each unit" is changed to "from the facility." This is an administrative change that reflects that Units 1 and 2 will be permanently shut down and defueled. The term "facility" is a more appropriate description of a site that is undergoing decommissioning. <u>TS 5.5.4.i Radioactive Effluent Control Program</u> - The term "from each unit" is changed to "from the facility." This is an administrative change that reflects that Units 1 and 2 will be permanently shut down and defueled. The term "facility" is a more appropriate description of a site that is undergoing decommissioning.	

Current DCPP TS	Basis for Deletion
TS 5.5.2, Primary Coolant Sources Outside Containment	This program was established to minimize leakage from portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. Once the plant is permanently shut down and defueled, there will no longer be any transient or accident conditions associated with primary coolant sources. Therefore, TS 5.5.2 is proposed for deletion.
TS 5.5.5, Component Cyclic or Transient Limit	This program was established to provide controls to track the cyclic and transient occurrences to ensure that the reactor vessel is maintained within design limits. Pursuant to 10 CFR 50.82(a)(2), once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels. The components monitored will no longer be subjected to cycles or transients after permanent shutdown. Therefore, TS 5.5.5 is proposed for deletion.

TS 5.5.7, Reactor Coolant Pump Flywheel Inspection Program	This program was established for the inspections of each RCP flywheel. Pursuant to 10 CFR 50.82(a)(2), once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels. The RCPs pertain only to reactor support systems that do not apply in a permanently defueled condition. Therefore, TS 5.5.7 is proposed for deletion.
TS 5.5.8, Inservice Testing Program	The purpose of the TS 5.5.8, “Inservice Testing Program,” is to demonstrate the operational readiness of Code Class 1, 2, and 3 components which are required to perform a specific function in shutting down the reactor to the safe shutdown condition, in maintaining the safe shutdown condition or in mitigating the consequences of an accident. Pursuant to 10 CFR 50.82(a)(2), once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels. In addition, the only remaining accident does not credit any Code Class 1, 2, or 3 components for mitigating the consequences. Therefore, TS 5.5.8 will no longer be needed and is proposed for deletion.
TS 5.5.9, Steam Generator (SG) Tube Inspection Program	This program was established to ensure that SG tube integrity is maintained. Pursuant to 10 CFR 50.82(a)(2), once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels. The SGs pertain only to reactor support systems that do not apply in a permanently defueled condition. Therefore, TS 5.5.9 is proposed for deletion.

TS 5.5.10, Secondary Water Chemistry Program	This program was established to provide controls for monitoring secondary water chemistry to inhibit SG tube degradation. Pursuant to 10 CFR 50.82(a)(2), once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels. The components that the program is established to protect are associated with reactor operation. Therefore, TS 5.5.10 is proposed for deletion.
TS 5.5.11, Ventilation Filter Testing Program (VFTP)	This program was established to implement the required testing of the ESF filter ventilation systems (CR, Auxiliary Building, and FHB). The accident analysis applicable to the permanently defueled condition does not rely on ventilation filters for accident mitigation. In addition, as previously discussed, TS 3.7.10, 3.7.12 and 3.7.13 which provided the operability requirements for the CR, Auxiliary Building, and FHBVS are proposed for deletion. Therefore, TS 5.5.11 will no longer be required and is proposed for deletion.
TS 5.5.13, Diesel Fuel Oil Testing Program	This program was established to implement the required testing of both new fuel oil and stored fuel oil used to supply the EDGs. The TS that provided operability requirements for the EDGs are proposed for elimination. Therefore, TS 5.5.13 will no longer be required and is proposed for deletion.
TS 5.5.15, Safety Function Determination Program (SFDP)	This program was established to ensure a loss of safety function is detected and appropriate actions taken. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. This program implements the requirements of LCO 3.0.6. As discussed above, the only DBA remaining is the FHA in the FHB. There are no active systems credited as part of the initial conditions of the analysis or as part of the primary success path for mitigation of the FHA in the FHB with DCPP Units 1 and 2 in the permanently shutdown and defueled condition. Therefore, the SFDP will no longer be required and TS 5.5.15 is proposed for deletion. In addition, LCO 3.0.6 is proposed for deletion.

TS 5.5.16, Containment Leakage Rate Testing Program	This program was established to implement the leakage rate testing of containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. Pursuant to 10 CFR 50.82(a)(2), once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels. This program pertains to verifying the operability of the containment systems. As described above, the requirements for containment systems (TS Section 3.6) are proposed for deletion. Therefore, TS 5.5.16 will no longer be required and is proposed for deletion.
TS 5.5.17, Battery Monitoring and Maintenance Program	This program was established to provide battery restoration and maintenance. As described above, the TS for station batteries is proposed for deletion. In addition, the accident analysis applicable to the permanently defueled condition does not rely on batteries for accident mitigation. Therefore, this program will no longer be required and TS 5.5.17 is proposed for deletion.
TS 5.5.18, Surveillance Frequency Control Program	This program provides controls for surveillance frequencies. The program ensures that SRs specified in the TS are performed at intervals sufficient to assure the associated LCOs are met. The remaining TS LCO proposed in the PDTS contain two SRs included in the Surveillance Frequency Control Program. There will be no further need to maintain this program and therefore TS 5.5.18 is proposed for deletion.
TS 5.5.19, Control Room Envelope Habitability Program	This program was established and implemented to ensure that control room envelope (CRE) habitability is maintained such that, with an operable CRVS, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under DBA conditions without personnel receiving radiation

	exposures in excess of 5 rem TEDE for the duration of the accident. With both units in a permanently defueled condition, the only postulated accident that remains applicable is the FHA in the FHB. As discussed above, the revised FHA analysis does not credit or require the use of the CR for mitigation. This analysis supports the elimination of TS requirements for the operability of CRVSs (TS 3.7.10). In addition, once PG&E dockets the certifications required by 10 CFR 50.82(a)(1), the licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors or placement or retention of fuel in the reactor vessels. Therefore, TS 5.5.19 will no longer be required and is proposed for deletion.

TS Section 5.6 Reporting Requirements Consistent with the current format for TS Section 5.5, programs that are proposed for deletion are revised to read "not used". Refer to the markups in Attachment 3.	
Current DCPP TS	**Proposed DCPP TS**
<u>TS 5.6.2 - Annual Radiological Environmental Operating Report</u> -----------------------NOTE------------------------ A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station. --- The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year.......	<u>TS 5.6.2 - Annual Radiological Environmental Operating Report</u> -----------------------NOTE----------------------- A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station. --- The Annual Radiological Environmental Operating Report covering the operation of the ~~unit~~ <u>*facility*</u> during the previous calendar year shall be submitted by May 1 of each year.......
<u>TS 5.6.3 - Radioactive Effluent Release Report</u> -----------------------NOTE------------------------ A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall	<u>TS 5.6.3 - Radioactive Effluent Release Report</u> -----------------------NOTE----------------------- A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the

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

Basis

TS 5.6.2 - Annual Radiological Environmental Operating Report - PG&E proposes to change the term "unit" to "facility" in this TS. This is an administrative change that reflects that Units 1 and 2 will be permanently shut down and defueled. The term "facility" is a more appropriate description of a site that is undergoing decommissioning.

TS 5.6.3 - Radioactive Effluent Release Report - PG&E proposes to change the term "unit" to "facility" in this TS. This is an administrative change that reflects that Units 1 and 2 will be permanently shut down and defueled. The term "facility" is a more appropriate description of a site that is undergoing decommissioning.

Current DCPP TS	Basis for Deletion
5.6.5, Core Operating Limits Report (COLR)	This TS establishes a requirement to determine and document the core operating limits prior to each reload cycle or prior to any remaining portion of a reload cycle. This report is submitted to NRC upon issuance for each reload cycle. Once the certifications required by 10 CFR 50.82(a)(1) have been docketed, the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement of fuel in the reactor vessels. Therefore, TS 5.6.5 will no longer be required and is proposed for deletion.
TS 5.6.6, Reactor Coolant System (RCS) Pressure	This TS establishes a requirement to determine and document RCS pressure and temperature limits for heat-up, cooldown, low-temperature operation, criticality,

and Temperature Limits Report (PTLR)	hydrostatic testing, LTOP arming, and PORV lift settings as well as heatup and cooldown rates. This report is required to be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Once the certifications required by 10 CFR 50.82(a)(1) have been docketed, the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement of fuel in the reactor vessels. Therefore, TS 5.6.6 will no longer be required and is proposed for deletion.
TS 5.6.8, PAM Report	This TS establishes a requirement to submit a report to the NRC within 14 days following entry into Condition B or F of LCO 3.3.3, “Post Accident Monitoring (PAM) Instrumentation.” As described above, LCO 3.3.3 will no longer be required and is proposed for deletion. Therefore, TS 5.6.8 will no longer be required and is also proposed for deletion.
TS 5.6.10, Steam Generator (SG) Tube Inspection Report	This TS establishes a requirement to submit a report within 180 days of entry into Mode 4 following completion of an inspection performed in accordance with TS 5.5.9, “Steam Generator (SG) Program.” As described above, TS 5.5.9 is proposed for deletion. Once the certifications required by 10 CFR 50.82(a)(1) have been docketed, the 10 CFR Part 50 licenses will no longer authorize operation of the reactors or placement of fuel in the reactor vessels. Therefore, the SG tubes will not be subjected to the temperature and pressure effects that the SG Program and associated inspection report were put in place to monitor and assess. In addition, as described above TS 5.5.9, is proposed for deletion. Therefore, TS 5.6.10 will no longer be required and is proposed for deletion.

TS Section 5.7 High Radiation Area	
Current DCPP TS	**Proposed DCPP TS**
TS 5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation:</u> a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition: 1. All such door and gate keys shall be maintained under the administrative control of the shift manager, radiation protection manager, or his or her designee. 2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.	TS 5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation:</u> a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition: 1. All such door and gate keys shall be maintained under the administrative control of the shift ~~manager~~<u>*supervisor*</u>, radiation protection manager, or his or her designee. 2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.
Basis	
TS 5.7.2.a - PG&E proposes to change "shift manager" to "shift supervisor." The shift supervisor will be the on-shift position which may have door and gate keys for high radiation areas under administrative control, in accordance with TS 5.7.2.	

3. REGULATORY ANALYSIS

3.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. PG&E has determined that the proposed changes do not require any exemptions or relief from regulatory requirements.

3.1.1 10 CFR 50.82, Termination of License

The portions of 10 CFR 50.82 providing the basis for this LAR are:

> (a) For power reactor licensees—
>
> (1) (i) When a licensee has determined to permanently cease operations the licensee shall, within 30 days, submit a written certification to the NRC, consistent with the requirements of § 50.4(b)(8);
>
> (ii) Once fuel has been permanently removed from the reactor vessel, the licensee shall submit a written certification to the NRC that meets the requirements of § 50.4(b)(9) and;
>
> (2) Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, or when a final legally effective order to permanently cease operations has come into effect, the 10 CFR part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel.

In Reference 1, PG&E notified the U.S. NRC that the decision was made to permanently cease operations at DCPP Units 1 and 2 upon expiration of the FOL. The FOL for DCPP Unit 1 expires on November 2, 2024, and the FOL for DCPP Unit 2 expires on August 26, 2025. After docketing the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(i) and (ii) and pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors, or emplacement or retention of fuel in the reactor vessels. As a result, PG&E will be authorized to only possess SNM.

3.1.2 10 CFR 50.36, Technical Specifications

In 10 CFR 50.36, the Commission established the regulatory requirements related to the content of the TS. Pursuant to 10 CFR 50.36, TS are required to include items in the following five categories: (1) SLs, LSSS, and limiting control settings; (2) LCOs; (3)

SRs (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS.

10 CFR 50.36(c)(2)(ii)(A) through (D) provide criteria that require a licensee to include TS for certain items. These criteria and the applicability to DCPP Units 1 and 2 in a permanently defueled condition are discussed below:

> Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

When the DCPP Units 1 and 2 reactors are permanently defueled, the RCS will no longer be in operation. Therefore, this criterion is no longer applicable.

> Criterion 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.

The intent of this criterion is to capture process variables, design features, or operating conditions assumed in the safety analysis of an operating facility. Although the facility will no longer be in an operating mode, and the majority of the design basis events will no longer be applicable, there are still design basis events applicable to a plant authorized to handle, store, and possess nuclear fuel. The DBAs still applicable to DCPP Units 1 and 2 are discussed within this proposed amendment.

> Criterion 3. A structure, system, or component 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.

The intent of this criterion is to capture TS for those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to DBAs and transients limits the consequences of these events to within the appropriate acceptance criteria. While there are no transients that continue to apply to DCPP Units 1 and 2, there are still DBAs that apply to a plant authorized to only handle, store, and possess nuclear fuel. The DBAs still applicable to DCPP Units 1 and 2 are discussed within this proposed amendment.

> Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The intent of this criterion is to factor risk insights and operating experience into the TS LCOs. Risk is significantly reduced with the DCPP Units 1 and 2 reactors in a permanently defueled condition.

10 CFR 50.36(c)(5), "Administrative controls," states in part:

> Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

The particular administrative controls to be included in the TS generally are requirements the NRC deems necessary to support the safe operation of a facility, which are not already covered by other regulations. These requirements are predominately specified in support of an operating plant. Once DCPP Units 1 and 2 are in a permanently shutdown and defueled condition, certain administrative controls described in the TS will no longer apply and are proposed for deletion or revision.

10 CFR 50.36(c)(6), Decommissioning, states:

> This paragraph applies only to nuclear power reactor facilities that have submitted the certifications required by § 50.82(a)(1) and to non-power reactor facilities which are not authorized to operate. Technical specifications involving safety limits, limiting safety system settings, and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis.

As previously discussed, in Reference 1, PG&E notified the U.S. NRC that the decision was made to permanently cease operations at DCPP Units 1 and 2 upon expiration of the FOL. The FOL for DCPP Unit 1 expires on November 2, 2024, and the FOL for DCPP Unit 2 expires on August 26, 2025. After docketing the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessels in accordance with 10 CFR 50.82(a)(1)(i) and (ii) and pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors, or emplacement or retention of fuel in the reactor vessels. As a result, PG&E will be authorized to only possess SNM. This proposed license amendment deletes DCPP Units 1 and 2 TS that will no longer be applicable in a permanently defueled condition, and modifies some of the remaining TS to correspond to the permanently shutdown condition.

3.1.3 10 CFR 50.48, Fire Protection

10 CFR 50.48(f) established the requirement for maintaining a fire protection program once licensees have submitted the certifications required by 10 CFR 50.82(a)(1):

> Licensees that have submitted the certifications required under §50.82(a)(1) shall maintain a fire protection program to address the potential for fires that could cause the release or spread of radioactive materials (i.e., that could result in a radiological hazard). A fire protection program that complies with NFPA 805 shall be deemed to be acceptable for complying with the requirements of this paragraph.

As previously discussed, in Reference 1, PG&E notified the U.S. NRC that the decision was made to permanently cease operations at DCPP Units 1 and 2 upon expiration of the FOL. Since the initial certification has been submitted pursuant to 10 CFR 50.82(a)(1)(i) and once the final certifications required by 10 CFR 50.82(a)(1)(ii) have been submitted, the requirements of 10 CFR 50.48(f) will be in full effect.

3.1.4 DCPP Units 1 and 2 Design Basis Accidents

The UFSAR describes the DBAs and transient scenarios applicable to DCPP Units 1 and 2. With the termination of reactor operations and the permanent removal of fuel from the reactors as certified in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and pursuant to 10 CFR 50.82(a)(2), the majority of the design basis scenarios postulated in the UFSAR will no longer be possible. During decommissioning, the SNF will initially be stored in the SFP, or the ISFSI. Eventually, all of the spent fuel will be stored at the ISFSI until it is shipped offsite in accordance with the scheduled provided in the Irradiated Fuel Management Plan and the PSDAR. The RCS, steam system, and turbine generator will no longer be in operation and will have no function related to the safe storage and management of the SNF. Table 2.1.1 lists the DBAs applicable to DCPP Units 1 and 2 after the units have been permanently defueled.

3.1.5 10 CFR 50.51, Continuation of License

10 CFR 50.51(b) states:

> (b) Each license for a facility that has permanently ceased operations, continues in effect beyond the expiration date to authorize ownership and possession of the production or utilization facility, until the Commission notifies the licensee in

writing that the license is terminated. During such period of continued effectiveness the licensee shall--

(1) Take actions necessary to decommission and decontaminate the facility and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition, and

(2) Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the provisions of the specific 10 CFR part 50 license for the facility.

3.1.6 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors”

10 CFR 50.46(a)(1)(i) states, "This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted."

3.1.7 10 CFR 50.62, “Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants”

10 CFR 50.62(a) states, "The requirements of this section apply to all commercial light-water-cooled nuclear power plants, other than nuclear power reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted."

3.1.8 10 CFR 50.54(m), “Minimum Requirements Per Shift for On-Site Staffing of Nuclear Power Units by Operators and Senior Operators Licensed Under 10 CFR Part 55”

10 CFR 50.54(m) establishes the requirements for having reactor operators and SROs licensed in accordance with 10 CFR 55 based on plant conditions. Upon permanent cessation of operations for DCPP Units 1 and 2, the requirements of this section will no longer apply once the certifications required by 10 CFR 50.82(a)(1) have been docketed, and it will be permissible to remove those positions from the TS.

3.2 No Significant Hazards Consideration Analysis

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," PG&E is proposing an amendment to FOL, Appendix A, TS, and Appendix D, Additional Conditions, for DCPP Units 1 and 2, FOL Numbers DPR-80 and DPR-82, respectively. The proposed LAR would revise the FOL, additional conditions, and TS to the PDTS consistent with the permanent cessation of power operation and permanent defueling of the reactors.

The proposed changes would revise and remove certain requirements contained within the FOL, additional conditions, and TS that are either historical in nature or would no longer be applicable once it has been certified that all fuel has been permanently removed from the reactors. Once the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessels are docketed, the 10 CFR Part 50 licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors, or emplacement or retention of fuel in the reactor vessels, pursuant to 10 CFR 50.82(a)(2).

PG&E has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below:

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

Response: No.

The proposed changes would not take effect until PG&E has submitted the certifications required by10 CFR 50.82(a)(1) for both Units 1 and 2, at least 45 days have passed since both Units have shut down, and a Certified Fuel Handler Training and Retraining Program has been implemented in accordance with 10 CFR 50.2. Because the 10 CFR Part 50 licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors, or emplacement or retention of fuel into the reactor vessels, as specified in 10 CFR 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation are no longer credible.

The remaining postulated DBA events that could potentially occur at a permanently defueled facility are a FHA in the FHB, a tank rupture, and external causes. The FHA analysis for DCPP Units 1 and 2, shows that after 45 days of decay time after the reactors have shut down and provided the SFP water level requirements of LCO 3.7.15 are met, the dose consequences are acceptable without relying on active SSCs to remain functional for accident mitigation during and following the event. The remaining DBAs that support the permanently shutdown and defueled condition do not rely on any active safety systems for mitigation.

The probability of occurrence of previously evaluated accidents is not increased, because safe storage and handling of SNF will be the only operations performed, and these activities are bounded by the existing analyses. Additionally, the occurrence of postulated accidents associated with reactor operation will no longer be credible with permanently defueled reactors. This significantly reduces the scope of applicable accidents.

The deletion of TS definitions and rules of usage and application requirements that will not be applicable in a defueled condition has no impact on facility SSCs or the methods of operation of such SSCs. The deletion of design features and SLs not applicable when DCPP Units 1 and 2 are permanently shut down and defueled has no impact on the remaining applicable DBAs.

The removal of LCO and SR that relate only to the operation of the nuclear reactors or the prevention, diagnosis, or mitigation of reactor-related transients or accidents do not affect the applicable DBAs previously evaluated because these accidents will no longer be applicable in the permanently defueled condition.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes to delete or modify the FOL, additional conditions, TS, and licensing basis have no impact on facility SSCs affecting the safe storage of SNF, or on the methods of operation of such SSCs, or on the handling and storage of SNF. The removal of TS that are related only to the operation of the nuclear reactors, or only to the prevention, diagnosis, or mitigation of reactor-related transients and accidents, cannot result in different or more adverse failure modes or accidents than previously evaluated because the reactors will be permanently shut down and defueled.

The proposed modifications and deletion of requirements in the DCPP Units 1 and 2 FOL, additional conditions, TS, and licensing basis do not affect systems credited in the accident analysis for the remaining credible accidents. The proposed license and PDTS will continue to require proper control and monitoring of safety significant parameter and activities. The TS regarding SFP water level, boron concentration, and SNF storage configurations are retained to preserve the current requirements for safe storage of SNF. The proposed amendment does not result in any new mechanisms that could initiate damage to the remaining relevant safety barriers for defueled plants. Since safe storage and handling of SNF will be the only operations allowed, and these activities are bounded by the existing analyses, the proposed changes do not create the possibility of a new or different kind of accident.

Therefore, the proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

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

Response: No.

The proposed changes include deleting or modifying certain FOL, additional conditions, TS, and licensing basis requirements once DCPP Units 1 and 2 have permanently shut down and defueled. Because the 10 CFR Part 50 licenses for DCPP Units 1 and 2 will no longer authorize operation of the reactors, or emplacement or retention of fuel into the reactor vessels, the occurrence of postulated accidents associated with reactor operation will no longer be credible. The remaining postulated DBA events that could potentially occur at a permanently defueled facility would be an FHA in the FHB, tank ruptures, and external causes. The proposed amendment does not adversely affect the inputs or assumption of any of the design basis analyses.

The proposed changes are limited to those portions of the FOL, additional conditions, TS, and licensing basis that are not related to the safe storage and handling of SNF. The requirements proposed to be revised or deleted from the FOL, additional conditions, TS and licensing basis are not credited in the updated applicable accident analyses for the remaining applicable postulated accidents, and as such, do not contribute to the margin of safety associated with the accident analysis. Postulated DBAs involving the reactors will no longer be possible because the reactors will be permanently shut down and defueled, and operation of DCPP Units 1 and 2 reactors will no longer be authorized.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluation, PG&E concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of “no significant hazards consideration” is justified.

3.3 Precedent

The NRC has reviewed and approved license and/or TS changes, similar to those being proposed in this LAR, for the following facilities:

- Davis-Besse Nuclear Power Station– Submittal and requests for additional information; however, the LAR was withdrawn prior to NRC approval (References 2, 3, and 4)
- Three Mile Island Nuclear Station (References 5 and 6)
- Oyster Creek Nuclear Generating Station (References 7 and 8
- Fort Calhoun Station (Reference 9)
- Crystal River Unit 3 Nuclear Generating Plant (Reference 10)

These submittals updated the FOL and/or the TS for these plants to reflect the permanently defueled status of the facilities.

3.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 ENVIRONMENTAL EVALUATION

PG&E has evaluated this LAR against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21.

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, or would change an inspection or SR. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be releases offsite, or (iii) a significant increase individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need to be prepared in connection with the proposed amendment.

5.0 REFERENCES

1. PG&E Letter DCL-18-096, “Certification of Permanent Cessation of Power Operations,” dated November 27, 2018 (ML18331A553)

2. FENOC Letter to NRC, “License Amendment Request - Proposed License and Associated Technical Specification Changes for Permanently Defueled Condition,” dated February 5, 2019 (ML19036A523)

3. FENOC Letter to NRC, “Response to Request for Additional Information Regarding License Amendment Request for Permanently Defueled Technical Specifications (EPID L-2019-LLA-0012),” dated June 26, 2019 (ML19177A289)

4. FENOC Letter to NRC, “License Amendment Request – Proposed Changes to Technical Specifications Sections 1.1, "Definitions," and 5.0, ‘Administrative Controls,’ for Permanently Defueled Condition,” dated October 22, 2018 (ML18295A289)

5. NRC Letter, “Three Mile Island Nuclear Station, Unit 1 – Issuance of Amendment No. 297 RE: Defueled Technical Specifications and Revised License Conditions (EPID L-2018-LLA-0204),” dated August 29, 2019 (ML19211D317)

6. NRC Letter, “Three Mile Island Nuclear Station, Unit 1 – Issuance of Amendment No. 295 Re: Changes to Technical Specifications 1.0,

'Definitions,' and 6.0, 'Administrative Controls,' for Permanently Defueled Condition (EPID L-2017-LLA-0384)," dated December 14, 2018 (ML18305B419)

7. NRC Letter, "Oyster Creek Nuclear Generating Station – Issuance of Amendment RE: License Amendment Request for Proposed Defueled Technical Specifications and Revised License Conditions for Permanently Defueled Condition (EPID L-2017-LLA-0395)," dated October 26, 2018 (ML18227A338)

8. NRC Letter, "Oyster Creek Nuclear Generating Station – Issuance of Amendment Regarding Changes to the Administrative Controls Section of the Technical Specifications (CAC NO. MF8108)," dated March 7, 2017 (ML16235A413)

9. NRC Letter, "Fort Calhoun Station, Unit 1 – Issuance of Amendment RE: Revised Technical Specifications to Align to Those Requirements for Decommissioning (CAC NO. MF9567; EPID L-2017-LLA-0192)," dated March 6, 2018 (ML18010A087)

10. NRC Letter, "Crystal River Unit 3 Nuclear Generating Plant – Issuance of Amendment for Permanently Shutdown and Defueled Operating License and Technical Specifications (TAC NO MF3089)," dated September 4, 2015 (ML15224B286)

11. NRC Letter, "Diablo Canyon Nuclear Power Plant, Units 1 and 2 – Issuance of Amendment Nos. 237 and 239 to Relocate Technical Specification 5.3, 'Unit Staff Qualifications,' to the Updated Final Safety Analysis Report (EPID L-2019-LLA-0268)," dated September 11, 2020 (ML20218A276)

12. PG&E Calculation Number DECOM-N-0003, (Vendor Calculation 14078100-C-M-00002), "Dose Consequences at the site Boundary, Existing Control Room, Proposed Alternate Control Rooms and the TSC following a Fuel Handling Accident in the Fuel Handling Building," dated October 6, 2020.

13. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, dated July 2000 (ML003716792)

14. PG&E Letter DCL-15-069, "License Amendment Request 15-03 Application of Alternative Source Term," dated June 17, 2015 (ML15176A527)

15. NRC Letter, "Diablo Canyon Power Plant, Units 1 and 2 – Issuance of Amendments RE: Revise Licensing Bases to Adopt Alternative Source Term (CAC NOS. MF6399 AND MF6400)," dated April 27, 2017 (ML17012A246)

16. United States Environmental Protection Agency, "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents," January 2017

17. PG&E Letter DCL-19-077, "Diablo Canyon Power Plant, Units 1 and 2 – Post-Shutdown Decommissioning Activities Report," dated December 4, 2019 (ML19338F173)

18. PG&E Letter DCL-19-081, "Diablo Canyon Power Plant, Units 1 and 2 Irradiated Fuel Management Plan," dated December 4, 2019 (ML19338F260)

19. NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980 (ML051400209)

20. NUREG-0737, Supplement No. 1, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," dated June 15, 1982 (ML102560009)

21. NRC Letter, "Closeout of Steam Generator Tube Rupture Analysis Issue for Diablo Canyon Power Plant, and Finding of Compliance with Condition 2.C.(9) of Unit 2 Operating License DPR-82 (TAC Nos. 68346 and 68347)," dated April 3, 1991

22. NRC Letter, "Diablo Canyon Pipeway Structure Review (TAC No. 59662)," dated September 14, 1987

23. NRC Letter, "Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Issuance of Amendments RE: Revision to Technical Specifications 3.7.17 and 4.3 for a Temporary Cask Pit Spent Fuel Storage Rack for Cycles 14 to 16 (TAC Nos. MC5143 AND MC5144)," dated November 21, 2005 (ML052970270)

24. NRC Letter, "Vermont Yankee Nuclear Power Station – Request for Exemption from the Requirements of 10 CFR 50.54(m) (TAC NO. MF2990)," dated June 18, 2014 (ML14147A216)

25. NRC Letter, "License Condition for Control of Heavy Loads, Diablo Canyon Unit 1," dated October 24, 1986

26. NRC Letter, "License Conditions for Masonry Walls," dated November 4, 1986

27. PG&E Letter DCL-98-080, "Notification of Implementation of License Amendments 120 (Unit 1) and 118 (Unit 2)," dated May 28, 1998 (ML16342E116)

28. PG&E Letter DCL-00-096, “Notification of Implementation of Improved Technical Specifications,” dated June 30, 2000 (ML003729396)

29. NRC Letter, “Diablo Canyon, Units 1 and 2 – Generic Letter 83-28, Item 1.1, Post-Trip Review (Program Description and Procedure),” dated May 15, 1985

30. NRC Letter, “Generic Letter 83-28, Item 1.2,” dated June 25, 1985

31. NRC Letter, “Generic Letter 83-28, Item 2.1 (Part 1),” dated October 27, 1986

32. NRC Letter, “Generic Letter 83-28, Item 2.1 (Part 2) (TAC Nos. 52832 and 61717),” dated July 24,1987

33. NRC Letter, “Generic Letter 83-28, Item 2.2.1 (TAC Nos. 53666 and 61718),” dated October 30, 1987

34. PG&E Letter DCL-87-156, “Generic Letter 83-28, Items 2.2.1 and 2.2.2,” dated June 30, 1987 (ML17083B881)

35. Industry Response to Generic Letter 83-28, Item 2.2.2, NUTAC 84-10, “Vendor Equipment Technical Information Program” (VETIP), submitted to NRC on March 23, 1984.

36. NRC Letter, “Generic Letter 83-28, Items 3.1.1 and 3.1.2,” dated April 10, 1986

37. NRC Letter, “Generic Letter 83-28, Items 3.1.3 and 3.2.3,” dated March 16, 1987

38. NRC Letter, “Generic Letter 83-28, Items 3.2.1 and 3.2.2,” dated October 21, 1986

39. NRC Letter, “Generic Letter 83-28, Items 4.1 and 4.5.1,” dated July 8, 1985

40. NRC Letter, “Generic Letter 83-28, Items 4.2.1 and 4.2.2,” dated June 24, 1985

41. PG&E Letter DCL-86-336, “Generic Letter 83-28, Items 4.2.3 and 4.2.4,” dated November 20, 1986 (ML16341D979)

42. NRC Letter, Supplement 1 to Generic Letter 83-28, “Required Actions Based on Generic Implications of Salem ATWS Events," dated October 7, 1992 (ML031210064)

43. NRC Letter, “Safety Evaluation for Generic Letter 83-28, Item 4.3, Automatic Actuation of Shunt Trip for Reactor Trip Breakers for Diablo Canyon Units 1 and 2 (TAC Nos. 53171 and 61721),” dated January 6, 1989

44. NRC Letter, “Generic Letter 83-28, Items 4.5.2,” dated March 18, 1987

45. NRC Letter, “Safety Evaluation of Generic Letter 83-28, Item 4.5.3 ‘Reactor Trip Reliability – On-Line Functional Testing of the Reactor Trip System,’ for Diablo Canyon Units 1 and 2 (TAC Nos. 53976 and 61723),” dated May 30, 1989

46. PG&E Letter DCL-20-063, "License Amendment Request 20-02, Non-Voluntary License Amendment Request to Revise Technical Specifications 3.2.1, $F_Q(Z)$, to Implement Methodology from WCAP-17661, Revision 1, 'Improved RAOC and CAOC F_Q Surveillance Technical Specifications,'" dated August 31, 2020 (ML20244A192)

Proposed Facility Operating License Changes (DPR-80 and DPR-82) – Markups

(22 Pages)

1. The Nuclear Regulatory Commission (the Commission) has found that:

A. The application for licenses by Pacific Gas and Electric Company complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

B. ~~Construction of the Diablo Canyon Nuclear Power Plant, Unit 1 (the facility), has been substantially completed in conformity with Provisional Construction Permit No. CPPR-39 and the application, as amended, the provisions of the Act, and the regulations of the Commission;~~Deleted per Amendment No. ###.

C. The facility will ~~operate~~ be maintained in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission~~, except as exempted from compliance in Section 2.D below~~;

D. There is reasonable assurance (i) that the activities authorized by this ~~operating~~ license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter I~~, except as exempted from compliance in Section 2.D below~~;

E. The Pacific Gas and Electric Company is technically qualified to engage in the activities authorized by this ~~operating~~ license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

F. The Pacific Gas and Electric Company has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements", of the Commission's regulations;

G. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility ~~Operating~~ License No. DPR-80, subject to the conditions for protection of the environment set forth herein, is in accordance with applicable Commission regulations governing environmental reviews (10 CFR Part 50, Appendix D and 10 CFR Part 51) and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Pursuant to Commission's Memorandum and Order CLI-84-13, dated August 10, 1984, Facility Operating License No. DPR-76 issued September 22, 1981, as subsequently amended, is superseded by Facility ~~Operating~~ License No. DPR-80, hereby issued to Pacific Gas and Electric Company to read as follows:

A. This License applies to the Diablo Canyon Nuclear Power Plant, Unit 1, a pressurized water nuclear reactor and associated equipment (the facility), owned by the Pacific Gas and Electric Company (PG&E). The facility is located in San Luis Obispo County, California, and is described in PG&E's ~~Final~~Defueled Safety Analysis Report as supplemented and amended, and the Environmental Report as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Pacific Gas and Electric Company:

(1) Pursuant to Section 104(b) of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess~~,~~ and use~~, and operate~~ the facility at the designated location in San Luis Obispo County, California, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, to ~~receive,~~ possess~~, and use~~ at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage ~~and amounts required for reactor operation~~, as described in the Defueled~~Final~~ Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in the calibration of radiation monitoring equipment ~~calibration~~, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials ~~as may be~~that were produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) ~~Maximum Power Level~~

~~The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.~~ Deleted per Amendment No. ###.

(2) Permanently Defueled Technical Specifications

The Permanently Defueled Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. ~~237~~###, are hereby incorporated in the license. Pacific Gas and Electric Company shall ~~operate~~maintain the facility in accordance with the Permanently Defueled Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) ~~Initial Test Program~~

~~The Pacific Gas and Electric Company shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Pacific Gas and Electric Company's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:~~

~~a. Elimination of any test identified in Section 14 of PG&E's Final Safety Analysis Report as amended as being essential;~~

~~b. Modification of test objectives, methods, or acceptance criteria for any test identified in section 14 of PG&E's Final Safety Analysis Report, as amended, as being essential;~~

~~c. Performance of any test at a power level different from that described in the program; and~~

~~d. Failure to complete any test included in the described program (planned or scheduled for power levels up to the authorized power level).~~

Deleted per Amendment No. ###.

(4) ~~Special Tests~~

~~PG&E is authorized to perform steam generator moisture carryover studies and turbine performance tests at the Diablo Canyon Nuclear Power Plant, Unit 1. These studies involve the use of an aqueous tracer solution of three (3) curies of sodium-24. PG&E's personnel shall be in charge of conducting these studies and be knowledgeable in the procedures. PG&E shall impose personnel exposure limits, posting, and survey requirements in conformance with those in 10 CFR Part 20 to minimize personnel exposure and contamination during the studies. Radiological controls shall be established in the areas of the chemical feed, feedwater, steam, condensate and sampling systems where the presence of the radioactive tracer is expected to warrant such controls. PG&E shall take special precautions to minimize radiation exposure and contamination during both the handling of the radioactive tracer prior to injection and the taking of system samples following injection of the tracer. PG&E shall ensure that all regulatory requirements for liquid discharge are met during disposal of all sampling effluents and when re-establishing continuous blowdown from the steam generators after completion of the studies.~~Deleted per Amendment No. ###.

(5) Fire Protection

a. PG&E shall implement and maintain all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the PG&E amendment request dated June 26, 2013, as supplemented by letters dated October 3, 2013, September 29, 2014, October 27, 2014, October 29, 2014, November 26, 2014, and December 31, 2014; February 25, 2015 (two letters), May 7, 2015, October 15, 2015, and December 31, 2015; and January 28, 2016, and as approved in the safety evaluation dated April 14, 2016. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, PG&E may make changes to the Fire Protection Program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

b. Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of a change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at DCPP. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed Fire Probabilistic Risk Assessment model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact:

(1) Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(2) Prior NRC review and approval is not required for individual changes that result in a risk increase less than $1x10^{-7}$/year (yr) for CDF and less than 1 $x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

c. Other Changes that May Be Made Without Prior NRC Approval

(1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental Fire Protection Program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. PG&E may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change will not affect the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

PG&E may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change will not affect the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and,
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

(2) Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to PG&E's Fire Protection Program that have been demonstrated to have no more than a minimal risk impact. PG&E may use its screening process as approved in the NRC safety evaluation dated April 14, 2016, to determine that certain Fire Protection Program changes meet the minimal criterion. PG&E shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the Fire Protection Program.

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

d. Transition License Conditions:

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) and (3) below, risk-informed changes to PG&E's Fire Protection Program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in c.(2) above.

(2) PG&E shall implement the modifications described in Attachment-S, Table S-2, "Plant Modifications Committed," of PG&E Letter DCL-16-014, dated January 28, 2016, by the end of the Units 1 and 2 refueling outages currently

scheduled for April/May 2017 (1R20) and February/March 2018 (2R20). PG&E shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

(3) PG&E shall implement the items as listed in Attachment-S, Table S-3, "Implementation Items," of PG&E Letter DCL-16-014, dated January 28, 2016, within 365 days after receipt of the safety evaluation/license amendment with the exception of Implementation Item S-3.24, which will be completed for each unit within 90 days after all modifications for the respective unit are operable (as listed in Attachment S, Table S-2).

(6) ~~NUREG-0737 Conditions~~

~~Each of the following conditions shall be completed to the satisfaction of the NRC as indicated below. Each of the following conditions references the appropriate Section in SER Supplements No. 10 and/or No. 12.~~

~~a. Shift Technical Advisor (Section I.A.1.1)~~

~~PG&E shall provide a fully-trained, on-shift technical advisor to the Shift Foreman.~~

~~b. Shift Staffing (Section I.A.1.3)~~

~~Until the plant has completed its startup test program, licensed personnel who are not regularly assigned members of the shift staff, including but not limited to the Operations Supervisor, shall not be assigned shift duties to satisfy the minimum staffing requirements for operation in Modes 1, 2, 3 and 4 except for cases of emergencies such as unexpected illness. Exceptions to this requirement may be made only after prior consultation with and approval by the NRC.~~

~~c. Management of Operations (Section I.B.1)~~

~~The Pacific Gas and Electric Company shall augment the plant staff to provide on each shift an individual experienced in comparable size pressurized water reactor operation. These individuals shall have at least one year of experience in operation of large pressurized water reactors or shall have participated in the startup of at least three pressurized water reactors. At least one such experienced individual shall be on duty on each shift through the startup test program whenever the reactor is not in a cold shutdown condition for at least the first year of operation or until the plant has attained a nominal 100% power level, whichever occurs first.~~

~~d. Procedures for Verifying Correct Performance of Operating Activities (Section I.C.6)~~

~~Procedures shall be available to verify the adequacy of the operating activities.~~

~~e. Deleted.~~

~~f. Relief and Safety Valve Test Requirements (Section II.D.1)~~

~~PG&E shall implement the results of the EPRI test program.~~

~~g. Containment Isolation Dependability (Section II.E.4.2)~~

~~PG&E shall limit the 12-inch vacuum/overpressure relief valve opening to less than or equal to 50 degrees.~~

~~h. Calculations for Small-Break LOCAs (Sections II.K.3.30 and II.K.3.31)~~

~~PG&E is participating in the Westinghouse Owners Group effort for this item and shall conform to the results of this effort. Within one year of staff approval of the Westinghouse generic methodology for calculating small break LOCAs (II.K.3.30), PG&E shall submit a plant specific calculation (II.K.3.31) for staff review and approval.~~

~~i. Long-Term Emergency Preparedness (Section III.A.2)~~

~~(1) PG&E shall submit a detailed control room design review summary report by December 31, 1984.~~

~~(2) PG&E shall complete operator training on the Safety Parameter Display System and emergency operating procedures by March 28, 1985.~~

~~(3) PG&E shall implement emergency operating procedures based upon Westinghouse Owners Group guidelines by March 28, 1985.~~Deleted per Amendment No. ###.

(7) Seismic Design Bases Reevaluation Program (SSER 27 Section IV.5)

PG&E shall develop and implement a program to reevaluate the seismic design bases used for the Diablo Canyon Nuclear Power Plant.

The program shall include the following Elements:

(1) PG&E shall identify, examine, and evaluate all relevant geologic and seismic data, information, and interpretations that have become available since the 1979 ASLB hearing in order to update the geology, seismology and tectonics in the region of the Diablo Canyon Nuclear Power Plant. If needed to define the earthquake potential of the region as it affects the Diablo Canyon Plant, PG&E

will also reevaluate the earlier information and acquire additional new data.

(2) PG&E shall reevaluate the magnitude of the earthquake used to determine the seismic basis of the Diablo Canyon Nuclear Plant using the information from Element 1.

(3) PG&E shall reevaluate the ground motion at the site based on the results obtained from Element 2 with full consideration of site and other relevant effects.

(4) PG&E shall assess the significance of conclusions drawn from the seismic reevaluation studies in Elements 1, 2 and 3, utilizing a probabilistic risk analysis and deterministic studies, as necessary, to assure adequacy of seismic margins.

PG&E shall submit for NRC staff review and approval a proposed program plan and proposed schedule for implementation by January 30, 1985. The program shall be completed and a final report submitted to the NRC three years following the approval of the program by the NRC staff.

PG&E shall keep the staff informed on the progress of the reevaluation program as necessary, but as a minimum will submit quarterly progress reports and arrange for semi-annual meetings with the staff. PG&E will also keep the ACRS informed on the progress of the reevaluation program as necessary, but not less frequently than once a year.

(8) ~~Control of Heavy Loads (SSER 27, Section IV. 6)~~

~~Prior to startup following the first refueling outage, the licensee shall submit commitments necessary to implement changes and modifications as required to satisfy the guidelines of Section 5.1.2 through 5.1.6 of NUREG-0612 (Phase II: 9-month responses to the NRC Generic Letter dated December 22, 1980).~~Deleted per Amendment No. ###.

(9) ~~Emergency Preparedness (SSER 27, Section IV.3)~~

~~In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.~~Deleted per Amendment No. ###.

(10) ~~Masonry Walls (SSER-27, Section IV.4; Safety Evaluation of November 2, 1984)~~

~~Prior to start-up following the first refueling outage, the licensee shall (1) evaluate the differences in margins between the staff criteria as set forth in the Standard Review Plan and the criteria used by the licensee, and (2) provide justification acceptable to the staff for those cases where differences exist between the staff's and the licensee's criteria~~.Deleted per Amendment No. ###.

(11) ~~Spent Fuel Pool Modification~~

~~The licensee is authorized to modify the spent fuel pool as described in the application dated October 30, 1985 (LAR 85-13) as supplemented. Amendment No. 8 issued on May 30, 1986 and stayed by the U.S. Court of Appeals for the Ninth Circuit pending completion of NRC hearings is hereby reinstated.~~

~~Prior to final conversion to the modified rack design, fuel may be stored, as needed, in either the modified storage racks described in Technical Specification 5.6.1.1 or in the unmodified storage racks (or both) which are designed and shall be maintained with a nominal 21-inch center-to-center distance between fuel assemblies placed in the storage racks.~~Deleted per Amendment No. ###.

(12) ~~Additional Conditions~~

~~The Additional Conditions contained in Appendix D, as revised through Amendment No. 230, are hereby incorporated into this license. Pacific Gas and Electric Company shall operate the facility in accordance with the Additional Conditions.~~Deleted per Amendment No. ###.

(13) Aging Management Program

If all spent fuel has not been removed from the Unit 1 spent fuel pool prior to November 2, 2028, an aging management program shall be submitted prior to this date for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Defueled Safety Analysis Report and shall remain in effect for Unit 1 until such time that all spent fuel has been removed from the Unit 1 spent fuel pool.

D. ~~Exemption~~

~~Exemption from certain requirements of Appendix J to 10 CFR Part 50 is described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 9. This exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public~~

~~interest. Therefore, this exemption, previously granted in Facility Operating License No. DPR-76, is hereby reaffirmed. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.~~Deleted per Amendment No. ###.

E. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54 (p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Diablo Canyon Power Plant, Units 1 and 2 Physical Security Plan, by Training and Qualification Plan, and Safeguards Contingency Plan," submitted by letter dated May 16, 2006.

PG&E shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The PG&E CSP was approved by License Amendment No. 210, as supplemented by a change approved by License Amendment No. 220.

Pursuant to NRC's Order EA-13-092, dated June 5, 2013, NRC reviewed and approved the license amendment 222 that permitted the security personnel of the licensee to possess and use certain specific firearms, ammunition, and other devices, such as large-capacity ammunition feeding devices, notwithstanding local, State, and certain Federal firearms laws that may prohibit such possession and use.

F. Deleted.

G. Deleted.

H. Financial Protection

PG&E shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

I. Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

J. Term of License

This License is effective as of the date of issuance and ~~shall expire at midnight on November 2, 2024~~is effective until the Commission notifies the licensee in writing that the license is terminated.

Attachments:

1. Appendix A – Permanently Defueled Technical Specifications
2. Appendix B – Environmental Protection Plan
3. Appendix C – Deleted
4. Appendix D – ~~Additional Conditions~~Deleted

1. The Nuclear Regulatory Commission (the Commission) has found that:

A. The application for licenses by Pacific Gas and Electric Company (PG&E) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

B. ~~Construction of the Diablo Canyon Nuclear Power Plant, Unit 2 (the facility), has been substantially completed in conformity with Provisional Construction Permit No. CPPR-69 and the application, as amended, the provisions of the Act, and the regulations of the Commission;~~Deleted per Amendment No. ###.

C. The facility will ~~operate~~ be maintained in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission~~, except as exempted from compliance in Section 2.D below~~;

D. There is reasonable assurance (i) that the activities authorized by this ~~operating~~ license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter I~~, except as exempted from compliance in Section 2.D below~~;

E. The Pacific Gas and Electric Company is technically qualified to engage in the activities authorized by this ~~operating~~ license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

F. The Pacific Gas and Electric Company has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements", of the Commission's regulations;

G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility ~~Operating~~ License No. DPR-82, subject to the conditions for protection of the environment set forth herein, is in accordance with applicable Commission regulations governing environmental reviews (10 CFR Part 50, Appendix D and 10 CFR Part 51) and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Pursuant to approval by the Nuclear Regulatory Commission in its Memorandum and Order (CLI-85-14) dated August 1, 1985, the license for fuel loading and low power testing, Facility Operating License No. DPR-81, issued on April 26, 1985, is superseded by Facility ~~Operating~~ License No. DPR-82, hereby issued to Pacific Gas and Electric Company to read as follows:

A. This License applies to the Diablo Canyon Nuclear Power Plant, Unit 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by PG&E. The facility is located in San Luis Obispo County, California, and is described in PG&E's ~~Final~~Defueled Safety Analysis Report as supplemented and amended, and the Environmental Report as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission herby licenses the Pacific Gas and Electric Company:

(1) Pursuant to Section 104(b) of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess~~,~~ and use~~, and operate~~ the facility at the designated location in San Luis Obispo County, California, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, to ~~receive,~~ possess~~, and use~~ at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage ~~and amounts required for reactor operation~~, as described in the Defueled~~Final~~ Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in the calibration of radiation monitoring equipment ~~calibration~~, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials ~~as may be~~ that were produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) ~~Maximum Power Level~~

~~The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.~~ Deleted per Amendment No.###.

(2) Permanently Defueled Technical Specifications

The Permanently Defueled Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. ###~~239~~, are hereby incorporated in the license. Pacific Gas and Electric Company shall ~~operate~~maintain the facility in accordance with the Permanently Defueled Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) ~~Initial Test Program (SSER 31, Section 4.4.1)~~

~~Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.~~ Deleted per Amendment No. ###.

(4) Fire Protection

a. PG&E shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the PG&E amendment request dated June 26, 2013, as supplemented by letters dated October 3, 2013, September 29, 2014, October 27, 2014, October 29, 2014, November 26, 2014, and December 31, 2014, February 25, 2015 (two letters), May 7, 2015, October 15, 2015, and December 31, 2015; and January 28, 2016, and as approved in the safety evaluation dated April 14, 2016. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, PG&E may make changes to the Fire Protection Program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

b. Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of a change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at DCPP. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed Fire Probabilistic Risk Assessment model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact:

(1) Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(2) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/year (yr) for CDF and less than 1×10^{-8}/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

c. Other Changes that May Be Made Without Prior NRC Approval

(1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental Fire Protection Program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. PG&E may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change will not affect the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

PG&E may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change will not affect the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and,
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

(2) Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to PG&E's Fire Protection Program that have been demonstrated to have no more than a minimal risk impact. PG&E may use its screening process as approved in the NRC safety evaluation dated April 14, 2016, to determine that certain Fire Protection Program changes meet the minimal criterion. PG&E shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the Fire Protection Program.

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

d. Transition License Conditions:

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) and (3) below, risk-informed changes to PG&E's Fire Protection Program may not be made without prior NRC review and approval unless the change has been

demonstrated to have no more than a minimal risk impact, as described in c.(2) above.

(2) PG&E shall implement the modifications described in Attachment-S, Table S-2, "Plant Modifications Committed," of PG&E Letter DCL-16-014, dated January 28, 2016, by the end of the Units 1 and 2 refueling outages currently scheduled for April/May 2017 (1R20) and February/March 2018 (2R20). PG&E shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

(3) PG&E shall implement the items as listed in Attachment-S, Table S-3, "Implementation Items," of PG&E Letter DCL-16-014, dated January 28, 2016, within 365 days after receipt of the safety evaluation/license amendment with the exception of Implementation Item S-3.24, which will be completed for each unit within 90 days after all modifications for the respective unit are operable (as listed in Attachment S, Table S-2).

(5) ~~NUREG-0737 Items~~

~~Each of the following conditions shall be completed to the satisfaction of the NRC as indicated below. Each condition references the appropriate Section in SER Supplements.~~

~~a. I.D.1 Detailed Control Room Design Review (SSER 31, Section 4.13)~~

~~PG&E shall comply with the requirements of Supplement 1 to NUREG-0737 for the conduct of a Detailed Control Room Design Review (DCRDR) in accordance with a schedule acceptable to the NRC staff.~~

~~b. II.E.4.2 Containment Isolation Dependability (SSER 31, Section 4.21)~~

~~PG&E shall limit the 12-inch vacuum/overpressure relief valve opening to less than or equal to 50 degrees.~~Deleted per Amendment No. ###.

(6) ~~Emergency Preparedness (SSER 31, Section 4.23.2 and SSER 32, Section 7)~~

~~In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the~~

~~provisions of 10 CFR Section 50.54(s)(2) will apply.~~Deleted per Amendment No. ###.

(7) ~~Masonry Walls (SSER 31, Section 4.7)~~

~~Prior to start-up following the first refueling outage, PG&E shall (1) evaluate the differences in margins between the staff criteria as set forth in the Standard Review Plan and the criteria used by the licensee, and (2) provide justification acceptable to the staff for those cases where differences exist between the staff's and PG&E's criteria.~~Deleted per Amendment No. ###.

(8) ~~Reactor Trip System Reliability – Generic Letter 83-28 (SSER 31, Section 4.8)~~

~~PG&E shall submit responses to and implement the requirements of Generic Letter 83-28 on a schedule which is consistent with that given in the PG&E letters dated January 24, and March 13, 1985.~~Deleted per Amendment No. ###.

(9) ~~Steam Generator Tube Rupture Analysis (SSER 31, Section 4.25)~~

~~By April 1988, PG&E shall submit for NRC review and approval an analysis which demonstrates that the steam generator tube rupture (SGTR) analysis presented in the FSAR is the most sever case with respect to the release of fission products and calculated doses. Consistent with the analytical assumptions, PG&E shall propose all necessary changes to the Technical Specifications (Appendix A) to this license.~~Deleted per Amendment No. ###.

(10) ~~Pipeway Structure DE and DDE Analysis (SSER 32, Section 4)~~

~~Prior to start-up following the first refueling outage PG&E shall complete a confirmatory analysis for the pipeway structure to further demonstrate the adequacy of the pipeway structure for load combinations that include the design earthquake (DE) and double design earthquake (DDE).~~Deleted per Amendment No. ###.

(11) ~~Spent Fuel Pool Modification~~

~~The licensee is authorized to modify the spent fuel pool as described in the application dated October 30, 1985 (LAR 85-13) as supplemented. Amendment No. 6 issued on May 30, 1986 and stayed by the U.S. Court of Appeals for the Ninth Circuit pending completion of NRC hearings is reinstated.~~

~~Prior to final conversion to the modified rack design, fuel may be stored, as needed, in either the modified storage racks described in Technical Specification 5.6.1.1 or in the unmodified storage racks (or both) which are designed and shall be maintained with a nominal 21-inch center-to-center~~

~~distance between fuel assemblies placed in the storage racks.~~Deleted per Amendment No. ###.

(12) ~~Additional Conditions~~

~~The Additional Conditions contained in Appendix D, as revised through Amendment No. 232, are hereby incorporated into this license. Pacific Gas and Electric Company shall operate the facility in accordance with the Additional Conditions.~~Deleted per Amendment No. ###.

(13) Aging Management Program

If all spent fuel has not been removed from the Unit 2 spent fuel pool prior to August 26, 2029, an aging management program shall be submitted prior to this date for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Defueled Safety Analysis Report and shall remain in effect for Unit 2 until such time that all spent fuel has been removed from the Unit 2 spent fuel pool.

D. ~~Exemption (SSER 31, Section 6.2.6)~~

~~An exemption from certain requirements of Appendix J to 10 CFR Part 50 is described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 9. This exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. Therefore, this exemption previously granted in Facility Operating License No. DPR-81 pursuant to 10 CFR 50.12 is hereby reaffirmed. The facility will operate, with the exemption authorized, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.~~Deleted per Amendment No. ###.

E. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provision of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Diablo Canyon Power Plant, Units 1 and 2 Physical Security Plan, Training and Qualification Plan and Safeguards Contingency Plan," submitted by letter dated May 16, 2006.

PG&E shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The PG&E CSP was approved by License Amendment No. 212, as supplemented by a change approved by License Amendment No. 222.

Pursuant to NRC's Order EA-13-092, dated June 5, 2013, NRC reviewed and approved the license amendment 224 that permitted the security personnel of the licensee to possess and use certain specific firearms, ammunition, and other devices, such as large-capacity ammunition feeding devices, notwithstanding local, State, and certain Federal firearms laws that may prohibit such possession and use.

F. Deleted.

G. Deleted.

H. <u>Financial Protection</u>

PG&E shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

I. <u>Mitigation Strategy</u>

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

J. Term of License

This License is effective as of the date of issuance and ~~shall expire at midnight on August 26, 2025~~is effective until the Commission notifies the licensee in writing that the license is terminated.

Attachments:

1. Appendix A – Permanently Defueled Technical Specifications ~~(NUREG-1151)~~
2. Appendix B – Environmental Protection Plan
3. Appendix C – Deleted
4. Appendix D – ~~Additional Conditions~~Deleted

Proposed Changes to Appendix D, Additional Conditions – Markups

(8 Pages)

Note – There is no clean copy of Appendix D provided as an attachment.
Appendix D is proposed for deletion in its entirety.

~~Appendix D~~

~~ADDITIONAL CONDITIONS~~

~~FACILITY OPERATING LICENSE NO. DPR-80~~

~~Pacific Gas & Electric Company shall comply with the following conditions on the schedules given below:~~

~~Amendment Number~~	~~Additional Conditions~~	~~Implementation Date~~
~~120~~	~~The licensee is authorized to relocate certain technical specifications requirements to the equipment control guidelines (ECGs) as referenced in the Updated Final Safety Analysis Report. Implementation of these amendments shall include relocation of these technical specification requirements to the ECGs as described in the licensee's application dated October 4, 1995, as supplemented by letters dated July 17, 1996, August 20, 1996, and June 2, 1997, and evaluated in the staff's safety evaluation dated February 3, 1998.~~	~~The amendment shall be implemented within 90 days of its issuance.~~
~~135~~	~~This amendment authorizes the relocation of certain Technical Specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these Technical Specification requirements to the appropriate documents, as described in Table LG of Details Relocated from Current Technical Specifications, Table R of Relocated Current Technical Specifications, Table LS of Less Restrictive Changes to Current Technical Specifications, and Table A of Administrative Changes to Current Technical Specifications that are attached to the NRC staff's Safety Evaluation enclosed with this amendment.~~	~~The amendment shall be implemented by June 30, 2000.~~

~~Amendment Number~~	~~Additional Conditions~~	~~Implementation Date~~
~~135~~	~~The schedule for the performance of new and revised Surveillance Requirements (SRs) shall be as follows:~~ ~~For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment.~~ ~~For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment.~~ ~~For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.~~ ~~For SRs that existed prior to this amendment whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to implementation of this amendment.~~	~~The amendment shall be implemented by June 30, 2000.~~
~~201~~	~~Determination of CRE unfiltered air inleakage as required by surveillance requirement (SR) 3.7.10.5, in accordance with TS 5.5.19.c.(i).~~ ~~The assessment of CRE habitability as required by TS 5.5.19.c.(ii).~~ ~~The measurement of CRE pressure as required by TS 5.5.19.d.~~	~~The amendment is effective as of the date of its issuance and the condition shall be implemented within 180 days of its issuance~~

~~Amendment Number~~	~~Additional Conditions~~	~~Implementation Date~~
	~~Following implementation, this condition will be performed as stated in the condition:~~ ~~The first performance of SR 3.7.10.5, in accordance with Specification 5.5.19.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful tracer gas test, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.~~ ~~The first performance of the periodic assessment of CRE habitability, Specification 5.5.19.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful tracer gas test, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.~~ ~~The first performance of the periodic measurement of CRE pressure, Specification 5.5.19.d, shall be within 24 months, plus the 182 days allowed by SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful pressure measurement test, or within 182 days if not performed previously.~~	

~~Amendment Number~~	~~Additional Conditions~~	~~Implementation Date~~
~~230~~	~~Implementation of the amendment adopting the alternative source term shall include the following plant modifications:~~ ~~Install shielding material, equivalent to that provided by the Control Room outer walls, at the external concrete west wall of the Control Room briefing room.~~ ~~Install a high efficiency particulate air filter in the Technical Support Center normal ventilation system.~~ ~~Re-classify a portion of the 40-inch Containment Penetration Area (GE/GW) Ventilation line from PG&E Design Class II to PG&E Design Class I and upgrade the damper actuators, pressure switches, and the damper solenoid valves to PG&E Design Class I.~~ ~~Update setpoints for the redundant safety related gamma sensitive area radiation monitors (1-RE 25/26, 2-RE 25/26).~~	~~The amendment is effective as of the date of its issuance and the condition shall be implemented within 365 days of its issuance~~

~~Appendix D~~

~~ADDITIONAL CONDITIONS~~

~~FACILITY OPERATING LICENSE NO. DPR-82~~

~~Pacific Gas & Electric Company shall comply with the following conditions on the schedules given below:~~

~~Amendment Number~~	~~Additional Conditions~~	~~Implementation Date~~
~~118~~	~~The licensee is authorized to relocate certain technical specifications requirements to the equipment control guidelines (ECGs) as referenced in the Updated Final Safety Analysis Report. Implementation of these amendments shall include relocation of these technical specification requirements to the ECGs as described in the licensee's application dated October 4, 1995, as supplemented by letters dated July 17, 1996, August 20, 1996, and June 2, 1997, and evaluated in the staff's safety evaluation dated February 3, 1998.~~	~~The amendment shall be implemented within 90 days of its issuance.~~
~~135~~	~~This amendment authorizes the relocation of certain Technical Specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these Technical Specification requirements to the appropriate documents, as described in Table LG of Details Relocated from Current Technical Specifications, Table R of Relocated Current Technical Specifications, Table LS of Less Restrictive Changes to Current Technical Specifications, and Table A of Administrative Changes to Current Technical Specifications that are attached to the NRC staff's Safety Evaluation enclosed with this amendment.~~	~~The amendment shall be implemented by June 30, 2000.~~

~~Amendment Number~~	~~Additional Conditions~~	~~Implementation Date~~
~~135~~	~~The schedule for the performance of new and revised Surveillance Requirements (SRs) shall be as follows:~~ ~~For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment.~~ ~~For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment.~~ ~~For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.~~ ~~For SRs that existed prior to this amendment whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to implementation of this amendment.~~	~~The amendment shall be implemented by June 30, 2000.~~
~~202~~	~~Determination of CRE unfiltered air inleakage as required by surveillance requirement (SR) 3.7.10.5, in accordance with TS 5.5.19.c.(i).~~ ~~The assessment of CRE habitability as required by TS 5.5.19.c.(ii).~~ ~~The measurement of CRE pressure as required by TS 5.5.19.d.~~	~~The amendment is effective as of the date of its issuance and the condition shall be implemented within 180 days of its issuance~~

~~Amendment Number~~	~~Additional Conditions~~	~~Implementation Date~~
	~~Following implementation, this condition will be performed as stated in the condition:~~ ~~The first performance of SR 3.7.10.5, in accordance with Specification 5.5.19.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful tracer gas test, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.~~ ~~The first performance of the periodic assessment of CRE habitability, Specification 5.5.19.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful tracer gas test, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.~~ ~~The first performance of the periodic measurement of CRE pressure, Specification 5.5.19.d, shall be within 24 months, plus the 182 days allowed by SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful pressure measurement test, or within 182 days if not performed previously.~~	

~~Amendment Number~~	~~Additional Conditions~~	~~Implementation Date~~
~~232~~	~~Implementation of the amendment adopting the alternative source term shall include the following plant modifications:~~ ~~Install shielding material, equivalent to that provided by the Control Room outer walls, at the external concrete west wall of the Control Room briefing room.~~ ~~Install a high efficiency particulate air filter in the Technical Support Center normal ventilation system.~~ ~~Re-classify a portion of the 40-inch Containment Penetration Area (GE/GW) Ventilation line from PG&E Design Class II to PG&E Design Class I and upgrade the damper actuators, pressure switches, and the damper solenoid valves to PG&E Design Class I.~~ ~~Update setpoints for the redundant safety related gamma sensitive area radiation monitors (1-RE 25/26, 2-RE 25/26).~~	~~The amendment is effective as of the date of its issuance and the condition shall be implemented within 365 days of its issuance~~

Proposed Changes to Appendix A, Technical Specifications – Markups

(72 Pages)

Note – Technical Specifications 3.1 through 3.9 that are deleted in their entirety are identified as such in the Technical Specification Table of Contents; however, the associated deletions are not included in this attachment. The remaining Technical Specifications are intentionally not re-numbered.

~~(continued)~~

~~(continued)~~

~~(continued)~~

1.0 USE AND APPLICATION

1.1 Definitions

--NOTE--
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with the provisions of the CERTIFIED FUEL HANDLER Training and Retraining Program required by Specification 5.3.2.
NON-CERTIFIED OPERATOR	A NON-CERTIFIED OPERATOR is an operator who complies with the qualification requirements of Specification 5.3.1, but is not a CERTIFIED FUEL HANDLER.
~~ACTUATION LOGIC TEST~~	~~An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.~~
~~AXIAL FLUX DIFFERENCE (AFD)~~	~~AFD shall be the difference in normalized flux signals between the top and bottom halves of an excore neutron detector.~~
~~CHANNEL CALIBRATION~~	~~A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY. Calibration of instrument channels with resistance temperature detectors (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping or total channel steps.~~
~~CHANNEL CHECK~~	~~A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.~~

~~(continued)~~

~~1.1 Definitions (continued)~~

~~CHANNEL FUNCTIONAL TEST (CFT)~~

~~A CFT shall be:~~

~~a. Analog channels - the injection of a simulated or actual signal into the channel as close to the sensor as practical to verify OPERABILITY of all devices in the channel required for channel OPERABILITY, or~~

~~b. Bistable channels - the injection of a simulated or actual signal into the sensor to verify OPERABILITY of all devices in the channel required for channel OPERABILITY, or~~

~~c. Digital channels - the injection of a simulated or actual signal into the channel as close to the sensor input to the process racks as practical to verify OPERABILITY of all devices in the channel required for channel OPERABILITY.~~

~~The CFT may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.~~

~~CHANNEL OPERATIONAL TEST (COT)~~

~~A COT shall be:~~

~~a. Analog, bistable, and Eagle 21 process protection system digital channels - the injection of a simulated or actual signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY.~~

~~b. Tricon/Advanced Logic System process protection system digital channels - the use of diagnostic programs to test digital hardware, manual verification that the setpoints and tunable parameters are correct, and the injection of simulated process data into the channel as close to the sensor input to the process racks as practical to verify channel OPERABILITY of all devices in the channel required for OPERABILITY.~~

~~The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping or total channel steps.~~

~~(continued)~~

~~CORE ALTERATION~~	~~CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.~~
~~CORE OPERATING LIMITS REPORT (COLR)~~	~~The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.~~

~~1.1 Definitions (continued)~~

~~DOSE EQUIVALENT I-131~~	~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using the committed thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."~~
~~DOSE EQUIVALENT XE-133~~	~~DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."~~

~~(continued)~~

~~1.1 Definitions (continued)~~

~~ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME~~	~~The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.~~
~~LEAKAGE~~	~~LEAKAGE shall be:~~

~~a. Identified LEAKAGE~~

~~1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;~~

~~2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or~~

~~(continued)~~

~~1.1 Definitions~~

~~LEAKAGE (continued)~~

~~3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE).~~

~~b. Unidentified LEAKAGE~~

~~All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE.~~

~~c. Pressure Boundary LEAKAGE~~

~~LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.~~

~~MASTER RELAY TEST~~

~~A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.~~

~~MODE~~

~~A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.~~

~~OPERABLE—OPERABILITY~~

~~A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).~~

~~PHYSICS TESTS~~

~~PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:~~

~~a. Described in Chapter 14 of the FSAR;~~

~~b. Authorized under the provisions of 10 CFR 50.59; or~~

~~c. Otherwise approved by the Nuclear Regulatory Commission.~~

~~(continued)~~

~~1.1 Definitions (continued)~~

~~PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~	~~The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve (PORV) lift settings and arming temperature associated with the Low Temperature Overpressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6.~~
~~QUADRANT POWER TILT RATIO (QPTR)~~	~~QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.~~
~~RATED THERMAL POWER (RTP)~~	~~RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt for each unit.~~
~~REACTOR TRIP SYSTEM (RTS) RESPONSE TIME~~	~~The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.~~
~~SHUTDOWN MARGIN (SDM)~~	~~SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:~~ ~~a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and~~ ~~b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.~~

~~(continued)~~

~~1.1 Definitions (continued)~~

~~SLAVE RELAY TEST~~	~~A SLAVE RELAY TEST shall consist of energizing all slave relays and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.~~
~~THERMAL POWER~~	~~THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.~~
~~TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)~~	~~A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping or total channel steps.~~

~~Table 1.1-1 (page 1 of 1)~~

~~MODES~~

~~MODE~~	~~TITLE~~	~~REACTIVITY CONDITION (k_{eff})~~	~~% RATED THERMAL POWER[a]~~	~~AVERAGE REACTOR COOLANT TEMPERATURE (°F)~~
~~1~~	~~Power Operation~~	~~≥ 0.99~~	~~> 5~~	~~NA~~
~~2~~	~~Startup~~	~~≥ 0.99~~	~~≤ 5~~	~~NA~~
~~3~~	~~Hot Standby~~	~~< 0.99~~	~~NA~~	~~≥ 350~~
~~4~~	~~Hot Shutdown[b]~~	~~< 0.99~~	~~NA~~	~~350 > T_{AVG} > 200~~
~~5~~	~~Cold Shutdown[b]~~	~~< 0.99~~	~~NA~~	~~≤ 200~~
~~6~~	~~Refueling[c]~~	~~NA~~	~~NA~~	~~NA~~

~~[a] Excluding decay heat.~~

~~[b] All reactor vessel head closure bolts fully tensioned.~~

~~[c] One or more reactor vessel head closure bolts less than fully tensioned.~~

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe ~~operation of the unit~~handling and storage of nuclear fuel. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., ~~inoperable equipment or~~ variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the ~~unit~~facility is in a ~~MODE or~~ specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the ~~unit~~facility is not within the LCO Applicability.

~~If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.~~

~~Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition~~

~~However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:~~

~~a. Must exist concurrent with the first inoperability; and~~

~~b. Must remain inoperable or not within limits after the first inoperability is resolved.~~

~~(continued)~~

~~1.3 Completion Times~~

~~DESCRIPTION (continued)~~

~~The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:~~

~~a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or~~

~~b. The stated Completion Time as measured from discovery of the subsequent inoperability.~~

~~The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.~~

~~The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ."~~

~~(continued)~~

~~1.3 Completion Times (continued)~~

~~EXAMPLES~~

~~The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.~~

~~EXAMPLE 1.3-1~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 5.~~	~~6 hours~~ ~~36 hours~~

~~Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.~~

~~The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.~~

~~If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.~~

~~(continued)~~

~~1.3 Completion Times~~

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-2~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One pump inoperable.~~	~~A.1 Restore pump to OPERABLE status.~~	~~7 days~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 5.~~	~~6 hours~~ ~~36 hours~~

~~When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.~~

~~When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.~~

~~While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.~~

~~While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.~~

~~On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.~~

~~(continued)~~

~~1.3 Completion Times~~

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-3~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One Function X train inoperable.~~	~~A.1 Restore Function X train to OPERABLE status.~~	~~7 days~~
~~B. One Function Y train inoperable.~~	~~B.1 Restore Function Y train to OPERABLE status.~~	~~72 hours~~
~~C. One Function X train inoperable.~~ ~~AND~~ ~~One Function Y train inoperable.~~	~~C.1 Restore Function X train to OPERABLE status.~~ ~~OR~~ ~~C.2 Restore Function Y train to OPERABLE status.~~	~~72 hours~~ ~~72 hours~~

~~(continued)~~

~~1.3 Completion Times~~

~~EXAMPLES~~

~~EXAMPLE 1.3-3 (continued)~~

~~When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).~~

~~If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A.~~

~~It is possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.~~

~~(continued)~~

~~1.3 Completion Times~~

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-4~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One or more valves inoperable.~~	~~A.1 Restore valve(s) to OPERABLE status.~~	~~4 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 4.~~	~~6 hours~~ ~~12 hours~~

~~A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.~~

~~Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.~~

~~If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.~~

~~(continued)~~

~~1.3 Completion Times~~

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-5~~

~~ACTIONS~~

~~-------------------------------------NOTE-------------------------------------~~
~~Separate Condition entry is allowed for each inoperable valve.~~
~~--~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One or more valves inoperable.~~	~~A.1 Restore valve to OPERABLE status.~~	~~4 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 4.~~	~~6 hours~~ ~~12 hours~~

~~The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.~~

~~The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.~~

~~If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.~~

~~Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.~~

~~(continued)~~

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

(continued)

~~1.3 Completion Times~~

~~EXAMPLES (continued)~~

~~EXAMPLE 1.3-7~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One subsystem inoperable.~~	~~A.1 Verify affected subsystem isolated.~~	~~1 hour~~ ~~AND~~ ~~Once per 8 hours thereafter~~
	~~AND~~	
	~~A.2 Restore subsystem to OPERABLE status.~~	~~72 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~	~~6 hours~~
	~~AND~~	
	~~B.2 Be in MODE 5.~~	~~36 hours~~

~~Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.~~

~~If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.~~

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.
EXAMPLES	The following examples ~~illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3~~ illustrate the type of frequency statements that appear in the Permanently Defueled Technical Specifications (PDTS).

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
~~Perform CHANNEL CHECK~~Verify level is within limits.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the ~~Technical Specifications (TS)~~PDTS. The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for ~~operational~~ flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when ~~the equipment is inoperable,~~ a variable is outside specified limits, or the ~~unit~~facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the ~~unit~~facility is in a ~~MODE or other~~ specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified ~~(refer to Example 1.4-3)~~, then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the ~~unit~~facility is not in a ~~MODE or other~~ specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the ~~MODE or other~~ specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify ~~flow is within limits~~...	~~Once within 12 hours after ≥ 25% RTP~~ ~~AND~~ ~~24 hours thereafter~~ Prior to each fuel assembly move…

~~Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.~~

~~The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.~~

Example 1.4-2 illustrates a one time performance Frequency.

This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2.

~~(continued)~~

~~1.4 Frequency~~

~~EXAMPLES (continued)~~

~~EXAMPLE 1.4-3~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~	~~FREQUENCY~~
~~------------------------------NOTE--------------------------~~ ~~Not required to be performed until 12 hours after ≥ 25% RTP.~~ ~~--~~ ~~Perform channel adjustment.~~	~~7 days~~

~~The interval continues, whether or not the unit operation is < 25% RTP between performances.~~

~~As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.~~

~~Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.~~

2.0 ~~SAFETY LIMITS (SLs)~~Deleted

~~2.1 SLs~~

~~2.1.1 Reactor Core SLs~~

~~In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.~~

~~2.1.2 RCS Pressure SL~~

~~In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.~~

~~2.2 SL Violations~~

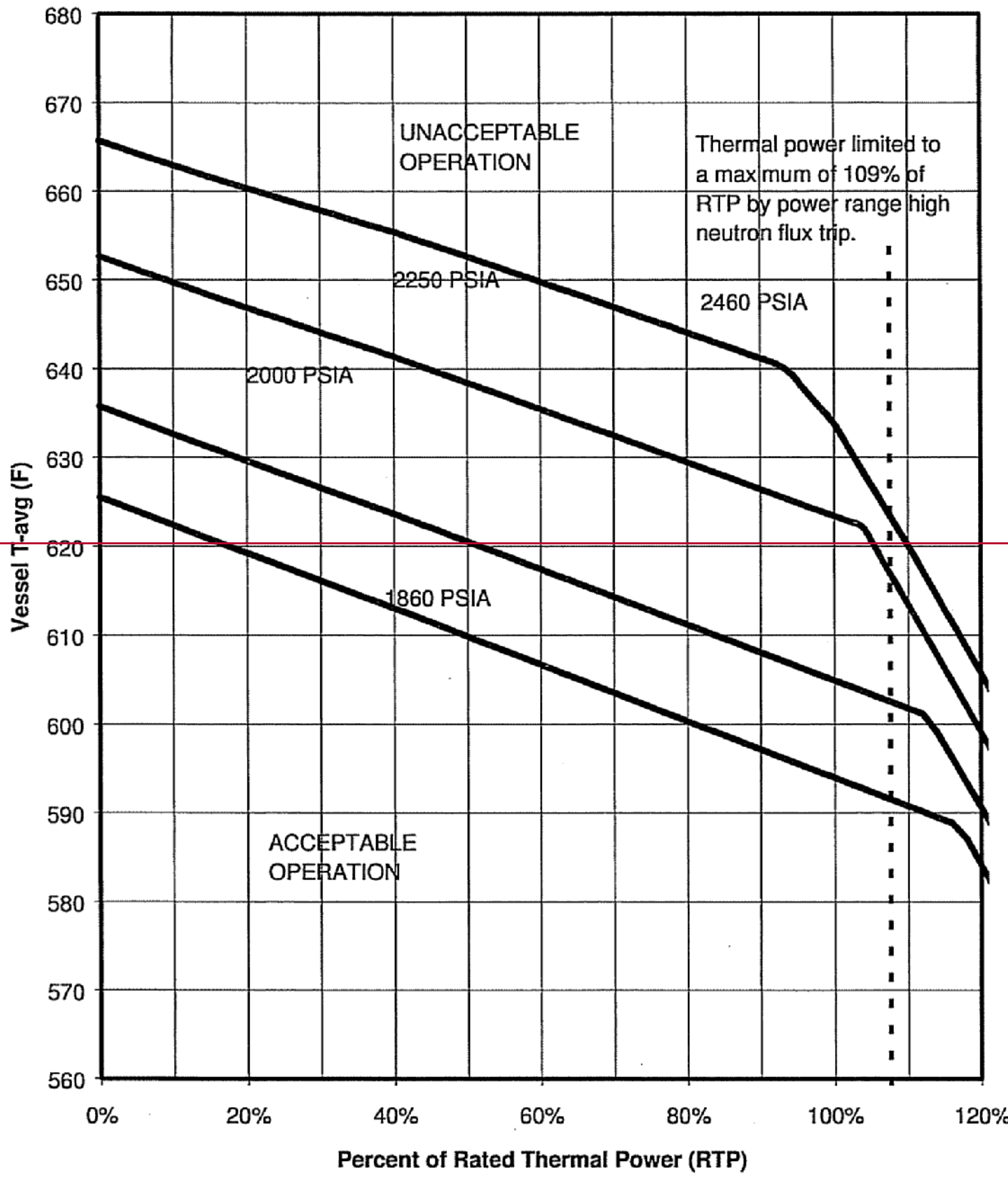
~~2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.~~

~~2.2.2 If SL 2.1.2 is violated:~~

~~2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.~~

~~2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.~~

UNITS 1 & 2



~~Figure 2.1.1-1~~

~~REACTOR CORE SAFETY LIMIT~~

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the ~~MODES or other~~ specified conditions in the Applicability, except as provided in LCO 3.0.2~~, LCO 3.0.7, and LCO 3.0.8~~.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met~~, except as provided in LCO 3.0.5 and LCO 3.0.6~~. ~~If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.~~
~~LCO 3.0.3~~	~~When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:~~ ~~a. MODE 3 within 7 hours;~~ ~~b. MODE 4 within 13 hours; and~~ ~~c. MODE 5 within 37 hours.~~ ~~Exceptions to this Specification are stated in the individual Specifications.~~ ~~Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.~~ ~~LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.~~
~~LCO 3.0.4~~	~~When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:~~ ~~a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;~~ ~~b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or~~ ~~c. When an allowance is stated in the individual value, parameter, or other Specification.~~ ~~This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.~~

~~(continued)~~

~~3.0 LCO APPLICABILITY (continued)~~

~~LCO 3.0.5~~	~~Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.~~
~~LCO 3.0.6~~	~~When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.15, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~ ~~When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.~~
~~LCO 3.0.7~~	~~Test Exception LCO 3.1.8, allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.~~
~~LCO 3.0.8~~	~~When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:~~ ~~a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or~~

~~(continued)~~

~~3.0 LCO APPLICABILITY (continued)~~

~~LCO 3.0.8 (continued)~~	~~b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.~~ ~~At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.~~

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the ~~MODES or other~~ specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on ~~inoperable equipment or~~ variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. ~~For Frequencies specified as "once," the above interval extension does not apply.~~ ~~If a Completion Time requires periodic performance on a "once per ..." basis, the above Frequency extension applies to each performance after the initial performance.~~ ~~Exceptions to this Specification are stated in the individual Specifications.~~
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a ~~MODE or other~~ specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. ~~When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.~~ This provision shall not prevent entry into ~~MODES or other~~ specified conditions in the Applicability that are required to comply with ACTIONS ~~or that are part of a shutdown of the unit~~.

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Water Level

LCO 3.7.15 The spent fuel pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool water level not within limit.	A.1 ~~-----------NOTE-------------~~ ~~LCO 3.0.3 is not applicable.~~ ~~------------------------------~~ Suspend movement of irradiated fuel assemblies in the spent fuel pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	~~In accordance with the Surveillance Frequency Control Program~~ 7 days

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Pool Boron Concentration

LCO 3.7.16 The spent fuel pool boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	~~-----------------NOTE----------------~~ ~~LCO 3.0.3 is not applicable.~~ ~~---~~	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify the spent fuel pool boron concentration is within limit.	~~In accordance with the Surveillance Frequency Control Program~~ 7 days

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17 Fuel assembly storage in the spent fuel pool shall be maintained such that:

a. In the permanent spent fuel storage racks any four cells shall be in a configuration as shown in Figure 3.7.17-1.~~, and~~

~~b. In the cask pit storage rack, for Cycles 14 – 16, the fuel assemblies shall have:~~

~~1. An initial enrichment ≤ 4.1 wt% U-235;~~

~~2. A discharge burnup in the "acceptable" region of Figure 3.7.17-4; and~~

~~3. A minimum decay time of 10 years since being discharged from the reactor.~~

~~c. The total combined spent fuel pool capacity in the permanent and cask pit storage racks, for Cycles 14 – 16, is limited to no more than 1433 irradiated fuel assemblies. This limit does not apply for an emergency core offload.~~

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ~~----------NOTE--------------~~ ~~LCO 3.0.3 is not applicable.~~ ~~--------------------------------~~ Initiate action to move the noncomplying fuel assembly into an acceptable storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means that the fuel assembly characteristics and its expected storage location is in accordance with LCO 3.7.17.	Prior to each fuel assembly move, when the assembly will be stored in the spent fuel pool.

A	A
A	A

All Cell Configuration

F	B
B	B

2x2 Array Configuration

G	W
W	G

Checkerboard Configuration

All Cell:

A Fuel assembly with a discharge burnup in the "acceptable" region of Figure 3.7.17-2.

2x2 Array:

F (a) Fuel assembly with an initial enrichment ≤ 4.9 wt% U-235; or

(b) Fuel assembly with an initial enrichment ≤ 5.0 wt% U-235 and an IFBA loading equivalent to 16 rods each with 1.5 mg ^{10}B/in over 120 inches.

B Fuel assembly with a discharge burnup in the "acceptable" region of Figure 3.7.17-3.

Checkerboard:

G Fuel assembly with an initial enrichment ≤ 5.0 wt% U-235.

W Water cell – locations where fuel assemblies are not present, non-fissile components are permitted.

FIGURE 3.7.17-1
ALLOWABLE STORAGE CONFIGURATIONS
(ALL CELL, 2X2 ARRAY, CHECKERBOARD)
FOR THE PERMANENT SPENT FUEL POOL STORAGE RACKS

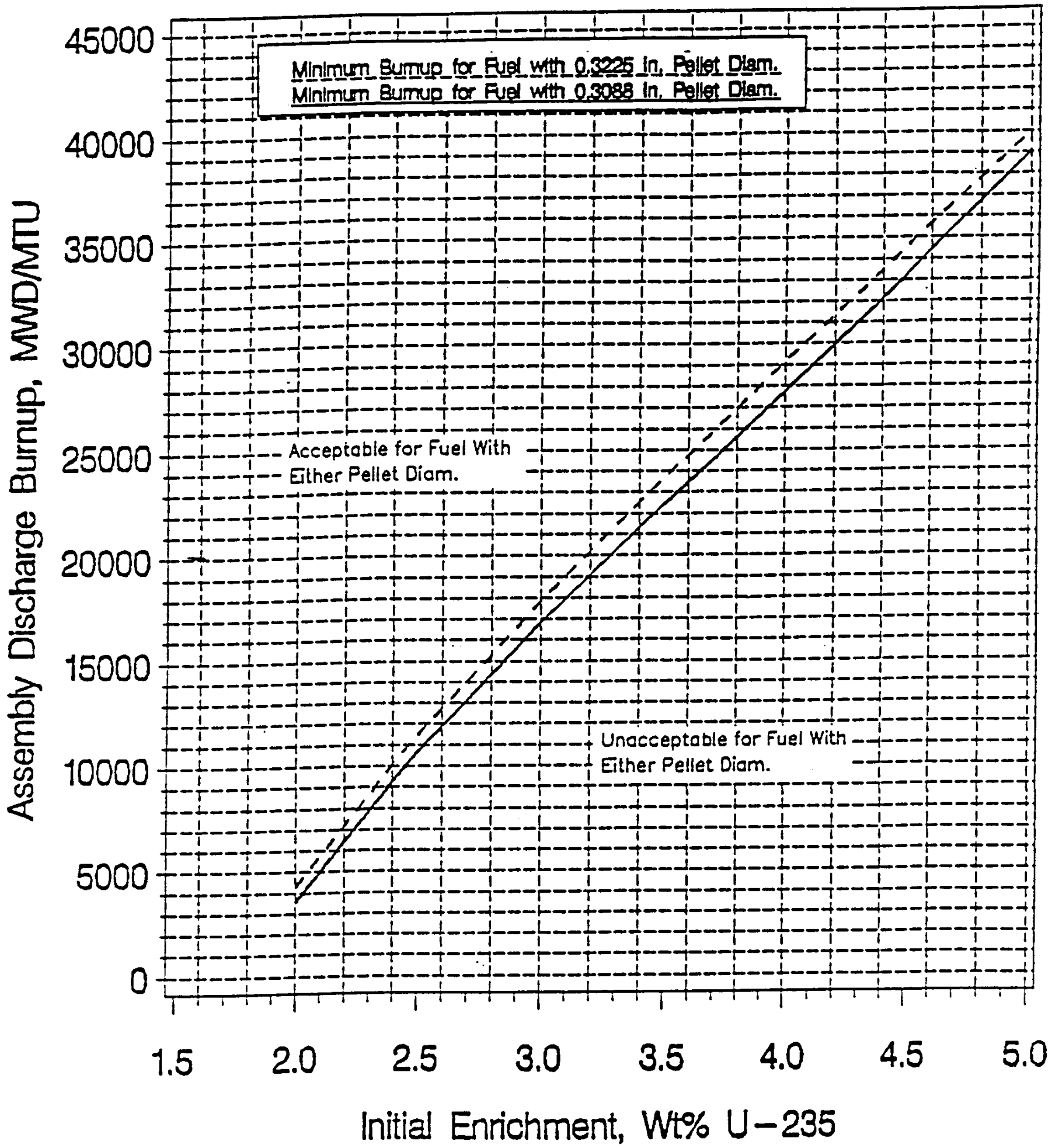


FIGURE 3.7.17-2
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT AND FUEL PELLET DIAMETER
FOR AN ALL CELL STORAGE CONFIGURATION FOR THE PERMANENT SPENT
FUEL POOL STORAGE RACKS

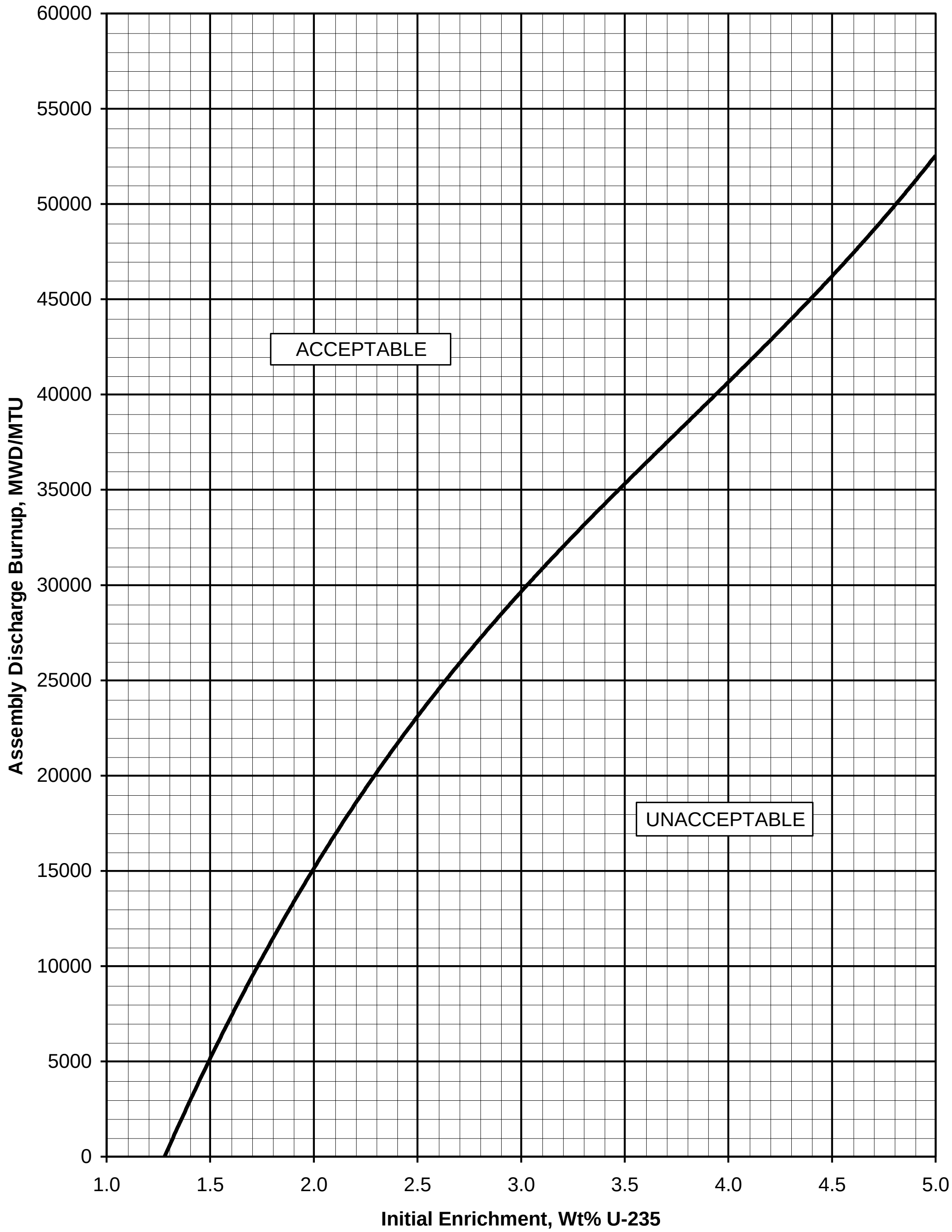
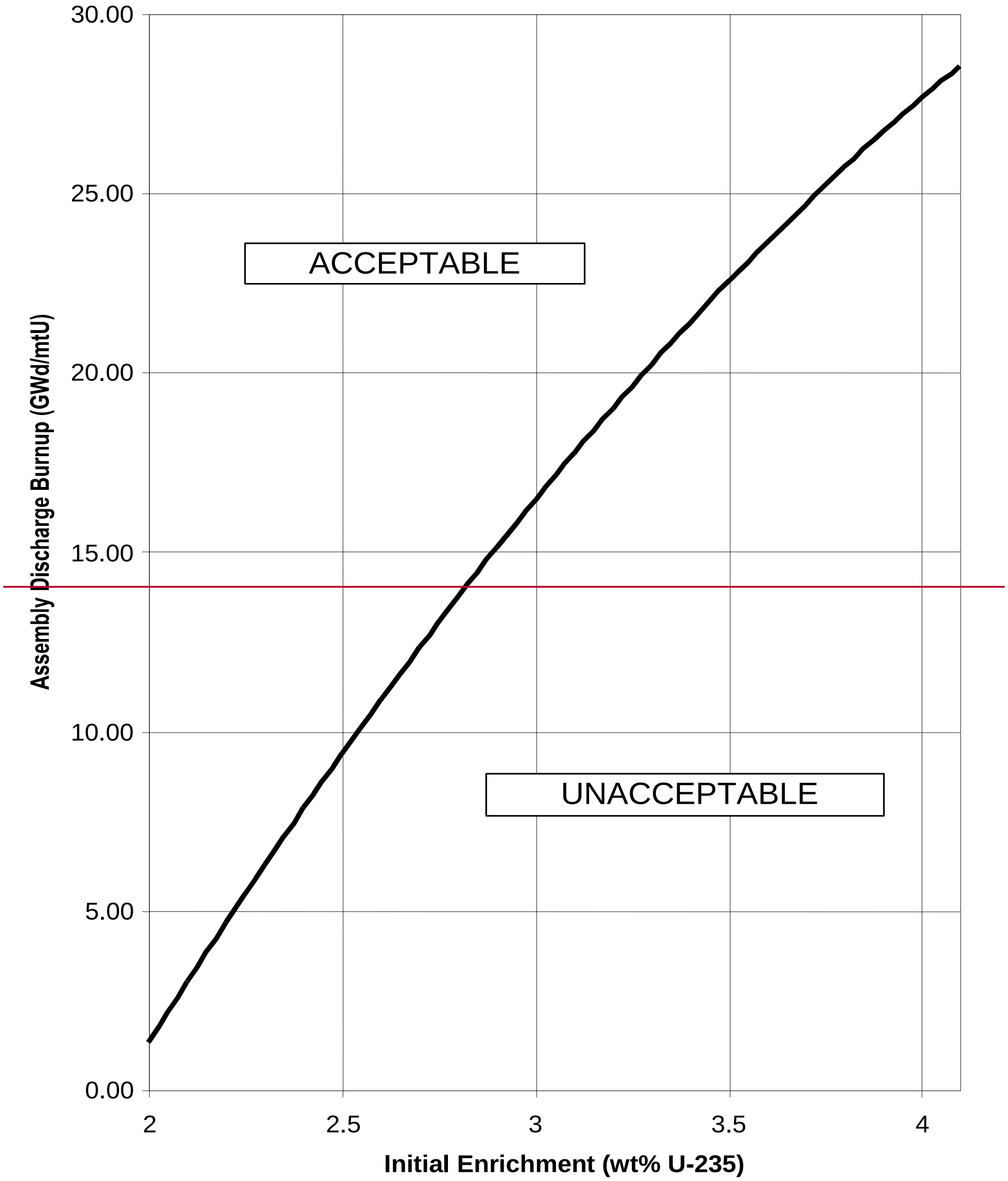


FIGURE 3.7.17-3
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT
FOR A 2X2 ARRAY STORAGE CONFIGURATION FOR THE PERMANENT SPENT
FUEL POOL STORAGE RACKS



~~FIGURE 3.7.17-4~~
~~MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP~~
~~AS A FUNCTION OF INITIAL ENRICHMENT~~
~~FOR SPENT FUEL STORAGE IN THE CASK PIT STORAGE RACK~~

~~NOTES:~~

~~I. INITIAL ENRICHMENT NOT TO EXCEED 4.1 WT %;~~

~~2. MINIMUM SPENT FUEL DECAY TIME OF 10 YEARS SINCE BEING DISCHARGED FROM THE REACTOR; AND~~

~~3. APPLICABLE DURING CYCLES 14 – 16 WITH CASK PIT RACK INSTALLED.~~

4.0 DESIGN FEATURES

4.1 Site Location

The DCPP site consists of approximately 750 acres which are adjacent to the Pacific Ocean in San Luis Obispo County, California, and is approximately twelve (12) miles west-southwest of the city of San Luis Obispo.

4.2 ~~Reactor Core~~

~~4.2.1 Fuel Assemblies~~

~~The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.~~

~~4.2.2 Control Rod Assemblies~~

~~The reactor core shall contain 53 control rod assemblies. The control rod material shall be silver, indium, and cadmium, as approved by the NRC.~~Deleted

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The permanent spent fuel pool storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the D~~F~~SAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 806 ppm, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the D~~F~~SAR;

d. A nominal 11 inch center to center distance between fuel assemblies placed in the fuel storage racks;

~~(continued)~~

~~4.0 DESIGN FEATURES~~

~~4.3 Fuel Storage (continued)~~

e. Fuel assemblies with a discharge burnup in the "acceptable" region of Figure 3.7.17-2 for the all cell configuration as shown in Figure 3.7.17-1;

f. Fuel assemblies with a discharge burnup in the "acceptable" region of Figure 3.7.17-3 for the 2x2 array configuration as shown in Figure 3.7.17-1.

~~4.3.1.2 The new fuel storage racks are designed and shall be maintained with:~~

~~a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;~~

~~b. k_{eff} ≤ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1.1 of the FSAR;~~

~~c. k_{eff} ≤ 0.98 if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1.1.1 of the FSAR; and~~

~~d. A nominal 22 inch center to center distance between fuel assemblies placed in the storage racks.~~

~~4.3.1.3 For cycles 14-16, the cask pit storage rack is designed and shall be maintained with:~~

~~a. Fuel assemblies having a maximum U-235 enrichment of 4.1 weight percent;~~

~~b. k_{eff} < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties;~~

~~c. k_{eff} ≤ 0.95 if fully flooded with water borated to 800 ppm, which includes an allowance for uncertainties;~~

~~d. A nominal 9 inch center to center distance between fuel assemblies placed in the cask pit fuel storage rack;~~

~~e. Fuel assemblies with discharge burnup in the "acceptable" region of Figure 3.7.17-4;~~

~~f. Fuel assemblies having a 10 year minimum decay time since being discharged from the reactor; and~~

~~g. A neutron absorbing material (Metamic™) between the stored fuel assemblies~~

~~(continued)~~

~~4.0 DESIGN FEATURES~~

~~4.3 Fuel Storage (continued)~~

4.3.2 Drainage

The spent fuel storage pools are designed and shall be maintained to prevent inadvertent draining of the pool below elevation 133 ft.

4.3.3 Capacity

The permanent spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1324 fuel assemblies. ~~For cycles 14-16, the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 154 fuel assemblies. For cycles 14-16, the total combined spent fuel pool capacity in the permanent and cask pit storage racks is limited to no more than 1478 fuel assemblies.~~

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall ~~unit~~facility operation and shall delegate in writing the succession to this responsibility ~~during his absence~~when absent.

The plant manager or ~~his~~ designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect ~~nuclear safety~~the safe handling and storage of nuclear fuel.

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

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for ~~unit operation~~facility staff and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safe~~ty~~ storage and handling of ~~the nuclear power plant~~spent nuclear fuel. The primary role of all nuclear workers is to protect the health and safety of the public.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all ~~operating~~facility organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the DSAR~~FSAR Update~~;

b. The plant manager shall be responsible for overall safe operation of the ~~plant~~facility and shall have control over those onsite activities necessary for ~~safe operation and maintenance of the plant~~safe storage and handling of the nuclear fuel;

c. A specified corporate officer shall have corporate responsibility for ~~overall plant nuclear safety~~the safe storage and handling of nuclear fuel and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the ~~plant to ensure nuclear safety~~facility to ensure safe storage and handling of nuclear fuel; and

d. The individuals who train the ~~operating staff~~CERTIFIED FUEL HANDLERs, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their ~~independence from operating pressures~~ability to perform their assigned functions.

5.2.2 ~~Unit~~Facility Staff

The ~~unit~~facility staff organization shall include the following:

a. ~~A non-licensed operator shall be assigned to each reactor containing fuel with a total of three non-licensed operators required for both units.~~Each on duty shift shall be composed of at least one Shift Supervisor shared between Units 1 and 2, and one NON-CERTIFIED OPERATOR per unit. The NON-CERTIFIED OPERATOR position may be filled by a CERTIFIED FUEL HANDLER.

b. Except for the Shift Supervisor, ~~Shift~~shift crew composition may be ~~one~~ less than the minimum requirement of ~~10 CFR 50.54(m)(2)(i) and~~ 5.2.2.a ~~and 5.2.2.f~~ for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements and all of

(continued)

5.2 Organization

5.2.2 ~~Unit~~Facility Staff (continued)

the following conditions are met:

(1) No fuel movements are in progress;

(2) No movement of loads over fuel are in progress; and

(3) No unmanned shift positions during shift turnover shall be permitted while the shift crew is less than the minimum.

c. A health physics technician shall be on site ~~when fuel is in the reactor~~during fuel handling operations and during movement of heavy loads over the fuel storage racks. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. Not used.

e. ~~The operations manager shall either hold a senior reactor operator license, have at one time held a senior reactor operator license for a pressurized water reactor, or be certified to a senior reactor operator equivalent level of knowledge. If the operations manager does not hold a senior reactor operator license, the person assigned to the Operations middle manager position shall hold a senior reactor operator license.~~ The Shift Supervisor shall be a CERTIFIED FUEL HANDLER.

f. ~~An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This position shall be manned in MODES 1, 2, 3, and 4 unless an individual with a SRO license meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.~~ At least one person qualified to stand watch in the control room (NON-CERTIFIED OPERATOR or CERTIFIED FUEL HANDLER) shall be present in the control room when nuclear fuel is stored in a spent fuel pool.

g. Oversight of fuel handling operations shall be provided by a CERTIFIED FUEL HANDLER.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit ~~Staff~~Facility Qualifications

5.3.1 Each member of the ~~plant~~facility staff shall meet or exceed the minimum qualifications referenced for comparable positions as specified in the ~~updated FSAR, Chapter 17, Quality Assurance~~Quality Assurance Program.

5.3.2 A training and retraining program for CERTIFIED FUEL HANDLERs shall be maintained.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The ~~applicable~~ procedures applicable to the safe storage of spent nuclear fuel, recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. ~~The emergency operating procedures required to implement the applicable requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33 and responses to the subject NUREGs;~~Not Used;

c. Quality assurance for effluent and environmental monitoring;

d. Not used; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

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

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

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

   2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the plant manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

(continued)

5.5 Programs and Manuals ~~(continued)~~

5.5.2 ~~Primary Coolant Sources Outside Containment~~

~~This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include portions of Recirculation Spray, Safety Injection, Chemical and Volume Control, Residual Heat Removal, RCS Sample, and Liquid and Gaseous Radwaste Treatment Systems. The program shall include the following:~~

~~a. Preventive maintenance and periodic visual inspection requirements; and~~

~~b. Integrated leak test requirements for each system at refueling cycle intervals or less.~~Not Used

5.5.3 Not Used

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

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

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

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

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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

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

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

2. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ.

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

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

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

k. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program Surveillance frequency.

5.5.5 ~~Component Cyclic or Transient Limit~~

~~This program provides controls to track the FSAR, Section 5.2 and 5.3, cyclic and transient occurrences to ensure that components are maintained within the design limits.~~
Not Used

5.5.6 Not Used

(continued)

5.5 Programs and Manuals ~~(continued)~~

5.5.7 ~~Reactor Coolant Pump Flywheel Inspection Program~~

~~This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.~~

~~In lieu of Position c.4.b(1) and c.4.b(2), a qualified in-place UT examination over the volume from the inner-bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at an interval not to exceed 20 years.~~Not Used

5.5.8 ~~Inservice Testing Program~~

~~This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:~~

~~a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:~~

~~ASME OM Code and applicable Addenda terminology for inservice testing activities~~	~~Required Frequencies for performing inservice testing activities~~
~~Weekly~~	~~At least once per 7 days~~
~~Monthly~~	~~At least once per 31 days~~
~~Quarterly or every 3 months~~	~~At least once per 92 days~~
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
~~Every 9 months~~	~~At least once per 276 days~~
~~Yearly or annually~~	~~At least once per 366 days~~
~~Biennially or every 2 years~~	~~At least once per 731 days~~

~~b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;~~

~~c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and~~

~~d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.~~ Not Used

(continued)

5.5 Programs and Manuals ~~(continued)~~

5.5.9 ~~Steam Generator (SG) Tube Inspection Program~~

~~A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:~~

~~a. Provisions for condition monitoring assessments.~~

~~Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.~~

~~b. Performance criteria for SG tube integrity.~~

~~SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.~~

~~1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.~~

~~2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Except during a SG tube rupture, leakage is also not to exceed 1 gallon per minute per SG.~~ Not Used

~~(continued)~~

~~5.5 Programs and Manuals~~

~~5.5.9 Steam Generator (SG) Tube Inspection Program (continued)~~

~~3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."~~

~~c. Provisions for SG tube repair criteria.~~

~~Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.~~

~~d. Provisions for SG tube inspections.~~

~~Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.~~

~~1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.~~

~~2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.~~

~~3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.~~

~~e. Provisions for monitoring operational primary to secondary LEAKAGE.~~

~~(continued)~~

~~5.5 Programs and Manuals (continued)~~

5.5.10 ~~Secondary Water Chemistry Program~~

~~This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:~~

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

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

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

~~d. Procedures for the recording and management of data;~~

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

~~f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.~~Not Used

5.5.11 ~~Ventilation Filter Testing Program (VFTP)~~

~~A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified below and in accordance with Regulatory Guide 1.52, Revision 2, ANSI N510 1980, and ASTM D3803-1989.~~

~~a. Demonstrate for each of the ESF systems that an in-place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested in accordance with ANSI N510-1980 at the system flowrate specified below ± 10% at least once per 24 months.~~

~~ESF Ventilation System~~	~~Flowrate~~
~~Control Room~~	~~2100 cfm~~
~~Auxiliary Building~~	~~73,500 cfm~~
~~Fuel Handling Building~~	~~35,750 cfm~~

~~b. Demonstrate for each of the ESF systems that an in-place test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with ANSI N510-1980 at the system flowrate specified below ± 10% at least once per 24 months.~~

~~ESF Ventilation System~~	~~Flowrate~~
~~Control Room~~	~~2100 cfm~~
~~Auxiliary Building~~	~~73,500 cfm~~
~~Fuel Handling Building~~	~~35,750 cfm~~

Not Used

~~(continued)~~

~~5.5 Programs and Manuals~~

~~5.5.11 Ventilation Filter Testing Program (VFTP) (continued)~~

~~c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal absorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and at the relative humidity specified below. Laboratory testing shall be completed at least once per 24 months and after every 720 hours of charcoal operation.~~

~~ESF Ventilation System~~	~~Penetration~~	~~RH~~
~~Control Room~~	~~2.5%~~	~~95%~~
~~Auxiliary Building~~	~~5.0%~~	~~95%~~
~~Fuel Handling Building~~	~~15.0%~~	~~95%~~

~~d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested in accordance with ANSI N510-1980 at the system flowrate specified below ± 10% at least once per 24 months.~~

~~ESF Ventilation System~~	~~Delta P~~	~~Flowrate~~
~~Control Room~~	~~3.5 in. WG~~	~~2100 cfm~~
~~Auxiliary Building~~	~~3.7 in. WG~~	~~73,500 cfm~~
~~Fuel Handling Building~~	~~4.1 in. WG~~	~~35,750 cfm~~

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

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in temporary unprotected outdoor liquid storage tanks.

The gaseous radioactivity quantities shall be determined following the methodology in Regulatory Guide 1.24 "Assumptions Used For Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure." The liquid radwaste quantities shall be maintained such that 10 CFR Part 20 limits are met.

~~(continued)~~

~~5.5 Programs and Manuals~~

~~5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)~~

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in temporary outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations 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, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 ~~Diesel Fuel Oil Testing Program~~

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

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

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

~~2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and~~

~~3. a clear and bright appearance with proper color; or water and sediment content within limits.~~

~~b. Other properties for ASTM 2D fuel oil are analyzed within 31 days following sampling and addition to storage tanks; and~~

~~c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A.~~

~~d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.~~Not Used

~~(continued)~~

~~5.5 Programs and Manuals (continued)~~

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. a change in the TS incorporated in the license; or

2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 ~~Safety Function Determination Program (SFDP)~~

~~This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:~~

~~a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;~~

~~b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;~~

~~c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and~~

~~d. Other appropriate limitations and remedial or compensatory actions.~~

~~A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:~~

~~a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or~~ Not Used

~~(continued)~~

~~5.5 Programs and Manuals~~

~~5.5.15 Safety Function Determination Program (SFDP) (continued)~~

~~b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or~~

~~c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.~~

~~The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~

5.5.16 ~~Containment Leakage Rate Testing Program~~

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

~~1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.~~

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.~~

~~3. The ten-year interval between performance of the integrated leakage rate (Type A) test, beginning May 4, 1994, for Unit 1 and April 30, 1993, for Unit 2, has been extended to 15 years.~~

~~b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43.5 psig. The containment design pressure is 47 psig.~~

~~c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.~~Not Used

~~(continued)~~

~~5.5 Programs and Manuals~~

~~5.5.16 Containment Leakage Rate Testing Program (continued)~~

~~d. Leakage rate acceptance criteria are:~~

~~1. Containment overall leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and Type C tests and ≤ 0.75 L_a for Type A tests;~~

~~2. Air lock testing acceptance criteria are:~~

~~a) Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.~~

~~b) For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 10 psig.~~

~~e. The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.~~

~~f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

5.5.17 ~~Battery Monitoring and Maintenance Program~~

~~This Program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer, of the following:~~

~~a. Actions to restore battery cells with float voltage < 2.13 V, and~~

~~b. Actions to equalize and test battery cells that have been discovered with electrolyte level below the top of the plates.~~Not Used

5.5.18 ~~Surveillance Frequency Control Program~~

~~This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.~~

~~a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.~~

~~b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.~~

~~c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.~~Not Used

~~5.5 Programs and Manuals (continued)~~

5.5.19 ~~Control Room Envelope Habitability Program~~

~~A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Ventilation System (CRVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. The program shall include the following elements:~~

~~a. The definition of the CRE and the CRE boundary.~~

~~b. Requirements for maintaining the CRE boundary in its design condition, including configuration control and preventive maintenance.~~

~~c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.~~

~~d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CRVS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.~~

~~e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.~~

~~f. The provisions of SR 3.0.2 are applicable to the Frequencies required by paragraphs c and d for determining CRE unfiltered inleakage and assessing CRE habitability, and measuring CRE pressure and assessing the CRE boundary.~~ Not Used

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Not Used

5.6.2 Annual Radiological Environmental Operating Report

---NOTE---
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the ~~unit~~facility during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

(continued)

5.6 Reporting Requirements ~~(continued)~~

5.6.3 Radioactive Effluent Release Report

---NOTE---

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

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

5.6.4 Not Used

5.6.5 ~~CORE OPERATING LIMITS REPORT (COLR)~~

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

~~1. Shutdown Bank Insertion Limits for Specification 3.1.5,~~

~~2. Control Bank Insertion Limits for Specification 3.1.6,~~

~~3. Axial Flux Difference for Specification 3.2.3,~~

~~4. Heat Flux Hot Channel Factor~~ ~~($F_Q(z)$)~~ ~~for Specification 3.2.1,~~

~~5. Nuclear Enthalpy Rise Hot Channel Factor~~ ~~($F^N_{\Delta H}$)~~ ~~for Specification 3.2.2,~~

~~6. SHUTDOWN MARGIN values in Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8,~~

~~7. Moderator Temperature Coefficient limits in Specification 3.1.3,~~

~~8. Refueling Boron Concentration limits in Specification 3.9.1, and~~

~~9. RCS Pressure, Temperature, and Flow DNB Limits in Specification 3.4.1.~~

Not Used

~~(continued)~~

~~5.6 Reporting Requirements~~

~~5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)~~

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

~~1. WCAP-10216-P-A, Relaxation of Constant Axial Offset Control~~ $\cancel{F_Q}$ ~~Surveillance Technical Specification, (Westinghouse Proprietary),~~

~~2. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, (Westinghouse Proprietary),~~

~~3. WCAP-8385, Power Distribution Control and Load Following Procedures, (Westinghouse Proprietary),~~

~~4. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology),"~~

~~5. WCAP-17661-P-A, Revision 1, "Improved RAOC and CAOC~~ $\cancel{F_Q}$ ~~Surveillance Technical Specifications,"~~

~~6. Not used.~~

~~7. Not used.~~

~~8. Not used.~~

~~9. WCAP-8567-P-A, "Improved Thermal Design Procedure,"~~

~~10. WCAP-16045-P-A, "Qualification of the Two Dimensional Transport Code PARAGON," and~~

~~11. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology."~~

~~(continued)~~

~~5.6 Reporting Requirements~~

~~5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)~~

~~c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.~~

~~d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.~~

5.6.6 ~~Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing, Low Temperature Overpressure Protection (LTOP) arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:~~

~~1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and~~

~~2. Specification 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~Not Used

~~(continued)~~

~~5.6 Reporting Requirements~~

~~5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)~~

~~b. The analytical methods used to determine the RCS pressure and temperature and LTOP limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:~~

~~1. WCAP 14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."~~

~~2. Chapter 6.0 of WCAP-15958, "Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program."~~

~~c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.~~

5.6.7 Not Used

5.6.8 ~~PAM Report~~

~~When a report is required by Condition B or F of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.~~Not Used

5.6.9 Not Used

~~(continued)~~

~~5.6 Reporting Requirements (continued)~~

5.6.10 ~~Steam Generator (SG) Tube Inspection Report~~

~~A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:~~

~~a. The scope of inspections performed on each SG,~~

~~b. Active degradation mechanisms found,~~

~~c. Nondestructive examination techniques utilized for each degradation mechanism,~~

~~d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,~~

~~e. Number of tubes plugged during the inspection outage for each active degradation mechanism,~~

~~f. Total number and percentage of tubes plugged to date, and~~

~~g. The results of condition monitoring, including the results of tube pulls and in-situ testing.~~ Not Used

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

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

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

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

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

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

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

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

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

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

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

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

(continued)

5.7 High Radiation Area

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

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

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

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

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

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

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

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

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

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

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

(continued)

5.7 High Radiation Area

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

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

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

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

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

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

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

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

**Proposed Changes to Technical Specification Bases – Markups
(For Information Only)**

(30 Pages)

Note – TS Bases that are deleted in their entirety are identified as such in the TS Bases Table of Contents; however, the associated deletions are not included in this attachment. The remaining TS Bases are intentionally not re-numbered.

(continued)

(continued)

For Information Only

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 ~~through LCO 3.0.8~~ and LCO 3.0.2 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the ~~unit~~ facility is in the ~~MODES or other~~ specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that~~:~~ ~~a. Completion~~ completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification~~; and~~. ~~b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.~~ ~~There are two basic types of Required Actions.~~ The ~~first type of~~ Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time ~~to restore an inoperable system or component to OPERABLE status or~~ to restore variables to within specified limits. If this ~~type of~~ Required Action is not completed within the specified Completion Time, ~~a shutdown~~action may be required to place the ~~unit~~ facility in a ~~MODE or~~ condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) ~~The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.~~ ~~Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.~~ ~~The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The~~

~~(continued)~~

BASES

~~LCO 3.0.2 (continued)~~	~~individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."~~ ~~The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.~~ ~~When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.~~
~~LCO 3.0.3~~	~~LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:~~ ~~a. An associated Required Action and Completion Time is not met and no other Condition applies; or~~ ~~b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.~~ ~~This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be~~

(continued)

BASES

~~LCO 3.0.3 (continued)~~

~~maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.~~

~~Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.~~

~~A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:~~

~~a. The LCO is now met.~~

~~b. A Condition exists for which the Required Actions have now been performed.~~

~~c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.~~

~~The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.~~

(continued)

BASES

~~LCO 3.0.3 (continued)~~	~~In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.~~ ~~Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.15, "Fuel Storage Pool Water Level." LCO 3.7.15 has an Applicability of "During movement of irradiated fuel assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.15 are not met while in MODE 1, 2, 3, or 4, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.15 of "Suspend movement of irradiated fuel assemblies in the fuel storage pool" is the appropriate Required Action to complete in lieu of the ACTIONS of LCO 3.0.3. These exceptions are addressed in the individual Specifications.~~
~~LCO 3.0.4~~	~~LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.~~ ~~LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.~~ ~~LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.~~

(continued)

BASES

~~LCO 3.0.4 (continued)~~

~~The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities to be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4 (b), must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.~~

~~LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.~~

~~The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions.~~

(continued)

BASES

~~LCO 3.0.4 (continued)~~

~~The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.~~

~~LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., Containment Air Temperature, Containment Pressure, MCPR, Moderator Temperature Coefficient), and may be applied to other Specifications based on NRC plant-specific approval.~~

~~The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.~~

~~The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that results from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, MODE 4 to MODE 5, and MODE 5 to MODE 6.~~

(continued)

BASES

~~LCO 3.0.4 (continued)~~

~~Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.~~

~~Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.~~

(continued)

For Information Only

BASES ~~(continued)~~

~~LCO 3.0.5~~	~~LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s) to allow the performance of required testing to demonstrate:~~ ~~a. The OPERABILITY of the equipment being returned to service;~~ ~~OR~~ ~~b. The OPERABILITY of other equipment.)~~ ~~The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.~~ ~~An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.~~ ~~An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.~~
~~LCO 3.0.6~~	~~LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.~~ ~~When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to~~

(continued)

BASES

~~LCO 3.0.6 (continued)~~	~~do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.~~ ~~However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.~~ ~~Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6. Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~
~~LCO 3.0.7~~	~~There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCO 3.1.8, allows specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to~~

(continued)

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~~LCO 3.0.7 (continued)~~	~~be changed to perform the special test or operation will remain in effect.~~ ~~The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.~~
~~LCO 3.0.8~~	~~LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.~~ ~~If allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.~~ ~~LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.~~

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BASES

~~LCO 3.0.8 (continued)~~	~~LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.~~ ~~LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.~~

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the ~~MODES or other~~ specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify ~~the OPERABILITY of systems and components, and~~ that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. ~~Systems and components are assumed to be OPERABLE when the associated SRs have been met.~~ Variables are assumed to be within limits when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that variables are within limits when ~~systems or components are OPERABLE when:~~

~~a. The systems or components are known to be inoperable, although still meeting the SRs; or~~

~~b. The~~ the requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the facility~~unit~~ is in a ~~MODE or other~~ specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. ~~The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.~~

~~Unplanned events may satisfy the requirements for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.~~

~~Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.~~

~~Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent~~

(continued)

BASES

SR 3.0.1 (continued)

~~performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.~~

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances. ~~and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.~~
SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers ~~plant operating~~ facility conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. ~~The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the interval cannot be extended by the TS, and the SR include a Note in the Frequency stating, “SR 3.0.2 is not applicable.” An example of an exception when the test interval is not specified in the regulations is the Note in the Containment Leakage Rate Testing Program, “SR 3.0.2 is not applicable.” This exception is provided because the program already includes extension of test interval.~~

~~As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse~~

(continued)

BASES

SR 3.0.2 (continued)

~~components or accomplishes the function of the inoperable equipment in an alternative manner.~~
The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals ~~(other than those consistent with refueling intervals)~~ or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring ~~affected equipment inoperable or~~ an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. SR 3.0.3 is only acceptable if it is discovered that a Surveillance was not performed after the specified Frequency had already expired. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of facility~~unit~~ conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified facility~~unit~~ conditions, ~~operating situations,~~ or requirements of regulations (e.g., ~~prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J,~~ as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

~~SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.~~

(continued)

BASES

SR 3.0.3 (continued)	Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. ~~The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action.~~ All missed Surveillances will be placed in the Corrective Action Program.
	If a Surveillance is not completed within the allowed delay period, then the ~~equipment is considered inoperable or the~~ variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then ~~the equipment is inoperable, or~~ the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.
	Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

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BASES (continued)

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a ~~MODE or other~~ specified condition in the Applicability.

This Specification ensures that ~~system and component OPERABILITY requirements and~~ variable limits are met before entry into ~~MODES or~~ other specified conditions in the Applicability for which these ~~systems and components ensure safe operation of the unit~~limits ensure facility safety. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring ~~systems or components to OPERABLE status~~ variable limits before entering ~~an associated MODE or other~~ another specified condition in the Applicability.

~~A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met in accordance with LCO 3.0.4.~~

~~However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided that requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.~~

The provisions of SR 3.0.4 shall not prevent entry into ~~MODES or~~ other specified conditions in the Applicability that are required to comply with ACTIONS. ~~In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, MODE 4 to MODE 5, and MODE 5 to MODE 6.~~

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BASES

SR 3.0.4 (continued)	The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the ~~MODE or other~~ specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. ~~A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.~~

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the spent fuel pool design is given in the DSAR~~UFSAR~~, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the ~~UFSAR~~DSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the ~~UFSAR~~DSAR, Sections ~~15.4.5 and 15.5.22~~15.4 and 15.5 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The design basis fuel handling accident analysis is evaluated in accordance with the guidance in Regulatory Guide 1.183 (Reference 4). The minimum water level in the spent fuel pool ~~meets the assumptions of the fuel handling accident~~ensures ~~described in Regulatory Guide 1.183 (Ref. 4)~~. t~~T~~he resultant ~~2 hour dose per person at the exclusion area boundary is a small fraction of the 10 CFR 50.67 (Ref. 5) limits.~~dose consequences remain below the acceptance criteria.

The fuel handling accident is assumed to occur during handling of a spent fuel assembly in the SFP. One fuel assembly is damaged, releasing all of the fuel gap activity associated with that assembly. The activity (consisting of noble gases, halogens, and alkali metals) is released in a “puff” to the SFP, which has a minimum of 21 feet of water above the damaged fuel assembly. A halogen decontamination factor of 142 is applied to the gap release (for all halogen species) to account for the scrubbing effect of the 21 feet of water above the damaged fuel assembly. The listed DF is developed using guidance provided in Reference B-1 of RG 1.183 (Reference 4). Noble gas and un-scrubbed iodines rise to the water surface where they are mixed in the minimum available air space in the FHB above the SFP. Per Appendix B of RG 1.183, Revision 0 (Reference 4), all of the alkali metals are retained in the SFP water.

~~According to Reference 4, there is 23 ft of water between the top of the damaged fuel rods and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. Although there are other spent fuel pool elevations where fuel handling accidents can occur, the design basis fuel handling accident, which uses the conservative assumptions of RG 1.183, is expected to be bounding. To add conservatism, the analysis assumes that all fuel rods of the damaged fuel assembly fail.~~

In practice, the water level maintained for fuel handling provides more than 23 feet of water over the top of irradiated fuel assemblies seated in the storage racks. ~~UFSAR~~ DSAR Section 9.1.4.3.6 requires the water level provide a minimum of 8 feet of water shielding during fuel handling. This assures more than 24 feet 6 inches of water shielding over the top of the fuel assemblies in the racks and more than 23 feet of water shielding over a fuel assembly lying horizontally on top of the racks.

The spent fuel pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The spent fuel pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Reference~~.~~ 5~~3~~). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

(continued)

BASES ~~(continued)~~

APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool, since the potential for a release of fission products exists.
ACTIONS	A.1 ~~Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.~~ When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pool water level is lower than the required level, the movement of irradiated fuel assembly in the spent fuel pool is immediately suspended. This does not preclude movement of a fuel assembly to a safe position. ~~If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~
SURVEILLANCE REQUIREMENTS	SR 3.7.15.1 This SR is done during the movement of irradiated fuel assemblies as stated in the Applicability. This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk~~ and is controlled under the Surveillance Frequency Control Program~~. ~~During refueling operations, the level in the spent fuel pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.7.1.~~
REFERENCES	1. ~~UFSAR~~DSAR, Section 9.1.2. 2. ~~UFSAR~~DSAR, Section 9.1.3. 3. ~~UFSAR~~DSAR, Sections ~~9.1.4.3.6, 15.4.5 and 15.5.22~~15.4 and 15.5. 4. Regulatory Guide 1.183, July 2000 5. PG&E Calculation Number DECOM-N-0003, (Vendor Calculation 14078100-C-M-00002), Dose Consequences at the site Boundary, Existing Control Room, Proposed Alternate Control Rooms and the TSC following a Fuel Handling Accident in the Fuel Handling Building. ~~10 CFR 50.67~~.

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

The DCPP Units 1 and 2 spent fuel pools (Ref. 4) each consist of 16 permanent stainless steel racks of various sizes, with a total of 1,324 fuel assembly storage cells. ~~For cycles 14 – 16, in addition to the 16 racks, a cask pit storage rack with a capacity of 154 cells is installed in each spent fuel pool's cask pit area. This extra rack expands the total storage capacity in each spent fuel pool to 1,478 fuel assembly storage cells.~~ The spent fuel pool storage racks have been analyzed for the storage of fuel assemblies that meet the requirements of LCO 3.7.17. ~~The fuel storage capacity during cycles 14 – 16 is 1478 assemblies, of which 1433 assemblies may be irradiated. Unfilled cells in the permanent storage racks may be utilized for the storage of unirradiated fuel assemblies. The limitation of a maximum of 1433 irradiated assemblies is based on the spent fuel pool bulk temperature analysis for cycles 14 – 16, while the cask pit storage rack is installed.~~

10 CFR 50.68(b)(4), requires that the spent fuel storage racks are designed to assure that with credit for soluble boron and with fuel of the maximum fuel assembly reactivity, a K_{eff} of less than or equal to 0.95 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with borated water, and a K_{eff} of less than 1.0 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with unborated water.

Criticality analyses have been performed for the permanent storage racks (Ref. 3 and 5) ~~and for the cask pit storage rack (Ref 8)~~, which demonstrate that the multiplication factor, k_{eff}, of the fuel assemblies in the spent fuel storage racks is less than or equal to 0.95. In order to maintain k_{eff} less than or equal to 0.95, the presence of soluble boron is credited in the spent fuel pool criticality analyses.

For the permanent storage racks, Reference 5 provides the analysis for the 2x2 array and checkerboard configurations, and Reference 3 provides the analysis for the all cell configuration. Both criticality analyses (Ref. 3 and 5) evaluate the region of the spent fuel pool that does not contain any Boraflex panels because the storage requirements are more restrictive and yield more conservative reactivity results than the region containing Boraflex. The results of the analyses may be conservatively applied to the less reactive region.

(continued)

BACKGROUND (continued)

~~Reference 8 provides the analysis for the cask pit storage rack. Storage configurations were defined in the criticality analysis (Ref. 8) to ensure that k_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95. A minimum soluble boron concentration of 500 ppm is required to maintain k_{eff} less than or equal to 0.95 under normal storage conditions including tolerances and uncertainties, which is well within the 2000 ppm requirement of LCO 3.7.16.~~

The criticality analyses considered accident conditions (Ref. 3, and 5~~, and 8~~). Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95 and to mitigate the worst accident reactivity insertion. For the permanent storage racks, a soluble boron concentration of 806 ppm is required to maintain k_{eff} less than or equal to 0.95 for all allowable storage configurations, which is well within the 2000 ppm requirement of LCO 3.7.16. ~~For the cask pit storage rack, a soluble boron concentration of 800 ppm is required to maintain keff less than or equal to 0.95 for all allowable storage configurations, which is well within the 2000 ppm requirement of LCO 3.7.16.~~

For such an occurrence, the double contingency principle of ANSI N16.1-1975 and the April 1978 NRC letter (Ref. 1) can be applied. The NRC letter states it is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for such a postulated reactivity insertion accident condition, the presence of soluble boron in the spent fuel pool water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

In addition to consideration of spent fuel pool criticality, a boron dilution analysis (Ref. 6) was performed to evaluate the time and water volumes required to dilute the spent fuel pool from 2000 to 800 ppm.

The results of the boron dilution analysis concluded that an unplanned or inadvertent event that would result in the dilution of the spent fuel pool boron concentration from 2000 ppm to 800 ppm is not a credible event since a dilution event large enough to result in a significant reduction in the spent fuel pool boron concentration would involve the transfer of a large quantity of water from a dilution source and a significant increase in spent fuel pool level, which would ultimately overflow the pool. The overflow of the spent fuel pool would be readily detected and terminated by plant personnel. In addition, because of the large quantities of water required and the low dilution flow rates available, any significant dilution of the spent fuel pool boron concentration would only occur over a long period of time (hours to days).

(continued)

BASES

BACKGROUND (continued)	Detection of a spent fuel pool boron dilution via pool level alarms, visual inspection during normal operator rounds, significant changes in spent fuel pool boron concentration, or significant changes in the unborated water source volume, would be expected before a dilution event sufficient to increase K_{eff} above 0.95 could occur. However, for the permanent storage racks analyses have been performed to demonstrate that even if the spent fuel pool boron concentration was diluted to zero ppm, which would take significantly more water than evaluated in the boron dilution analysis, the spent fuel would be expected to remain subcritical and the health and safety of the public would be assured.
APPLICABLE SAFETY ANALYSES	Most accident conditions result in a negligible reactivity effect in the spent fuel pool (Ref. 3, and 5~~, and 8~~). However, scenarios can be postulated that could have more than a negligible positive reactivity effect. Examples of such accident scenarios for the permanent storage racks are the misplacement of a fuel assembly, a significant increase in spent fuel pool temperature above 150°F, or a cask drop accident (Ref. 4). A soluble boron concentration of 806 ppm is required to maintain k_{eff} less than or equal to 0.95 under accident conditions, which is well within the 2000 ppm requirement of LCO 3.7.16. ~~Examples of accident scenarios for the cask pit storage rack are the misplacement of a fuel assembly and a dropped assembly. A soluble boron concentration of 800 ppm is required to maintain k_{eff} less than or equal to 0.95 under accident conditions, which is well within the 2000 ppm requirement of LCO 3.7.16.~~ The negative reactivity effect of the soluble boron more than compensates for the increased reactivity caused by the postulated accident scenarios. The accident analyses is provided in the ~~UFSAR~~ DSAR (Ref. 4). The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The spent fuel pool boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in References 3, 4, and 5~~, and 8~~. The specified boron concentration of 2000 ppm ensures that the spent fuel pool k_{eff} will remain less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level, for a postulated reactivity insertion accident or boron dilution event.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

(continued)

BASES ~~(continued)~~

ACTIONS

A.1 and A.2

~~The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.~~

When the concentration of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies and immediately taking actions to restore the spent fuel pool boron concentration to greater than or equal to 2000 ppm. This suspension of fuel movement does not preclude movement of fuel assemblies to a safe position.

~~If the LCO is not met while moving fuel assemblies LCO 3.0.3 would not be applicable since the inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by chemical analysis that the concentration of boron in the spent fuel pool is at or above the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk~~ and is controlled under the Surveillance Frequency Control Program~~.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
2. Not used.
3. "Criticality Safety Evaluation of Region 2 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment," S.E.Turner, October 1993, Holtec Report HI-931077.
4. D~~UF~~SAR, Sections 9.1, 15.4~~.5~~, and 15.5~~.22~~.
5. "Diablo Canyon Units 1 and 2 Spent Fuel Criticality Analysis," February 14, 2001, Paul F. O'Donnell, Westinghouse Doc. No. A-DP1-FE-0001.
6. "Diablo Canyon Units 1 and 2 Spent Fuel Pool Boron Dilution Source Analysis," M-1188 (SAP Calculation No, 9000041656), December 2016
7. License Amendment 154/154, September 25, 2002.
8. ~~"Spent Fuel Storage Expansion at Diablo Canyon Units 1 & 2 for Pacific Gas & Electric Co.", October 2004, Holtec Report HI-2043162.~~Not used.
9. ~~License Amendment 183/185, November 21, 2005.~~Not used.

For Information Only

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

The DCPP Units 1 and 2 spent fuel pools (Ref. 2) each consist of 16 stainless steel racks of various sizes, with a total of 1,324 fuel assembly storage cells. ~~For cycles 14 – 16, in addition to the 16 permanent racks, a cask pit storage rack with a capacity of 154 cells is installed in each spent fuel pool's cask pit area. This extra rack expands the total storage capacity in each spent fuel pool to 1,478 fuel assembly storage cells.~~ The spent fuel pool storage racks have been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.17. ~~The fuel storage capacity during cycles 14 – 16 is 1478 assemblies, of which 1433 assemblies may be irradiated.~~ Unfilled cells in the permanent storage racks may be utilized for the storage of unirradiated fuel assemblies. ~~The limitation of a maximum of 1433 irradiated assemblies is based on the spent fuel pool bulk temperature analysis for cycles 14 – 16, while the cask pit storage rack is installed.~~ The 16 permanent spent fuel storage racks are designed to accommodate three different storage configurations as shown in Figure 3.7.17-1. ~~The cask pit fuel storage rack is designed to accommodate only fuel with an initial enrichment of ≤ 4.1 weight % U-235, a minimum 10 year decay time and a discharge burnup in the acceptable region of Figure 3.7.17-4.~~

10 CFR 50.68(b)(4), requires that the spent fuel storage racks are designed to assure that with credit for soluble boron and with fuel of the maximum fuel assembly reactivity, a K_{eff} of less than or equal to 0.95 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with borated water, and a K_{eff} of less than 1.0 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with unborated water.

Criticality analyses have been performed (Ref. 3, and 4~~, and 6~~) which demonstrate that the multiplication factor, k_{eff}, of the fuel assemblies in the spent fuel storage racks is less than or equal to 0.95. In order to maintain k_{eff} less than or equal to 0.95, the presence of soluble boron is credited in the spent fuel pool criticality analysis. Reference 3 provides the analysis for the 2x2 array and checkerboard configurations, and Reference 4 provides the analysis for the all cell configuration~~, and Reference 6 provides the analysis for the cask pit storage rack~~.

(continued)

BASES

BACKGROUND (continued)	For the 16 permanent storage racks, both criticality analyses (Ref. 3 and 4) evaluate the region of the spent fuel pool that does not contain any Boraflex panels because the storage requirements are more restrictive and yield more conservative reactivity results than the region containing Boraflex. The results of the analyses may be conservatively applied to the less reactive region. A discussion of how soluble boron is credited for the storage of fuel assemblies in the spent fuel pool is contained in the background for TS 3.7.16 Bases. Storage configurations were defined in the criticality analyses (Ref. 3, and 4~~, and 6~~) to ensure that k_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95 and to mitigate the worst accident reactivity insertion. For the permanent storage racks, a soluble boron concentration of 806 ppm is required to maintain k_{eff} less than or equal to 0.95 for all allowable storage configurations, which is well within the 2000 ppm requirement of LCO 3.7.16. ~~For the cask pit storage rack, a soluble boron concentration of 800 ppm is required to maintain k_{eff} less than or equal to 0.95 for all allowable storage configurations, which is well within the 2000 ppm requirement of LCO 3.7.16.~~ Prior to movement of an assembly, it is necessary to verify that SR 3.7.16.1 is current.
APPLICABLE SAFETY ANALYSES	For the permanent storage racks, the analyzed accidents that could have significant reactivity effects are the misplacement of a fuel assembly, a significant increase in spent fuel pool temperature above 150°F, or a cask drop accident (Ref. 2, 3, and 4). ~~For the cask pit storage rack, accidents that could have significant reactivity effects are misplacement of a fuel assembly and a dropped assembly (Ref. 6).~~ For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.16, "Spent Fuel Pool Boron Concentration") ensures that k_{eff} will remain at or below 0.95. The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with LCO 3.7.17, ensure the k_{eff} of the spent fuel storage pool will always remain ≤ 0.95 at a 95 percent probability, 95 percent confidence level, for a postulated reactivity insertion accident or a boron dilution event.
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in the spent fuel pool.

(continued)

BASES ~~(continued)~~

ACTIONS	A.1 ~~The Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since the inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.~~ When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with LCO 3.7.17, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with LCO 3.7.17 which will return the fuel pool to an analyzed condition.
SURVEILLANCE REQUIREMENTS	SR 3.7.17.1 This SR verifies by administrative means that each fuel assembly and its expected storage location are in accordance with LCO 3.7.17 prior to each fuel assembly move when the assembly is to be stored in the spent fuel pool. A complete record of initial enrichment, initial integral boron content, fuel pellet diameter, and the cumulative burnup analysis shall be maintained for the time period that each fuel assembly remains in the spent fuel pool. ~~In addition, for fuel assemblies stored in the cask pit storage rack, the record will also include fuel assembly decay-time.~~
REFERENCES	1. Not used. 2. ~~UFSAR~~DSAR, Sections 9.1, 15.4~~.5~~, and 15.5~~.22~~. 3. "Diablo Canyon Units 1 and 2 Spent Fuel Pool Criticality Analysis," February 14, 2001, Paul F. O'Donnell, Westinghouse Doc. No. A-DP1-FE-0001. 4. "Criticality Safety Evaluation of Region 2 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment," S.E.Turner, October 1993, Holtec Report HI-931077. 5. License Amendment 154/154, September 25, 2002. 6. ~~"Diablo Canyon Units 1 and 2 Spent Fuel Storage Expansion Licensing Report", October 2004, Holtec Report HI – 2043162.~~Not used. 7. ~~License Amendment 183/185, November 21, 2005.~~Not used.

Proposed Facility Operating License Changes (DPR-80 and DPR-82) – Clean

(17 Pages)

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for licenses by Pacific Gas and Electric Company complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Deleted per Amendment No. ###.

   C. The facility will be maintained in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   D. There is reasonable assurance (i) that the activities authorized by this license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter I;

   E. The Pacific Gas and Electric Company is technically qualified to engage in the activities authorized by this license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. The Pacific Gas and Electric Company has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements", of the Commission's regulations;

   G. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility License No. DPR-80, subject to the conditions for protection of the environment set forth herein, is in accordance with applicable Commission regulations governing environmental reviews (10 CFR Part 50, Appendix D and 10 CFR Part 51) and all applicable requirements have been satisfied; and

   I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Pursuant to Commission's Memorandum and Order CLI-84-13, dated August 10, 1984, Facility Operating License No. DPR-76 issued September 22, 1981, as subsequently amended, is superseded by Facility License No. DPR-80, hereby issued to Pacific Gas and Electric Company to read as follows:

   A. This License applies to the Diablo Canyon Nuclear Power Plant, Unit 1, a pressurized water nuclear reactor and associated equipment (the facility), owned by the Pacific Gas and Electric Company (PG&E). The facility is located in San Luis Obispo County, California, and is described in PG&E's Defueled Safety Analysis Report as supplemented and amended, and the Environmental Report as supplemented and amended.

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Pacific Gas and Electric Company:

      (1) Pursuant to Section 104(b) of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess and use the facility at the designated location in San Luis Obispo County, California, in accordance with the procedures and limitations set forth in this license;

      (2) Pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Defueled Safety Analysis Report, as supplemented and amended;

      (3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in the calibration of radiation monitoring equipment, and as fission detectors in amounts as required;

      (4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

      (5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials that were produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Deleted per Amendment No. ###.

(2) Permanently Defueled Technical Specifications

The Permanently Defueled Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. ###, are hereby incorporated in the license. Pacific Gas and Electric Company shall maintain the facility in accordance with the Permanently Defueled Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Deleted per Amendment No. ###.

(4) Deleted per Amendment No. ###.

(5) Fire Protection

a. PG&E shall implement and maintain all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the PG&E amendment request dated June 26, 2013, as supplemented by letters dated October 3, 2013, September 29, 2014, October 27, 2014, October 29, 2014, November 26, 2014, and December 31, 2014; February 25, 2015 (two letters), May 7, 2015, October 15, 2015, and December 31, 2015; and January 28, 2016, and as approved in the safety evaluation dated April 14, 2016. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, PG&E may make changes to the Fire Protection Program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

b. Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of a change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at DCPP. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed Fire Probabilistic Risk Assessment model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact:

(1) Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(2) Prior NRC review and approval is not required for individual changes that result in a risk increase less than $1x10^{-7}$/year (yr) for CDF and less than $1\ x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

c. Other Changes that May Be Made Without Prior NRC Approval

(1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental Fire Protection Program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. PG&E may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change will not affect the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

PG&E may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change will not affect the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and,
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

(2) Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to PG&E's Fire Protection Program that have been demonstrated to have no more than a minimal risk impact. PG&E may use its screening process as approved in the NRC safety evaluation dated April 14, 2016, to determine that certain Fire Protection Program changes meet the minimal criterion. PG&E shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the Fire Protection Program.

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

d. Transition License Conditions:

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) and (3) below, risk-informed changes to PG&E's Fire Protection Program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in c.(2) above.

(2) PG&E shall implement the modifications described in Attachment-S, Table S-2, "Plant Modifications Committed," of PG&E Letter DCL-16-014, dated January 28, 2016, by the end of the Units 1 and 2 refueling outages currently

scheduled for April/May 2017 (1R20) and February/March 2018 (2R20). PG&E shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

(3) PG&E shall implement the items as listed in Attachment-S, Table S-3, "Implementation Items," of PG&E Letter DCL-16-014, dated January 28, 2016, within 365 days after receipt of the safety evaluation/license amendment with the exception of Implementation Item S-3.24, which will be completed for each unit within 90 days after all modifications for the respective unit are operable (as listed in Attachment S, Table S-2).

(6) Deleted per Amendment No. ###.

(7) <u>Seismic Design Bases Reevaluation Program (SSER 27 Section IV.5)</u>

PG&E shall develop and implement a program to reevaluate the seismic design bases used for the Diablo Canyon Nuclear Power Plant.

The program shall include the following Elements:

(1) PG&E shall identify, examine, and evaluate all relevant geologic and seismic data, information, and interpretations that have become available since the 1979 ASLB hearing in order to update the geology, seismology and tectonics in the region of the Diablo Canyon Nuclear Power Plant. If needed to define the earthquake potential of the region as it affects the Diablo Canyon Plant, PG&E will also reevaluate the earlier information and acquire additional new data.

(2) PG&E shall reevaluate the magnitude of the earthquake used to determine the seismic basis of the Diablo Canyon Nuclear Plant using the information from Element 1.

(3) PG&E shall reevaluate the ground motion at the site based on the results obtained from Element 2 with full consideration of site and other relevant effects.

(4) PG&E shall assess the significance of conclusions drawn from the seismic reevaluation studies in Elements 1, 2 and 3, utilizing a probabilistic risk analysis and deterministic studies, as necessary, to assure adequacy of seismic margins.

PG&E shall submit for NRC staff review and approval a proposed program plan and proposed schedule for implementation by January 30, 1985. The program shall be completed and a final report submitted to the NRC three years following the approval of the program by the NRC staff.

PG&E shall keep the staff informed on the progress of the reevaluation program as necessary, but as a minimum will submit quarterly progress reports and arrange for semi-annual meetings with the staff. PG&E will also keep the ACRS informed on the progress of the reevaluation program as necessary, but not less frequently than once a year.

(8) Deleted per Amendment No. ###.

(9) Deleted per Amendment No. ###.

(10) Deleted per Amendment No. ###.

(11) Deleted per Amendment No. ###.

(12) Deleted per Amendment No. ###.

(13) Aging Management Program

If all spent fuel has not been removed from the Unit 1 spent fuel pool prior to November 2, 2028, an aging management program shall be submitted prior to this date for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Defueled Safety Analysis Report and shall remain in effect for Unit 1 until such time that all spent fuel has been removed from the Unit 1 spent fuel pool.

D. Deleted per Amendment No. ###.

E. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54 (p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Diablo Canyon Power Plant, Units 1 and 2 Physical Security Plan, by Training and Qualification Plan, and Safeguards Contingency Plan," submitted by letter dated May 16, 2006.

PG&E shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The PG&E

CSP was approved by License Amendment No. 210, as supplemented by a change approved by License Amendment No. 220.

Pursuant to NRC's Order EA-13-092, dated June 5, 2013, NRC reviewed and approved the license amendment 222 that permitted the security personnel of the licensee to possess and use certain specific firearms, ammunition, and other devices, such as large-capacity ammunition feeding devices, notwithstanding local, State, and certain Federal firearms laws that may prohibit such possession and use.

F. Deleted.

G. Deleted.

H. Financial Protection

PG&E shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

I. Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

J. Term of License

This License is effective as of the date of issuance and is effective until the Commission notifies the licensee in writing that the license is terminated.

Attachments:

1. Appendix A – Permanently Defueled Technical Specifications
2. Appendix B – Environmental Protection Plan
3. Appendix C – Deleted
4. Appendix D – Deleted

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for licenses by Pacific Gas and Electric Company (PG&E) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Deleted per Amendment No. ###.

   C. The facility will be maintained in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   D. There is reasonable assurance (i) that the activities authorized by this license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter I;

   E. The Pacific Gas and Electric Company is technically qualified to engage in the activities authorized by this license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. The Pacific Gas and Electric Company has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements", of the Commission's regulations;

   G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

   H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility License No. DPR-82, subject to the conditions for protection of the environment set forth herein, is in accordance with applicable Commission regulations governing environmental reviews (10 CFR Part 50, Appendix D and 10 CFR Part 51) and all applicable requirements have been satisfied; and

   I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Pursuant to approval by the Nuclear Regulatory Commission in its Memorandum and Order (CLI-85-14) dated August 1, 1985, the license for fuel loading and low power testing, Facility Operating License No. DPR-81, issued on April 26, 1985, is superseded by Facility License No. DPR-82, hereby issued to Pacific Gas and Electric Company to read as follows:

A. This License applies to the Diablo Canyon Nuclear Power Plant, Unit 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by PG&E. The facility is located in San Luis Obispo County, California, and is described in PG&E's Defueled Safety Analysis Report as supplemented and amended, and the Environmental Report as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission herby licenses the Pacific Gas and Electric Company:

(1) Pursuant to Section 104(b) of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess and use the facility at the designated location in San Luis Obispo County, California, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Defueled Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in the calibration of radiation monitoring equipment, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials that were produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Deleted per Amendment No. ###.

(2) Permanently Defueled Technical Specifications

The Permanently Defueled Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. ###, are hereby incorporated in the license. Pacific Gas and Electric Company shall maintain the facility in accordance with the Permanently Defueled Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Deleted per Amendment No. ###.

(4) Fire Protection

a. PG&E shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the PG&E amendment request dated June 26, 2013, as supplemented by letters dated October 3, 2013, September 29, 2014, October 27, 2014, October 29, 2014, November 26, 2014, and December 31, 2014, February 25, 2015 (two letters), May 7, 2015, October 15, 2015, and December 31, 2015; and January 28, 2016, and as approved in the safety evaluation dated April 14, 2016. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, PG&E may make changes to the Fire Protection Program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

b. Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of a change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at DCPP. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed Fire Probabilistic Risk Assessment model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact:

(1) Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(2) Prior NRC review and approval is not required for individual changes that result in a risk increase less than $1x10^{-7}$/year (yr) for CDF and less than $1 x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

c. Other Changes that May Be Made Without Prior NRC Approval

(1) Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program

Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental Fire Protection Program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. PG&E may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3, element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change will not affect the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

PG&E may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3, elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change will not affect the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

- "Fire Alarm and Detection Systems" (Section 3.8);
- "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
- "Gaseous Fire Suppression Systems" (Section 3.10); and,
- "Passive Fire Protection Features" (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

(2) Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval are not required for changes to PG&E's Fire Protection Program that have been demonstrated to have no more than a minimal risk impact. PG&E may use its screening process as approved in the NRC safety evaluation dated April 14, 2016, to determine that certain Fire Protection Program changes meet the minimal criterion. PG&E shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the Fire Protection Program.

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

d. Transition License Conditions:

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) and (3) below, risk-informed changes to PG&E's Fire Protection Program may not be made without prior NRC review and approval unless the change has been

demonstrated to have no more than a minimal risk impact, as described in c.(2) above.

(2) PG&E shall implement the modifications described in Attachment-S, Table S-2, "Plant Modifications Committed," of PG&E Letter DCL-16-014, dated January 28, 2016, by the end of the Units 1 and 2 refueling outages currently scheduled for April/May 2017 (1R20) and February/March 2018 (2R20). PG&E shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

(3) PG&E shall implement the items as listed in Attachment-S, Table S-3, "Implementation Items," of PG&E Letter DCL-16-014, dated January 28, 2016, within 365 days after receipt of the safety evaluation/license amendment with the exception of Implementation Item S-3.24, which will be completed for each unit within 90 days after all modifications for the respective unit are operable (as listed in Attachment S, Table S-2).

(5) Deleted per Amendment No. ###.

(6) Deleted per Amendment No. ###.

(7) Deleted per Amendment No. ###.

(8) Deleted per Amendment No. ###.

(9) Deleted per Amendment No. ###.

(10) Deleted per Amendment No. ###.

(11) Deleted per Amendment No. ###.

(12) Deleted per Amendment No. ###.

(13) Aging Management Program

If all spent fuel has not been removed from the Unit 2 spent fuel pool prior to August 26, 2029, an aging management program shall be submitted prior to this date for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Defueled Safety Analysis Report and shall remain in effect for Unit 2 until such time that all spent fuel has been removed from the Unit 2 spent fuel pool.

D. Deleted per Amendment No. ###.

E. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provision of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Diablo Canyon Power Plant, Units 1 and 2 Physical Security Plan, Training and Qualification Plan and Safeguards Contingency Plan," submitted by letter dated May 16, 2006.

PG&E shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The PG&E CSP was approved by License Amendment No. 212, as supplemented by a change approved by License Amendment No. 222.

Pursuant to NRC's Order EA-13-092, dated June 5, 2013, NRC reviewed and approved the license amendment 224 that permitted the security personnel of the licensee to possess and use certain specific firearms, ammunition, and other devices, such as large-capacity ammunition feeding devices, notwithstanding local, State, and certain Federal firearms laws that may prohibit such possession and use.

F. Deleted.

G. Deleted.

H. Financial Protection

PG&E shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

I. Mitigation Strategy

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets

3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

J. <u>Term of License</u>

This License is effective as of the date of issuance and is effective until the Commission notifies the licensee in writing that the license is terminated.

Attachments:
1. Appendix A – Permanently Defueled Technical Specifications
2. Appendix B – Environmental Protection Plan
3. Appendix C – Deleted
4. Appendix D – Deleted

Proposed Changes to Appendix A, Technical Specifications – Clean

(33 Pages)

1.0 USE AND APPLICATION

1.1 Definitions

---NOTE---
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with the provisions of the CERTIFIED FUEL HANDLER Training and Retraining Program required by Specification 5.3.2.
NON-CERTIFIED OPERATOR	A NON-CERTIFIED OPERATOR is an operator who complies with the qualification requirements of Specification 5.3.1, but is not a CERTIFIED FUEL HANDLER.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe handling and storage of nuclear fuel. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability.
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.
EXAMPLES	The following examples illustrate the type of frequency statements that appear in the Permanently Defueled Technical Specifications (PDTS).

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify level is within limits.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the PDTS. The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when a variable is outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified, then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify ...	Prior to each fuel assembly move…

Example 1.4-2 illustrates a one time performance Frequency.

This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2.

2.0 Deleted

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS.

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Water Level

LCO 3.7.15 The spent fuel pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies in the spent fuel pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Pool Boron Concentration

LCO 3.7.16 The spent fuel pool boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify the spent fuel pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17 Fuel assembly storage in the spent fuel pool shall be maintained such that:

a. In the permanent spent fuel storage racks any four cells shall be in a configuration as shown in Figure 3.7.17-1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly into an acceptable storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means that the fuel assembly characteristics and its expected storage location is in accordance with LCO 3.7.17.	Prior to each fuel assembly move, when the assembly will be stored in the spent fuel pool.

A	A
A	A

All Cell Configuration

F	B
B	B

2x2 Array Configuration

G	W
W	G

Checkerboard Configuration

<u>All Cell:</u>

A Fuel assembly with a discharge burnup in the “acceptable” region of Figure 3.7.17-2.

<u>2x2 Array:</u>

F (a) Fuel assembly with an initial enrichment ≤ 4.9 wt% U-235; or

(b) Fuel assembly with an initial enrichment ≤ 5.0 wt% U-235 and an IFBA loading equivalent to 16 rods each with 1.5 mg ^{10}B/in over 120 inches.

B Fuel assembly with a discharge burnup in the “acceptable” region of Figure 3.7.17-3.

<u>Checkerboard:</u>

G Fuel assembly with an initial enrichment ≤ 5.0 wt% U-235.

W Water cell – locations where fuel assemblies are not present, non-fissile components are permitted.

FIGURE 3.7.17-1
ALLOWABLE STORAGE CONFIGURATIONS
(ALL CELL, 2X2 ARRAY, CHECKERBOARD)
FOR THE PERMANENT SPENT FUEL POOL STORAGE RACKS

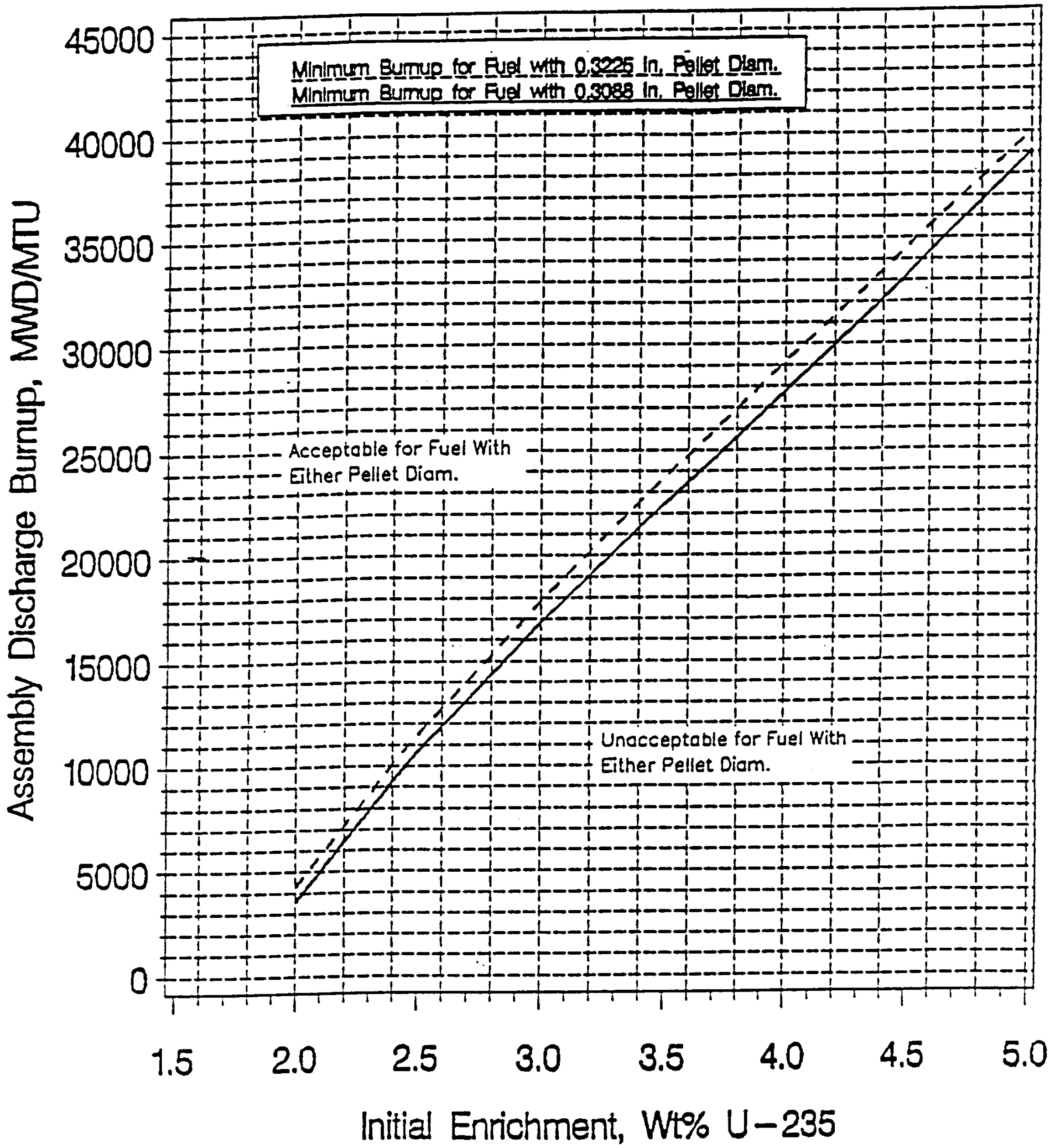

FIGURE 3.7.17-2
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT AND FUEL PELLET DIAMETER
FOR AN ALL CELL STORAGE CONFIGURATION FOR THE PERMANENT SPENT
FUEL POOL STORAGE RACKS

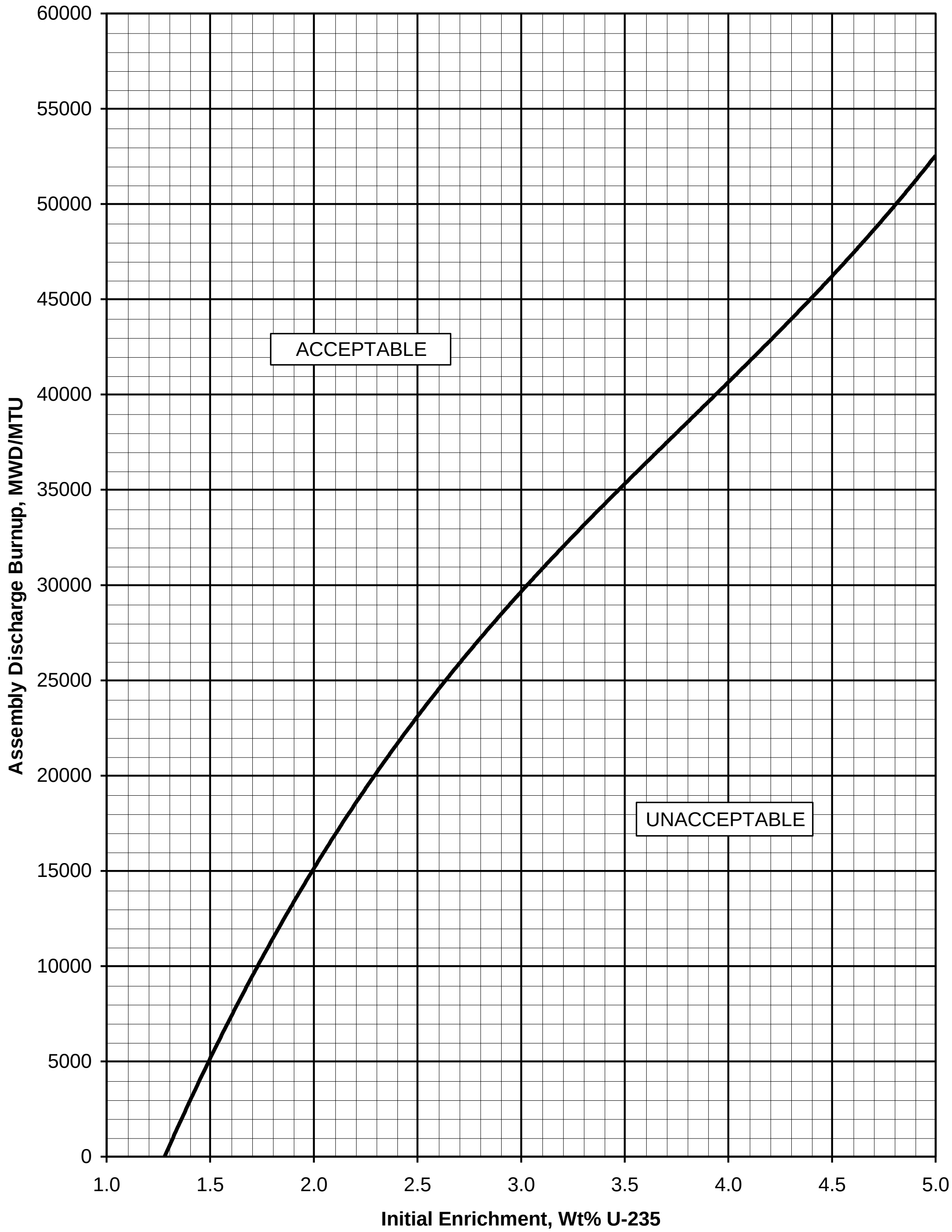

FIGURE 3.7.17-3
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT
FOR A 2X2 ARRAY STORAGE CONFIGURATION FOR THE PERMANENT SPENT
FUEL POOL STORAGE RACKS

4.0 DESIGN FEATURES

4.1 Site Location

The DCPP site consists of approximately 750 acres which are adjacent to the Pacific Ocean in San Luis Obispo County, California, and is approximately twelve (12) miles west-southwest of the city of San Luis Obispo.

4.2 Deleted

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The permanent spent fuel pool storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the DSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 806 ppm, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the DSAR;

d. A nominal 11 inch center to center distance between fuel assemblies placed in the fuel storage racks;

e. Fuel assemblies with a discharge burnup in the "acceptable" region of Figure 3.7.17-2 for the all cell configuration as shown in Figure 3.7.17-1;

f. Fuel assemblies with a discharge burnup in the "acceptable" region of Figure 3.7.17-3 for the 2x2 array configuration as shown in Figure 3.7.17-1.

4.3.2 Drainage

The spent fuel storage pools are designed and shall be maintained to prevent inadvertent draining of the pool below elevation 133 ft.

4.3.3 Capacity

The permanent spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1324 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility when absent.

The plant manager or designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect the safe handling and storage of nuclear fuel.

5.1.2 The Shift Supervisor shall be responsible for the shift command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for facility staff and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safe storage and handling of spent nuclear fuel. The primary role of all nuclear workers is to protect the health and safety of the public.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all facility organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the DSAR;

b. The plant manager shall be responsible for overall safe operation of the facility and shall have control over those onsite activities necessary for safe storage and handling of the nuclear fuel;

c. A specified corporate officer shall have corporate responsibility for the safe storage and handling of nuclear fuel and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the facility to ensure safe storage and handling of nuclear fuel; and

d. The individuals who train the CERTIFIED FUEL HANDLERs, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

5.2.2 Facility Staff

The facility staff organization shall include the following:

a. Each on duty shift shall be composed of at least one Shift Supervisor shared between Units 1 and 2, and one NON-CERTIFIED OPERATOR per unit. The NON-CERTIFIED OPERATOR position may be filled by a CERTIFIED FUEL HANDLER.

b. Except for the Shift Supervisor, shift crew composition may be less than the minimum requirement of 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements and all of the following conditions are met:

(continued)

5.2 Organization

5.2.2 Facility Staff (continued)

(1) No fuel movements are in progress;

(2) No movement of loads over fuel are in progress; and

(3) No unmanned shift positions during shift turnover shall be permitted while the shift crew is less than the minimum.

c. A health physics technician shall be on site during fuel handling operations and during movement of heavy loads over the fuel storage racks. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. Not used.

e. The Shift Supervisor shall be a CERTIFIED FUEL HANDLER.

f. At least one person qualified to stand watch in the control room (NON-CERTIFIED OPERATOR or CERTIFIED FUEL HANDLER) shall be present in the control room when nuclear fuel is stored in a spent fuel pool.

g. Oversight of fuel handling operations shall be provided by a CERTIFIED FUEL HANDLER.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Facility Qualifications

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications referenced for comparable positions as specified in the Quality Assurance Program.

5.3.2 A training and retraining program for CERTIFIED FUEL HANDLERs shall be maintained.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The procedures applicable to the safe storage of spent nuclear fuel, recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. Not Used;

c. Quality assurance for effluent and environmental monitoring;

d. Not used; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

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

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

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

   2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the plant manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

(continued)

5.5 Programs and Manuals

5.5.2 Not Used

5.5.3 Not Used

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

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

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

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

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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

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

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

2. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ.

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

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

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

k. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program Surveillance frequency.

5.5.5 Not Used

5.5.6 Not Used

5.5.7 Not Used

5.5.8 Not Used

5.5.9 Not Used

5.5.10 Not Used

5.5.11 Not Used

(continued)

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in temporary unprotected outdoor liquid storage tanks.

The gaseous radioactivity quantities shall be determined following the methodology in Regulatory Guide 1.24 "Assumptions Used For Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure." The liquid radwaste quantities shall be maintained such that 10 CFR Part 20 limits are met.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in temporary outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations 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, in the event of an uncontrolled release of the tanks' contents.

   The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Not Used

(continued)

5.5 Programs and Manuals

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. a change in the TS incorporated in the license; or

   2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Not Used

5.5.16 Not Used

5.5.17 Not Used

5.5.18 Not Used

5.5.19 Not Used

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Not Used

5.6.2 Annual Radiological Environmental Operating Report

---NOTE---
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the facility during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

(continued)

5.6 Reporting Requirements

5.6.3 Radioactive Effluent Release Report

---NOTE---

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

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

5.6.4 Not Used

5.6.5 Not Used

5.6.6 Not Used

5.6.7 Not Used

5.6.8 Not Used

5.6.9 Not Used

5.6.10 Not Used

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

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

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

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

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

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

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

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

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

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

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

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

(continued)

5.7 High Radiation Area

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

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

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

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

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

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

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

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

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

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

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

(continued)

5.7 High Radiation Area

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

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

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

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

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

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

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

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

Proposed Changes to Technical Specification Bases – Clean
(For Information Only)

(17 Pages)

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 and LCO 3.0.2 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification. The Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore variables to within specified limits. If this Required Action is not completed within the specified Completion Time, action may be required to place the facility in a condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.)

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
SR 3.0.1	SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Variables are assumed to be within limits when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that variables are within limits when the requirements of the Surveillance(s) are known not to be met between required Surveillance performances. Surveillances do not have to be performed when the facility is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

(continued)

BASES

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications.

(continued)

BASES

SR 3.0.2 (continued)	The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.
SR 3.0.3	SR 3.0.3 establishes the flexibility to defer declaring an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. SR 3.0.3 is only acceptable if it is discovered that a Surveillance was not performed after the specified Frequency had already expired. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance. The basis for this delay period includes consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions, or requirements of regulations (e.g., as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

(continued)

BASES

SR 3.0.3
(continued)

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. All missed Surveillances will be placed in the Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

(continued)

BASES

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that variable limits are met before entry into other specified conditions in the Applicability for which these limits ensure facility safety. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring variable limits before entering another specified condition in the Applicability.

The provisions of SR 3.0.4 shall not prevent entry into other specified conditions in the Applicability that are required to comply with ACTIONS.

(continued)

BASES

SR 3.0.4 (continued)

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance.

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the spent fuel pool design is given in the DSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the DSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the DSAR, Sections 15.4 and 15.5 (Ref. 3).
APPLICABLE SAFETY ANALYSES	The design basis fuel handling accident analysis is evaluated in accordance with the guidance in Regulatory Guide 1.183 (Reference 4). The minimum water level in the spent fuel pool ensures the resultant dose consequences remain below the acceptance criteria. The fuel handling accident is assumed to occur during handling of a spent fuel assembly in the SFP. One fuel assembly is damaged, releasing all of the fuel gap activity associated with that assembly. The activity (consisting of noble gases, halogens, and alkali metals) is released in a "puff" to the SFP, which has a minimum of 21 feet of water above the damaged fuel assembly. A halogen decontamination factor of 142 is applied to the gap release (for all halogen species) to account for the scrubbing effect of the 21 feet of water above the damaged fuel assembly. The listed DF is developed using guidance provided in Reference B-1 of RG 1.183 (Reference 4). Noble gas and un-scrubbed iodines rise to the water surface where they are mixed in the minimum available air space in the FHB above the SFP. Per Appendix B of RG 1.183, Revision 0 (Reference 4), all of the alkali metals are retained in the SFP water. In practice, the water level maintained for fuel handling provides more than 23 feet of water over the top of irradiated fuel assemblies seated in the storage racks. DSAR Section 9.1.4.3.6 requires the water level provide a minimum of 8 feet of water shielding during fuel handling. This assures more than 24 feet 6 inches of water shielding over the top of the fuel assemblies in the racks and more than 23 feet of water shielding over a fuel assembly lying horizontally on top of the racks. The spent fuel pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO	The spent fuel pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Reference 5). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool, since the potential for a release of fission products exists.
ACTIONS	<u>A.1</u> When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pool water level is lower than the required level, the movement of irradiated fuel assembly in the spent fuel pool is immediately suspended. This does not preclude movement of a fuel assembly to a safe position.
SURVEILLANCE REQUIREMENTS	<u>SR 3.7.15.1</u> This SR is done during the movement of irradiated fuel assemblies as stated in the Applicability. This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk.
REFERENCES	1. DSAR, Section 9.1.2. 2. DSAR, Section 9.1.3. 3. DSAR, Sections 15.4 and 15.5. 4. Regulatory Guide 1.183, July 2000 5. PG&E Calculation Number DECOM-N-0003, (Vendor Calculation 14078100-C-M-00002), Dose Consequences at the site Boundary, Existing Control Room, Proposed Alternate Control Rooms and the TSC following a Fuel Handling Accident in the Fuel Handling Building. .

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

The DCPP Units 1 and 2 spent fuel pools (Ref. 4) each consist of 16 permanent stainless steel racks of various sizes, with a total of 1,324 fuel assembly storage cells. The spent fuel pool storage racks have been analyzed for the storage of fuel assemblies that meet the requirements of LCO 3.7.17.

10 CFR 50.68(b)(4), requires that the spent fuel storage racks are designed to assure that with credit for soluble boron and with fuel of the maximum fuel assembly reactivity, a K_{eff} of less than or equal to 0.95 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with borated water, and a K_{eff} of less than 1.0 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with unborated water.

Criticality analyses have been performed for the permanent storage racks (Ref. 3 and 5), which demonstrate that the multiplication factor, k_{eff}, of the fuel assemblies in the spent fuel storage racks is less than or equal to 0.95. In order to maintain k_{eff} less than or equal to 0.95, the presence of soluble boron is credited in the spent fuel pool criticality analyses.

For the permanent storage racks, Reference 5 provides the analysis for the 2x2 array and checkerboard configurations, and Reference 3 provides the analysis for the all cell configuration. Both criticality analyses (Ref. 3 and 5) evaluate the region of the spent fuel pool that does not contain any Boraflex panels because the storage requirements are more restrictive and yield more conservative reactivity results than the region containing Boraflex. The results of the analyses may be conservatively applied to the less reactive region.

(continued)

BACKGROUND (continued)

The criticality analyses considered accident conditions (Ref. 3, and 5). Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95 and to mitigate the worst accident reactivity insertion. For the permanent storage racks, a soluble boron concentration of 806 ppm is required to maintain k_{eff} less than or equal to 0.95 for all allowable storage configurations, which is well within the 2000 ppm requirement of LCO 3.7.16.

For such an occurrence, the double contingency principle of ANSI N16.1-1975 and the April 1978 NRC letter (Ref. 1) can be applied. The NRC letter states it is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for such a postulated reactivity insertion accident condition, the presence of soluble boron in the spent fuel pool water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

In addition to consideration of spent fuel pool criticality, a boron dilution analysis (Ref. 6) was performed to evaluate the time and water volumes required to dilute the spent fuel pool from 2000 to 800 ppm.

The results of the boron dilution analysis concluded that an unplanned or inadvertent event that would result in the dilution of the spent fuel pool boron concentration from 2000 ppm to 800 ppm is not a credible event since a dilution event large enough to result in a significant reduction in the spent fuel pool boron concentration would involve the transfer of a large quantity of water from a dilution source and a significant increase in spent fuel pool level, which would ultimately overflow the pool. The overflow of the spent fuel pool would be readily detected and terminated by plant personnel. In addition, because of the large quantities of water required and the low dilution flow rates available, any significant dilution of the spent fuel pool boron concentration would only occur over a long period of time (hours to days).

(continued)

BASES

BACKGROUND (continued)	Detection of a spent fuel pool boron dilution via pool level alarms, visual inspection during normal operator rounds, significant changes in spent fuel pool boron concentration, or significant changes in the unborated water source volume, would be expected before a dilution event sufficient to increase K_{eff} above 0.95 could occur. However, for the permanent storage racks analyses have been performed to demonstrate that even if the spent fuel pool boron concentration was diluted to zero ppm, which would take significantly more water than evaluated in the boron dilution analysis, the spent fuel would be expected to remain subcritical and the health and safety of the public would be assured.
APPLICABLE SAFETY ANALYSES	Most accident conditions result in a negligible reactivity effect in the spent fuel pool (Ref. 3, and 5). However, scenarios can be postulated that could have more than a negligible positive reactivity effect. Examples of such accident scenarios for the permanent storage racks are the misplacement of a fuel assembly, a significant increase in spent fuel pool temperature above 150°F, or a cask drop accident (Ref. 4). A soluble boron concentration of 806 ppm is required to maintain k_{eff} less than or equal to 0.95 under accident conditions, which is well within the 2000 ppm requirement of LCO 3.7.16. The negative reactivity effect of the soluble boron more than compensates for the increased reactivity caused by the postulated accident scenarios. The accident analyses is provided in the DSAR (Ref. 4). The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The spent fuel pool boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in References 3, 4, and 5. The specified boron concentration of 2000 ppm ensures that the spent fuel pool k_{eff} will remain less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level, for a postulated reactivity insertion accident or boron dilution event.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

(continued)

BASES

ACTIONS

A.1 and A.2

When the concentration of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies and immediately taking actions to restore the spent fuel pool boron concentration to greater than or equal to 2000 ppm. This suspension of fuel movement does not preclude movement of fuel assemblies to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by chemical analysis that the concentration of boron in the spent fuel pool is at or above the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
2. Not used.
3. "Criticality Safety Evaluation of Region 2 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment," S.E.Turner, October 1993, Holtec Report HI-931077.
4. DSAR, Sections 9.1, 15.4, and 15.5.
5. "Diablo Canyon Units 1 and 2 Spent Fuel Criticality Analysis," February 14, 2001, Paul F. O'Donnell, Westinghouse Doc. No. A-DP1-FE-0001.
6. "Diablo Canyon Units 1 and 2 Spent Fuel Pool Boron Dilution Source Analysis," M-1188 (SAP Calculation No, 9000041656), December 2016
7. License Amendment 154/154, September 25, 2002.
8. Not used.
9. Not used.

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

The DCPP Units 1 and 2 spent fuel pools (Ref. 2) each consist of 16 stainless steel racks of various sizes, with a total of 1,324 fuel assembly storage cells. The spent fuel pool storage racks have been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.17. Unfilled cells in the permanent storage racks may be utilized for the storage of unirradiated fuel assemblies. The 16 permanent spent fuel storage racks are designed to accommodate three different storage configurations as shown in Figure 3.7.17-1.

10 CFR 50.68(b)(4), requires that the spent fuel storage racks are designed to assure that with credit for soluble boron and with fuel of the maximum fuel assembly reactivity, a K_{eff} of less than or equal to 0.95 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with borated water, and a K_{eff} of less than 1.0 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with unborated water.

Criticality analyses have been performed (Ref. 3, and 4) which demonstrate that the multiplication factor, k_{eff}, of the fuel assemblies in the spent fuel storage racks is less than or equal to 0.95. In order to maintain k_{eff} less than or equal to 0.95, the presence of soluble boron is credited in the spent fuel pool criticality analysis. Reference 3 provides the analysis for the 2x2 array and checkerboard configurations, and Reference 4 provides the analysis for the all cell configuration.

(continued)

BASES

BACKGROUND (continued)	For the 16 permanent storage racks, both criticality analyses (Ref. 3 and 4) evaluate the region of the spent fuel pool that does not contain any Boraflex panels because the storage requirements are more restrictive and yield more conservative reactivity results than the region containing Boraflex. The results of the analyses may be conservatively applied to the less reactive region. A discussion of how soluble boron is credited for the storage of fuel assemblies in the spent fuel pool is contained in the background for TS 3.7.16 Bases. Storage configurations were defined in the criticality analyses (Ref. 3, and 4) to ensure that k_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95 and to mitigate the worst accident reactivity insertion. For the permanent storage racks, a soluble boron concentration of 806 ppm is required to maintain k_{eff} less than or equal to 0.95 for all allowable storage configurations, which is well within the 2000 ppm requirement of LCO 3.7.16. Prior to movement of an assembly, it is necessary to verify that SR 3.7.16.1 is current.
APPLICABLE SAFETY ANALYSES	For the permanent storage racks, the analyzed accidents that could have significant reactivity effects are the misplacement of a fuel assembly, a significant increase in spent fuel pool temperature above 150°F, or a cask drop accident (Ref. 2, 3, and 4). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.16, "Spent Fuel Pool Boron Concentration") ensures that k_{eff} will remain at or below 0.95. The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with LCO 3.7.17, ensure the k_{eff} of the spent fuel storage pool will always remain ≤ 0.95 at a 95 percent probability, 95 percent confidence level, for a postulated reactivity insertion accident or a boron dilution event.
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in the spent fuel pool.

(continued)

BASES

ACTIONS	A.1 When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with LCO 3.7.17, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with LCO 3.7.17 which will return the fuel pool to an analyzed condition.
SURVEILLANCE REQUIREMENTS	SR 3.7.17.1 This SR verifies by administrative means that each fuel assembly and its expected storage location are in accordance with LCO 3.7.17 prior to each fuel assembly move when the assembly is to be stored in the spent fuel pool. A complete record of initial enrichment, initial integral boron content, fuel pellet diameter, and the cumulative burnup analysis shall be maintained for the time period that each fuel assembly remains in the spent fuel pool.
REFERENCES	1. Not used. 2. DSAR, Sections 9.1, 15.4, and 15.5. 3. "Diablo Canyon Units 1 and 2 Spent Fuel Pool Criticality Analysis," February 14, 2001, Paul F. O'Donnell, Westinghouse Doc. No. A-DP1-FE-0001. 4. "Criticality Safety Evaluation of Region 2 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment," S.E.Turner, October 1993, Holtec Report HI-931077. 5. License Amendment 154/154, September 25, 2002. 6. Not used. 7. Not used.