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PG&E Letter DCL-21-040

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Diablo Canyon Units 1 and 2
Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Responses to NRC Requests for Additional Information on LAR 20-03, Proposed Technical Specifications and Revised License Conditions for the Permanently Defueled Condition

Reference:

1. PG&E Letter DCL-20-092, License Amendment Request 20-03, Proposed Technical Specifications and Revised License Conditions for the Permanently Defueled Condition, dated December 3, 2020 (ML20338A546)
2. PG&E Letter DCL-21-019, Supplement to License Amendment Request 20-03, Proposed Technical Specifications and Revised License Conditions for the Permanently Defueled Condition, dated April 1, 2021 (ML21091A069)
3. Email from NRC Project Manager, Samson Lee, Request for additional information – Diablo Canyon proposed technical specifications and revised license conditions for the permanently defueled condition (EPID: L-2020-LLA-0261)

Dear Commissioners and Staff:

In Reference 1, Pacific Gas and Electric Company (PG&E) submitted License Amendment Request (LAR) 20-03, to propose revisions to the technical specifications and license conditions to reflect the permanently defueled condition. In Reference 2, PG&E supplemented LAR 20-03 to request additional changes. In Reference 3, the NRC provided requests for additional information (RAIs) regarding Reference 1. The Enclosure to this letter provides PG&E responses to the RAIs.

PG&E makes no new or revised regulatory commitments (as defined in NEI 99-04) in this letter.

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 May 13, 2021.

Sincerely,

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

Enclosure

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

Responses to NRC Request for additional information – Diablo Canyon proposed technical specifications and revised license conditions for the permanently defueled condition (EPID: L-2020-LLA-0261)

Applicable Regulatory Requirements

The provisions in Title 10 of the Code of Federal Regulations (10 CFR) 50.36(c)(6), "Decommissioning," in part, apply only to nuclear power reactor facilities that have submitted the certifications required by 10 CFR 50.82(a)(1). For such facilities, 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. In the application the licensee states that the proposed changes to the operating licenses and TSs are in accordance with 10 CFR 50.36(c)(1) through (c)(5) and also include administrative changes.

Fuel Handling Accident (FHA) Analysis

Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Rev. 0, July 2000, provides the methodology for analyzing the radiological consequences of several design-basis accidents (DBAs) to show compliance with 10 CFR 50.67. RG 1.183 provides guidance to licensees on acceptable application of alternate source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000, provides review guidance to the staff for the review of AST amendment requests. Section 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in RG 1.183. The dose acceptance criteria for the FHA are a Total Effective Dose Equivalent (TEDE) of 6.3 rem at the exclusion area boundary (EAB) for the worst 2 hours, 6.3 rem at the outer boundary of the low population zone (LPZ), and 5 rem in the control room for the duration of the accident.

In Section 2.1 of the license amendment request (LAR), PG&E states that a new FHA analysis was performed to determine the Control Room (CR), EAB, and LPZ doses at Diablo Canyon.

Staff Request for FHA

The LAR indicates that the analysis demonstrates compliance with 10 CFR 50.67 and conforms to RG 1.183 and that the limits of RG 1.183 continue to be met, however, many of the assumptions and parameters used are not specified and the doses

calculated are not specified. The LAR also specifies that the calculations demonstrate that the EAB dose will be less than 1 rem when projected over four days, as specified in the EPA Protective Action Guides (PAG) Manual. The licensee specifies that the calculations were performed using RADTRAD 3.0.3.

RAI-1

Please provide sufficient technical details of the new FHA to allow for NRC staff evaluation. The NRC staff requests this information to include, but not necessarily limited to: key input variables, calculated dose to CR, EAB, and LPZ, source term use in the new FHA analysis, fall height of fuel assembly, water coverage, decontamination factors, and atmospheric dispersion factors.

PG&E Response to RAI-1:

As noted in License Amendment Request (LAR) 20-03, after the permanent cessation of power operations, spent nuclear fuel will be removed from the reactor vessels and stored in the spent fuel pools (SFPs) located in the Fuel Handling buildings (FHBs) of either unit until the fuel is transferred to the independent spent fuel storage installation (ISFSI). To that end, the revised Fuel Handling Accident (FHA) analysis was performed to determine the dose consequences at the Exclusion Area Boundary (EAB), Control Room (CR), and the Technical Support Center (TSC) following a FHA in the FHB after permanent cessation of operations at Diablo Canyon Power Plant.

This event postulates that a spent fuel assembly is dropped during fuel handling in the SFP. In accordance with current licensing basis documented in License Amendment No. 230 / 232, dated April 27, 2017 (Reference 3), all of the fuel rods in the dropped fuel assembly are assumed to be damaged; thus all of the activity in the fuel gap of the dropped assembly is assumed to be instantaneously released into the SFP. As documented in current licensing basis:

- The assumption that all fuel rods in one assembly ruptures, is conservative, because the kinetic energy available for causing damage to a fuel assembly dropped through water is fixed by the drop distance.
- The kinetic energy associated with the maximum drop height for a FHA is not considered sufficient to rupture the equivalent number of fuel rods of one assembly in both the dropped assembly and the impacted assembly.

As discussed earlier, and in accordance with current licensing basis, the analysis assumes that one (1) fuel assembly is damaged, thus 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 gap activity is released in a “puff” to the fuel pool.

As documented in Reference 3, the core gap fractions are based on Draft Guide (DG)-1199, Table 3. Use of these fractions is acceptable because Diablo Canyon Power Plant (DCPP) operations fall within the maximum allowable power

operating envelop for Pressurized Water Reactors (PWRs) shown in Figure 1 of DG-1199.

In accordance with DG-1199, the core *gap activity* is assumed to be comprised of 8 percent of the core I-131 inventory, 23 percent of the core I-132 inventory, 35 percent of the core Kr-85 inventory, 4 percent of the remaining core noble gas inventory, 5 percent of the remaining core halogen inventory, and 46 percent of the core alkali metal (Cesium and Rubidium) inventory.

In accordance with current licensing basis and per Regulatory Guide (RG) 1.183, the radioiodine released from the fuel gap is assumed to be 95 percent particulate (cesium iodide [CsI]), 4.85 percent elemental, and 0.15 percent organic. Due to the acidic nature of the water in the fuel pool (pH less than 7), the CsI is assumed to immediately disassociate and re-evolve as elemental iodine, thus changing the chemical form of iodine to 99.85 percent elemental and 0.15 percent organic.

In accordance with RG 1.183, as a result of the FHA, the noble gases and non-particulate iodines in the fuel gap of the damaged assembly (all particulates are retained in the water) are discharged to the environment within 2 hours of the accident.

The depth of water above the damaged assembly in the SFP is assumed to be reduced from the current licensing basis value of 23 ft to 21 feet. This value is based on the Plant Technical Specifications (TS) with no credit taken for the additional depth of water provided by operating procedures. Reference B-1 of RG 1.183 Appendix B, was used to develop the pool decontamination factor when the pool depth above the dropped fuel assembly is less than 23 ft. The 21 ft water depth results in a conservatively estimated overall iodine pool decontamination factor (DF) of 142, and a resulting iodine chemical species mix of 70 percent elemental and 30 percent organic above the SFP.

For purposes of conservative analysis, the updated calculation does not credit the CR or TSC ventilation systems (e.g., filtration, flowrates, intake location if beneficial), the CR or TSC structure (e.g., reduced source volume, associated shielding), or the FHB ventilation system (which exhausts through the plant vent and provides beneficial atmospheric dispersion factors). The above conservative assumptions are intended to support minimizing the number of systems, structures, and operating procedures credited to demonstrate habitability, and support declassification of safety-related equipment.

Even though DCPP is not requesting an exemption from Emergency Planning Requirements at this time, the acceptance criteria for the dose at the EAB used for the FHA reanalysis is conservatively based on the Environmental Protection Agency (EPA) Protective Action Guides (PAGs). The current licensing basis acceptance criteria cites Section 4.4, Table 6 of RG 1.183, and for the FHA, is 6.3 rem Total Effective Dose Equivalent (TEDE). The intent of using the more limiting acceptance criteria from the EPA PAGs is to enable DCPP to support a future emergency planning exemption without having to perform a reanalysis.

Section 2.1.1 of the 2017 EPA PAG Manual indicates that for purposes of comparison with the PAGs, the projected dose needs to consider: (a) the direct shine exposure to the passing plume (cloudshine or submersion), (b) the inhalation dose as a result of submersion in the passing plume, and (c) the dose due to deposition of radioiodine / particulates from the plume (groundshine). In accordance with the EPA PAG manual, exposure pathways that contribute less than 10 percent to the total dose incurred need not be considered.

In accordance with the scenario outlined above, ground deposition is limited to radioiodines (which do not have very long-half-lives); thus the dose contribution due to iodine, and consequently, ground deposition, becomes increasingly insignificant as the time after reactor shutdown increases. For purposes of this analysis, the dose contribution due to ground shine was demonstrated to be less than 10 percent of the total dose incurred and was therefore neglected.

Based on the above, the acceptance criteria of the FHA reanalysis are as follows:

- The acceptance criteria for the EAB is based on the EPA PAG guidelines for initiating sheltering-in-place or evacuation of the public during the "Early Phase" of an accident, i.e., an integrated dose that is less than 1 rem when projected over 4 days. This is more limiting than the currently estimated licensing basis dose for the operating plant of 1 rem TEDE (documented in Reference 3) to an individual at any point on the boundary of the exclusion area for any 2-hour period following the onset of the accident.

- The acceptance criteria for the LPZ is less than or equal to the currently estimated licensing basis dose (documented in Reference 3) for the operating plant of 0.1 rem TEDE.

The acceptance criteria for habitability of the CR and TSC is less than or equal to the currently estimated licensing basis dose (documented in Reference 3) for the operating plant of 1 rem TEDE and 4.1 rem TEDE, respectively.

As part of this analysis, three decay periods (i.e., 30 days, 60 days, and 90 days after permanent reactor shutdown) were evaluated. The analysis concludes that a decay time of 45 days is required to meet the above acceptance criteria, the most limiting criteria being that associated with the control room.

Computer Programs Used to Calculate Dose

The computer codes utilized to support this analysis are part of current DCPP licensing basis and are listed below. These computer programs have been verified and validated under the Stone & Webster, LLC, Quality Assurance Program and the associated Quality Assurance Manual (QAM) that governs Stone & Webster activities in engineering, design, procurement, construction, modification repair, and

decommissioning of nuclear facilities. The referenced QAM was prepared in accordance with the requirements of Title 10, Part 50, Appendix B of the Code of Federal Regulations.

- NRC sponsored program RADTRAD 3.03 was used to calculate the airborne dose to the operator in the CR and to a member of the public located at the EAB following a FHA in the FHB.
- Industry computer program SCALE4.3-ORIGEN-S was used to determine the isotopic activity in the fuel gap at the various decay times analyzed.
- Stone & Webster (formerly WECTEC) computer program PERC2 was used to estimate the dose due to ground shine from deposited iodine following an FHA.

Key Input Variables

Table 1 - Key Input Variables for the FHA in the FHB	
Parameter	**Value Assigned**
Reactor core thermal power	3580 Mega Watt thermal (MWt) (includes margin for power uncertainty)
Number of damaged fuel assemblies	1 (264 fuel rods/assembly)
Fraction of Core Activity in the Fuel gap	Noble gases (0.04) (except Kr-85) Halogens (0.05) (except I-131, and I-132) Kr-85 (0.35) I-131 (0.08) I-132 (0.23) Alkali Metals (0.46)
Gap activity per assembly as a function of decay time	See Table 2 below
Radial Peaking Factor	1.65
Water Depth in Spent Fuel Pool	21 feet (Amount of water coverage over a damaged fuel assembly assumed to be lying horizontally on top of the storage racks, and is based on the Plant Technical Specifications with no credit taken for the additional depth of water provided by operating procedures.)
Pool Water Decontamination Factors (DF) (based on 21 ft of water coverage over the damaged fuel assembly)	Halogens (142) (overall DF) Noble gas (1) Particulates (∞)
Chemical Form of Iodine Released Before Scrubbing	99.85 percent elemental 0.15 percent organic
Exclusion Area Boundary X/Q	2.50E-04 seconds per cubic meter (sec/m^3)
Offsite Breathing Rate	0-8 hr; 3.5E-04 m^3/sec 8-24 hr; 1.8E-04 m^3/sec 1-30 day; 2.3E-04 m^3/sec

Table 1 - Key Input Variables for the FHA in the FHB	
Parameter	**Value Assigned**
CR Breathing Rate	0-30 day; 3.5E-04 m^3/sec
CR Occupancy Factors	0-1 day; 1.0 1-4 day; 0.6 4-30 day; 0.4
Minimum CR Envelope Free Volume	170,000 cubic feet (ft^3)
Maximum Unfiltered CR Normal Operation Ventilation Intake Flow Rate	Unit 1 2100 cubic feet per minute (cfm) ± 10 percent Unit 2 2100 cfm ± 10 percent
Total CR Unfiltered In-leakage During Normal Ventilation Mode and During Emergency Mode	70 cfm
CR Related Dispersion Factors (0-2 hr)	Unit 1 Plant Vent to Unit 1 CR Normal Intake 1.67E-03 sec/m^3 Unit 1 Plant Vent to Unit 2 CR Normal Intake 9.08E-04 sec/m^3 Unit 1 Plant Vent to CR Centerline 1.25E-03 sec/m^3 Unit 2 Plant Vent to Unit 1 CR Normal Intake 7.79E-04 sec/m^3 Unit 2 Plant Vent to Unit 2 CR Normal Intake 1.49E-03 sec/m^3 Unit 2 Plant Vent to CR Centerline 1.11E-03 sec/m^3 Unit 1 FHB to Unit 1 CR Normal Intake 6.68E-03 sec/m^3 Unit 1 FHB to Unit 2 CR Normal Intake 2.69E-03 sec/m^3 Unit 1 FHB to CR Centerline 3.54E-03 sec/m^3 Unit 2 FHB to Unit 1 CR Normal Intake 2.68E-03 sec/m^3 Unit 2 FHB to Unit 2 CR Normal Intake 6.68E-03 sec/m^3 Unit 2 FHB to CR Centerline 3.61E-03 sec/m^3 Based on the above, the most limiting value is used for purposes of atmospheric dispersion to the CR following a FHA at either unit, i.e., <u>6.68E-03 sec/m^3</u>
Minimum Technical Support Center (TSC) Free Volume	51,250 ft^3
Maximum Unfiltered TSC Normal Operation Ventilation Intake Flow Rate	500 cfm
Total TSC Unfiltered In-Leakage	60 cfm
TSC Related Dispersion Factors (0-2 hr)	Unit 1 Plant Vent to TSC Normal Intake 3.04E-04 sec/m^3 Unit 1 Plant Vent to TSC Centerline 3.41E-04 sec/m^3 Unit 2 Plant Vent to TSC Normal Intake 5.47E-04 sec/m^3 Unit 2 Plant Vent to TSC Centerline 5.41E-04 sec/m^3

Table 1 - Key Input Variables for the FHA in the FHB	
Parameter	**Value Assigned**
	Unit 1 FHB to TSC Normal Intake 3.77E-04 sec/m^3 Unit 1 FHB to TSC Centerline 4.21E-04 sec/m^3 Unit 2 FHB to TSC Normal Intake 4.88E-04 sec/m^3 Unit 2 FHB to TSC Centerline 5.26E-04 sec/m^3 Based on the above, the most limiting value is used for purposes of atmospheric dispersion to the TSC following a FHA at either unit, i.e., 5.47E-04 sec/m^3

Table 2 – Gap activity per Assembly (Curies) versus Decay times after Reactor Shutdown			
Nuclide	**T=30 days**	**T=60 days**	**T=90 days**
I129	1.04E-03	1.04E-03	1.04E-03
I130	2.72E-15	7.95E-33	0.00E+00
I131	3.22E+03	2.42E+02	1.82E+01
I132	2.93E+02	4.96E-01	8.38E-04
I133	2.03E-06	7.72E-17	2.93E-27
I135	5.05E-29	0.00E+00	0.00E+00
KR 83M	0.00E+00	0.00E+00	0.00E+00
KR 85	1.99E+03	1.99E+03	1.98E+03
KR 85M	0.00E+00	0.00E+00	0.00E+00
KR 88	0.00E+00	0.00E+00	0.00E+00
XE127	2.30E-03	1.30E-03	7.36E-04
XE129M	2.57E-01	2.49E-02	2.40E-03
XE131M	9.89E+01	2.07E+01	3.90E+00
XE133	9.72E+02	1.84E+01	3.48E-01
XE133M	1.61E-01	1.21E-05	9.08E-10
XE135	2.16E-19	0.00E+00	0.00E+00
XE135M	6.61E-30	0.00E+00	0.00E+00
CS132	5.53E-01	2.24E-02	9.03E-04
CS134	5.58E+04	5.43E+04	5.29E+04
CS134M	0.00E+00	0.00E+00	0.00E+00
CS135	1.38E-01	1.38E-01	1.38E-01
CS136	3.43E+03	7.08E+02	1.46E+02
CS137	3.27E+04	3.24E+04	3.24E+04
BA137M	3.07E+04	3.07E+04	3.07E+04

Table 2 – Gap activity per Assembly (Curies) versus Decay times after Reactor Shutdown			
Nuclide	**T=30 days**	**T=60 days**	**T=90 days**
RB 86	1.95E+02	6.39E+01	2.10E+01
RB 87	3.38E-10	3.38E-10	3.38E-10
RB 88	0.00E+00	0.00E+00	0.00E+00

Radiological Dose Consequences

1. Site Boundary

The dose to an individual located at the EAB over a 4-day period, following an FHA in the FHB (assumed to occur at 30, 60, and 90 days after permanent reactor shutdown) is presented in the table below.

Table 3 - FHA in the FHB 4-Day Dose at the Exclusion Area Boundary (rem, TEDE)*	
Time After Shutdown	Dose
30 days	0.11
60 days	0.009
90 days	0.001

* The dose consequences reported are based on the estimated inhalation and submersion dose at the EAB. The contribution due to ground shine is less than 10 percent of the total dose incurred, and therefore, per Section 2.1.1 of the EPA PAG Manual, need not be considered.

As shown in Table 3, the estimated 4-day integrated dose of 0.11 rem TEDE based on fuel movement at t=30 days is well below the 1 rem threshold for the PAGS, as well as the currently estimated licensing basis worst case 2-hour integrated dose of 1 rem TEDE applicable to the operating plant.

Based on the current licensing basis 2-hour EAB dose of 1 rem TEDE, and the corresponding 30-day LPZ dose of 0.1 rem TEDE following a FHA in the FHB, the dose at the LPZ as part of this reanalysis will be significantly less than the EAB. Note that the LPZ dose has not been explicitly calculated for this application. Engineering judgment indicates that based on fuel movement at t=30 days and a corresponding EAB dose of 0.11 rem TEDE, the 30-day integrated dose at the LPZ will be well below the currently estimated licensing basis 30-day integrated dose of 0.1 rem TEDE for the operating plant.

2. Control Room and TSC

The 30-day integrated dose in the CR and in the TSC following an FHA in the FHB (evaluated at 30, 60, and 90 days after permanent reactor shutdown) are presented in Table 4 below.

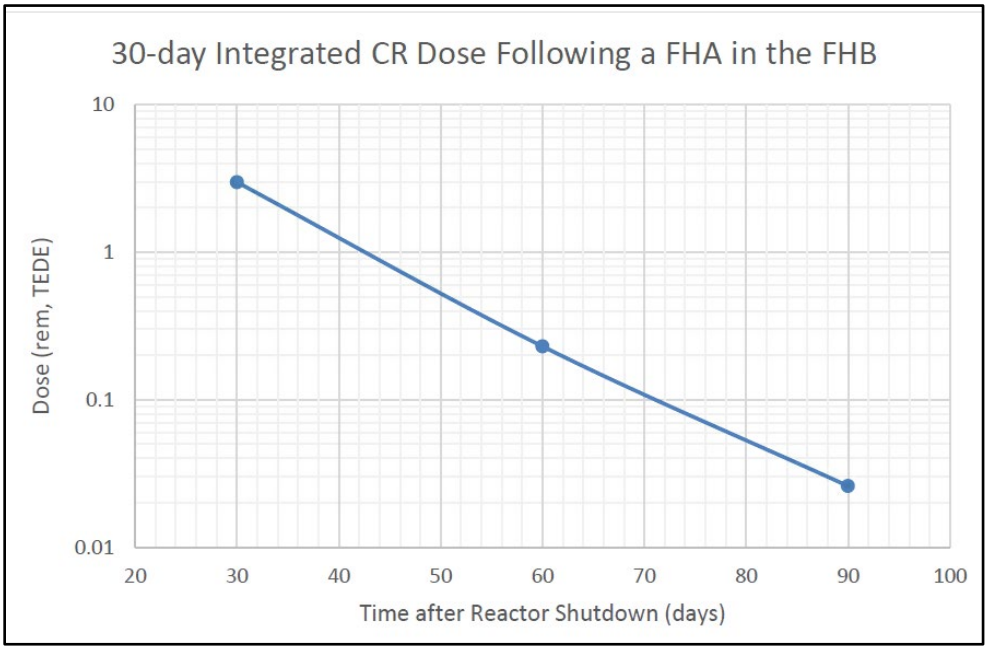
Table 4. Thirty Day Integrated Dose Following FHA in the FHB (rem, TEDE)			
Location	Time After Shutdown		
	30 days	60 days	90 days
Control Room	2.98	0.23	0.026
TSC	0.244	0.019	0.0022

Notes
1. Doses are conservatively calculated assuming the individual is located on the building roof (outside) for 30 days (with occupancy factors). This means that there is no dose dependence on the CR/TSC ventilation rates or structure.

As demonstrated in Table 4, the calculated dose to a hypothetical individual located in the CR or TSC are well below the acceptance criteria of less than or equal to 5 rem TEDE, 30 days after shutdown. Also, based on fuel movement at t=30 days, the dose in the TSC is less than the currently estimated licensing basis dose for the operating plant of 4.1 rem TEDE.

As demonstrated below in Figure 1, the time after permanent reactor shutdown when the CR dose is equal to the CR dose of 1.0 rem TEDE reported in the current design basis analysis of record for the FHA in the FHB is a little over 42 days. For purposes of conservatism, a minimum of “45 days after reactor shutdown” is used to stipulate when credit for operation of the CR ventilation system or FHB ventilation system is not required during fuel movement for “post-permanent cessation of power operations”, since by that time, the resultant dose in the CR would be below the current licensing basis value of 1 rem TEDE.

Figure 1



In summary, it was determined that a FHA assumed to occur 45 days after reactor shutdown, with conservative assumptions intended to minimize the number of systems, structures, and operating procedures credited for accident mitigation, would result in estimated doses that remain within the current licensing basis values for the FHA, specifically:

- A worst case 2-hour integrated dose to an individual at any point on the boundary of the EAB that is less than the currently estimated licensing basis dose for the operating plant of 1 rem TEDE.
- A 30-day integrated dose to an individual at any point on the outer boundary of the LPZ that is less than the currently estimated licensing basis dose for the operating plant of 0.1 rem TEDE.
- A 30-day integrated dose to an operator in the CR that is less than the currently estimated licensing basis dose for the operating plant of 1 rem TEDE.
- A 30-day integrated dose to an operator in the TSC that is less than the currently estimated licensing basis dose for the operating plant of 4.1 rem TEDE.

In addition, it was determined that the 4-day integrated dose at the EAB is less than the 1 rem EPA PAGS for initiating sheltering-in-place or evacuation of the public during the "Early Phase" of an accident.

As described in Reference 2, PG&E has proposed new license conditions for DCPP Units 1 and 2. The new proposed license conditions will restrict handling of spent nuclear fuel and heavy loads over the spent fuel pool following implementation of the Permanently Defueled Technical Specifications (PDTS) until a minimum of 45-days following permanent shutdown needed to meet the current licensing basis dose estimates for the FHA. If approved, these new license conditions will allow PG&E to implement the PDTS prior to the 45-day decay time assumed in the post shutdown FHA analysis. As described in Reference 2, the PDTS will 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;
- and a Certified Fuel Handler Training and Retraining Program has been implemented in accordance with 10 CFR 50.2.

Quality Assurance (QA) and Human Factors

The staff also reviewed the December 5, 2019, letter (ADAMS Accession No. ML19339F388) in which PG&E requested Amendment Nos. 237 and 239. The amendments requested to relocate TS 5.3, "Unit Staff Qualifications," to Chapter 17, "Quality Assurance," of the Updated Final Safety Analysis Report (UFSAR). PG&E stated that moving the personnel qualification requirements (i.e., TS 5.3.1 a, b, c, and 5.3.2) from the Diablo Canyon TS to the UFSAR will provide flexibility in adopting updated NRC endorsed standards and will eliminate the need for future license amendments for each specific position. This is consistent with the guidance contained

in the NRC's Administrative Letter (AL) 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance." The staff concluded that the change was consistent with the guidance in AL 95-06 for relocating a licensee's TS administrative controls to a licensee-controlled document and that there is reasonable assurance that the requirements of 10 CFR 50.36 and 10 CFR 50.120 will continue to be met. Therefore, the NRC staff found the proposed change acceptable as documented in safety evaluation (SE) dated September 11, 2020 (ADAMS Accession No. ML20218A276).

The regulations in 10 CFR 50.4(b)(7)(ii) requires that a change to the Safety Analysis Report QA program description under 10 CFR 50.54(a)(3) or 50.55(f)(3), or a change to a licensee's NRC-accepted QA topical report under 10 CFR 50.54(a)(3) or 50.55(f)(3), must be submitted to the NRC's Document Control Desk.

The regulations in 10 CFR 50.34, "Content of applications; technical information," require that every applicant for an operating license include information in its FSAR on the managerial and administrative controls to be used to ensure safe operation. The information on the controls shall also include a discussion on how the applicable requirements of Appendix B will be satisfied.

The regulations in 10 CFR 50.54 require each power plant subject to the requirements of Appendix B to implement a QA program and 10 CFR 50.54(a)(4) require licensees to submit to the NRC, changes to their QA Program that reduce commitments.

Staff Request for QA and Human Factors

Based on PG&E's request and the NRC's approval of Amendment Nos. 237 and 239, the staff is seeking clarification for the following that affects the current proposed amendment to revise the Operating Licenses:

RAI-2

Clarify why TS 5.3 was not moved to the Quality Assurance Program Description (QAPD) in its entirety consistent with AL 95-06. Changing the pointer from the UFSAR to the QAPD does not change the fact that a LAR would be needed if there was a change to TS 5.3.

PG&E Response to RAI-2:

PG&E is currently planning to remove the Quality Assurance Program (QAP) from the Updated Final Safety Analysis Report (UFSAR) to establish a separate program document prior to the transition to decommissioning. The UFSAR Chapter 17 will continue to provide high level guidance regarding Quality Assurance upon completing this transition. The changes to TS 5.3.1 proposed in Reference 1, were only to support needing to maintain the minimum staff qualifications in the QAP regardless of where the program is located (UFSAR or separate program document). PG&E currently plans to

leave a reference to the QAP in the TS until all spent nuclear fuel is transferred to the ISFSI. At that time and consistent with industry precedent, PG&E plans to submit a LAR to propose the removal of the administrative controls from the TS consistent with Administrative Letter 95-06.

RAI-3

Following the basis of Amendment Nos. 237 and 239, as approved by the NRC staff, clarify why the requirements of proposed TS 5.3.2 are not located in Chapter 17, "Quality Assurance," of the UFSAR.

PG&E Response to RAI-3:

As described within the response to RAI-2, PG&E plans to transition the QAP from Chapter 17, "Quality Assurance" of the UFSAR to a separate license basis document. The staff qualification requirements will continue to reside within this chapter of the UFSAR until implementation of the PDTS. As part of the implementation of the License Amendment, those qualification requirements would transition to the QAP. As currently described in the UFSAR Chapter 17, nuclear generation personnel are required to meet or exceed the minimum qualifications specified in ANSI/ANS 3.1, 1978 for comparable positions, with the following defined exceptions:

(a) The radiation protection manager shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 2, April 1987, for radiation protection manager.

(b) The operations manager shall meet or exceed the minimum qualifications as specified in Technical Specification 5.2.2.e.

(c) The licensed Reactor Operators (ROs) and Senior Reactor Operators (SROs) shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1993 as endorsed by Regulatory Guide 1.8, Revision 3, May 2000 with the exceptions clarified in the current revision to the Operator Licensing Examination Standards for Power Reactors, NUREG-1021, Section ES-202.

(d) For the purpose of 10 CFR 55.4, a licensed SRO and a licensed RO are those individuals who, in addition to meeting the requirements specified in (c), perform the functions described in 10 CFR 50.54(m).

Some modifications to these exceptions will be required to reflect the transition from Licensed SROs and ROs to Certified Fuel Handlers (CFHs). The CFHs described in proposed TS 5.3.2 will be required to meet or exceed the minimum applicable qualifications currently required for licensed SROs and ROs.

RAI-4

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 Diablo Canyon, 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.

Proposed TS 5.3.2., states "A training and retraining program for Certified Fuel Handlers shall be maintained." Clarify what these measures are that will establish the certification program for the Certified Fuel Handlers.

PG&E Response to RAI-4:

As described in the response to RAI-3, Chapter 17 of the UFSAR currently requires all nuclear generation personnel to meet or exceed the minimum qualifications specified in ANSI/ANS 3.1-1978, with exceptions noted. These minimum qualification requirements will remain applicable and the CFHs will be required to meet or exceed the minimum applicable qualifications currently required for licensed SROs and ROs. As specified in Reference 1, PG&E will not implement the PDTS until a CFH Training and Retraining Program has been implemented in accordance with 10 CFR 50.2. The training program will implement the requirements of 10 CFR 50.120, such that it will apply a Systems Approach to Training (SAT) and will provide training and qualification for a CFH. PG&E is tracking industry activities related to proposed rulemaking for, "Regulatory Improvements for Production and Utilization Facilities Transitioning to Decommissioning," and will submit the CFH Training and Retraining Program for NRC approval if required.

RAI-5

Requirements in 10 CFR 50.54(a)(4) would be applicable to the QAPD developed for the decommissioning phase as the existing QAPD in Chapter 17 of the UFSAR is not the same as any other license facility that has undergone decommissioning. Clarify if PG&E is planning to submit a request for staff evaluation of the QA program that would be implemented during the decommissioning phase of Diablo Canyon, Units 1 and 2.

PG&E Response to RAI-5:

PG&E is currently planning to move the QAP out of the UFSAR to a separate program document prior to the transition to decommissioning. The separate program document will be controlled in accordance with 10 CFR 50.54. Consistent with recent industry precedent for plants transitioning to decommissioning, PG&E currently plans to make minimal changes to the QAP to support the transition to decommissioning. All changes to the QAP will be evaluated against the previously accepted QAP and reductions in commitments will be submitted for NRC review and approval in accordance with the

requirements of 10 CFR 50.54(a). At this time, PG&E does not believe the changes will require NRC review and approval prior to implementation.

Final Safety Analysis Report (FSAR)

The FSAR is the principal document upon which the NRC bases its safety evaluation supporting the issuance of an operating license for a nuclear power plant. The UFSAR incorporates changes made to the FSAR in accordance with 10 CFR 50.71(e). The UFSAR serves as a major source of information on the current plant design and supporting analyses.

NRC decommissioning guidance (e.g., RG 1.184, "Decommissioning of Nuclear Power Reactors") discusses that the FSAR, which provides a licensing basis for the evaluation of licensing activities under 10 CFR 50.59, will have to be updated to cover decommissioning activities.

The Diablo Canyon LAR Attachment 1 reflects revisions to change the term "Final Safety Analysis Report" to "Defueled Safety Analysis Report" in License Conditions 2.A and 2.B.(2) and to add the term "Defueled Safety Analysis Report" to new License Condition 2.C.(13). LAR Attachment 3, "Proposed Technical Specification Changes – Markup," reflects in the Administrative Controls Section under TS 5.2, "Organization," a revision in TS 5.2.1.a to change the reference from "FSAR" to "DSAR." In TS 4.3.1.1.b and TS 4.3.1.1.c, the references to "FSAR" have been revised to "DSAR."

Staff Request for FSAR

RAI-6a

Given that NRC regulations, such as 10 CFR 50.59, are written in terms of FSAR, and DSAR is not a term that is described, defined, or required in NRC regulations; please explain how the DSAR (replaces FSAR in TSs 4.3.1.1.b, 4.3.1.1.c, and 5.2.1.a above) will remain subject to the provisions of 10 CFR 50.59.

PG&E Response to RAI-6a:

The Defueled Safety Analysis Report (DSAR) is the UFSAR or updated FSAR, retitled to reflect a permanently shutdown and defueled facility. The same regulations and controls that currently apply to the existing UFSAR (e.g., 10 CFR 50.59 and 10 CFR 50.71(e)) will continue to apply to the DSAR. In addition, in response to the editorial comments below, TS 5.5.14 is being updated to reflect that the DSAR is subject to 10 CFR 50.59. Revising the title to the DSAR was administrative and there was no underlying intention of relaxing or modifying any applicable requirements.

RAI-6b

Additionally, 10 CFR 50.71(e)(6) states, "The updated FSAR [UFSAR] shall be maintained by the licensee until the Commission terminates their license." Given NRC requirements for licensees to maintain the updated FSAR until the Commission terminates their license and that a DSAR is not described, defined, or required in NRC regulations; please explain how the Diablo Canyon TSs (DSAR replaces FSAR in TSs 4.3.1.1.b, 4.3.1.1.c, and 5.2.1.a above) will be maintained consistent with the updated FSAR under this proposed title change.

PG&E Response to RAI-6b:

As stated in the response to RAI-6a above, the same regulations and controls that currently apply to the existing UFSAR will continue to apply to the DSAR. Revising the title to the DSAR was administrative and there was no underlying intention of relaxing or modifying any applicable requirements.

Editorial Comments

There appears to be an editorial error in Attachment 3 of the LAR (pg. 251/381). There is a reference to the use of the acronym "DSAR" for the first time in the TS without spelling it out or providing a description/definition. If DSAR is retained (could be changed based on a response to other RAI questions), please spell out the acronym on its first use in the TS.

PG&E Response:

Attachment 1 provides a revised marked up and clean version of TS 4.3.1.1 with the acronym "DSAR" spelled out. Attachment 1, page 1 replaces Enclosure 1, Attachment 3, Page 4.0-1 in Reference 1 in its entirety. Attachment 1, page 2, replaces Enclosure 1, Attachment 6, Page 4.0-1 in Reference 1 in its entirety. Refer to the response below for additional details related to the proposed revisions to TS 4.3.1.1.

There appears to be an editorial error in Enclosure 1 of the LAR in the basis discussion for proposed changes to TS 4.3.1.1 (pg. 141/381). The basis states that the TSs are revised to reflect the conversion of the Final Safety Analysis Report" to the "Defueled Safety Analysis Report" upon implementation of this LAR. In Attachment 3 of the LAR (pg. 251/381), TS 4.3.1.1 is revised to change "FSAR" to "DSAR." There should be no difference between the changes proposed in LAR Enclosure 1 and those reflected in LAR Attachment 3. Please identify the desired proposed change and update the LAR as appropriate.

PG&E Response:

Below is an updated description and basis for the proposed changes to TS 4.3.1.1. Proposed revisions are shown with additions in *underlined italics* and deletions using ~~strikethrough~~.

Current DCPP TS	Proposed DCPP TS
TS 4.3 Fuel Storage 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} ≤ 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;	TS 4.3 Fuel Storage 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 *Defueled Safety Analysis Report (DSAR)*~~FSAR~~; c. k_{eff} ≤ 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;
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 FSAR to the Defueled Safety Analysis Report (DSAR) or DSAR upon implementation of this LAR. Revising the title is administrative and the proposed terminology is consistent with a decommissioning plant.	

As described in the response to the first editorial comment above, Attachment 1 provides the revised marked up and clean version of TS 4.3.1.1.

The staff notes the TS Bases Control Program in TS 5.5.14 did not have any proposed changes. If changes are being made to change FSAR to DSAR in the other sections of the TS, those changes should also be reflected in TS 5.5.14 and justification provided according to the RAI above.

PG&E Response:

Below is a description and basis for change to TS 5.5.14 to reflect the change in terminology from FSAR to DSAR. Proposed revisions are shown with additions in *underlined italics* and deletions using ~~strikethrough~~.

Current DCPP TS	Proposed DCPP TS
TS 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).	TS 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~~ *D*~~F~~SAR 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 *D*~~F~~SAR. 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).
Basis	
TS 5.5.14 – TS Section 5.5.14.b and 5.5.14.c are revised to reflect the conversion of the updated FSAR or FSAR to the DSAR upon implementation of this LAR. The same regulations and controls that currently apply to the existing updated FSAR will continue to apply to the DSAR. Therefore, the TS Bases will be maintained consistent with the DSAR as it was maintained consistent with the updated FSAR.	

Revising the title is administrative and the proposed terminology is consistent with a decommissioning plant.

Attachment 2 provides a marked up and clean version of TS 5.5.14 reflecting the changes described above. Attachment 2, page 1 replaces Enclosure 1, Attachment 3, marked up Page 5.0-15 in Reference 1 in its entirety. Attachment 2, page 2, replaces Enclosure 1, Attachment 6, Page 5.0-10 in Reference 1 in its entirety.

Additional Editorial Change:

PG&E identified that the markup of the title of TS 5.3 included in Attachment 3, of Reference 1 did not match the corresponding change requested within Enclosure 1 of Reference 1. In addition, the header of the TS page for TS 5.3 was not revised in Reference 1, to match the title change requested for TS 5.3. Enclosure 1 of Reference 1, requested the title for TS 5.3 be changed from "Unit Staff Qualifications" to "Facility Staff Qualifications." PG&E is providing updated marked up and clean pages for the changes to TS 5.3 and the Table of Contents for the TS. Attachment 3, page 1 replaces Enclosure 1, Attachment 3, marked up Page 5.0-4 in Reference 1 in its entirety. Attachment 3, page 2 replaces Enclosure 1, Attachment 6, Page 5.0-4 in Reference 1 in its entirety. Attachment 3, page 3 replaces Enclosure 1, Attachment 3, fourth page of the markups of the Table of Contents in Reference 1 in its entirety. Attachment 3, page 4 replaces Enclosure 1, Attachment 6, Table of Contents in Reference 1 in its entirety.

References

1. PG&E Letter DCL-20-092, License Amendment Request 20-03, Proposed Technical Specifications and Revised License Conditions for the Permanently Defueled Condition, dated December 3, 2020 (ML20338A546)

2. PG&E Letter DCL-21-019, Supplement to License Amendment Request 20-03, Proposed Technical Specifications and Revised License Conditions for the Permanently Defueled Condition, dated April 1, 2021 (ML21091A069)

3. 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)

Enclosure
Attachment 1
PG&E Letter DCL-21-040

Revised Marked up (page 1) and Clean Version (page 2) of TS 4.3.1.1

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 Defueled Safety Analysis Report (DSAR) ~~FSAR~~;

c. k_{eff} ≤ 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

4.1 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 Defueled Safety Analysis Report (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.

Marked up (page 1) and Clean Version (page 2) of TS 5.5.14

~~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~~ DF~~F~~SAR 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 DF~~F~~SAR.

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

5.5.14 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 DSAR 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 DSAR.

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

Enclosure
Attachment 3
PG&E Letter DCL-21-040

Revised Marked up (page 1) and Clean Version (page 2) of TS 5.3

Revised Marked up (page 3) and Clean Version (page 4) of TS Table of Contents

5.0 ADMINISTRATIVE CONTROLS

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.

5.0 ADMINISTRATIVE CONTROLS

5.3 Facility Staff 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.