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PG&E Letter HIL-05-003

U.S. Nuclear Regulatory Commission
Director, Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555-0001

ATTN: Document Control Desk

Docket No. 72-27
Humboldt Bay Independent Spent Fuel Storage Installation
Response to NRC Request for Additional Information for the Humboldt Bay Independent Spent Fuel Storage Installation Application

Dear Commissioners and Staff:

On December 15, 2003, Pacific Gas and Electric Company (PG&E) submitted an application to the Nuclear Regulatory Commission (NRC), in PG&E Letter HIL-03-001, requesting a site-specific license for an Independent Spent Fuel Storage Installation (ISFSI) at the Humboldt Bay Power Plant to store the Unit 3 spent nuclear fuel. The application included a Safety Analysis Report, Environmental Report, and other required documents in accordance with 10 CFR 72.

By letter dated July 22, 2004, the NRC staff requested additional information needed to continue their review of the Humboldt Bay ISFSI License Application. PG&E responded to the NRC request for additional information (RAI) in PG&E Letter HIL-04-007, dated October 1, 2004. As part of the NRC review of the responses, additional NRC staff questions have been raised concerning structural issues, thermal issues, and greater than class C waste. Enclosed is PG&E's response to the supplemental RAI.

Holtec International proprietary documents referenced in the Enclosure will be submitted to the NRC at a later date.

NMSS01

If you have any questions regarding this response, please contact Mr. Terence Grebel at (805) 545-4160.

Sincerely,

Donna Jacobs
Vice President – Nuclear Services

emb/3522
Enclosure
cc: PG Fossil Gen HBPP Humboldt Distribution
cc/enc: James R. Hall

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY Humboldt Bay Independent Spent Fuel Storage Installation	Docket No. 72-27

AFFIDAVIT

Donna Jacobs, of lawful age, first being duly sworn upon oath says that she is Vice President, Nuclear Services of Pacific Gas and Electric Company; that she is familiar with the content thereof; that she has executed this supplemental response to NRC requests for additional information regarding the Humboldt Bay Independent Spent Fuel Storage Installation license application on behalf of said company with full power and authority to do so; and that the facts stated therein are true and correct to the best of her knowledge, information, and belief.

Donna Jacobs
Vice President - Nuclear Services

Subscribed and sworn to before me this 8th day of April 2005.

Notary Public
State of California
County of San Luis Obispo

CHUCK MACKEY
Commission # 1397547
Notary Public - California
San Luis Obispo County
My Comm. Expires Feb 1, 2007

**PG&E Response to Request for Additional Information
Humboldt Bay Independent Spent Fuel Storage Installation (ISFSI)
License Application**

Question 3-1

This Question was raised in the January 13, 2005, NRC meeting.

SAR Section 3.1 states that all fuel assemblies utilize Zircaloy cladding. During various meetings, PG&E has indicated that the loose fuel debris being handled may be stainless steel clad. The SAR does not address stainless steel cladding. Please clarify if stainless steel clad fuel debris is to be to be stored. If storage of stainless steel clad fuel debris is proposed, please discuss the effect of the shielding analyses and whether this material is compatible with other fuel debris materials being stored and the amount of this material.

PG&E Response to Question 3-1

Humboldt Bay Power Plant (HBPP), Unit 3, originally used stainless steel clad fuel. However, this fuel experienced fuel structural failures and all of the "intact" stainless steel clad fuel assemblies were shipped offsite and reprocessed. Some stainless steel clad debris remains in the spent fuel pool. Stainless steel clad debris was considered during the design and analyses for the ISFSI and determined to not be significant. SAR Section 3.1 did not explicitly describe stainless cladding for loose fuel debris as there is no credit taken for cladding integrity, and the fuel characteristics assumed in the analyses bound all other parameters. Some of the considerations were: (1) the stainless steel clad fuel had less burnup than the Zircaloy clad fuel; (2) the cooling time for the stainless clad fuel was approximately 10 years more; and (3) the overall amount (less than one fuel bundle's worth (out of 80 in any one cask) was considered below consideration. In summary, the amount of stainless steel clad debris is less than the equivalent of one fuel assembly, it has no effect on the shielding analyses and is compatible from a materials aspect with other materials proposed to be stored. SAR Section 3.1 will be revised as follows to explicitly state that stainless steel clad debris is proposed to be stored in the ISFSI.

SAR Section 3.1.1, paragraph 1 will read as follows:

3.1.1 MATERIAL TO BE STORED

The materials to be stored at the ISFSI consist of intact fuel assemblies, damaged fuel assemblies, and GTCC waste. The fuel assemblies may be stored with, or without channels. There are 390 fuel assemblies in the HBPP inventory, and a quantity of loose debris that could constitute an equivalent of one additional assembly. Each fuel assembly contains approximately 192 pounds (87 kg) of UO_2. Damaged fuel is stored in a damaged fuel container (DFC) in an MPC-HB in accordance with ISG-1, Revision 1 (Reference 1). Damaged fuel in the form of loose fuel rods, fuel pellets, etc. can be consolidated; however, the amount of fuel material in a single DFC is limited to the total fissile material and weight of a single intact fuel assembly. The loose debris may be stored in one or two DFCs to optimize the retrieval and handling operations.

SAR Section 3.1.1.1, paragraph 1, will read as follows:

3.1.1.1 Physical Characteristics

The spent fuel assemblies to be stored consist of General Electric Type II (a 7 x 7 array of fuel rods), General Electric Type III, Exxon Type III, and Exxon Type IV (a 6 x 6 array of fuel rods) fuel assemblies. Construction details for each type are similar (References 2 through 5). The main support structure for an assembly consists of fuel rods used as tie rods between upper and lower tie plates. All assemblies use three spacer grids attached to a single spacer capture rod to maintain fuel rod spacing. The licensing basis fuel cladding material for all assemblies is any zirconium-based alloy, consistent with the HI-STAR 100 System Certificate of Compliance. Fuel records indicate that all HBPP fuel cladding material for intact and damaged assemblies is Zircaloy-2. Channels are fabricated from Zircaloy material. The loose fuel debris described in section 3.1.1 may have either Zircaloy cladding, stainless steel cladding, or may be unclad pellets. A summary of the physical characteristics of the Humboldt Bay fuel proposed for storage at the ISFSI is shown in Table 3.1-2.

SAR Section 3.1.1.2, paragraph 1, will read as follows:

3.1.1.2 Thermal and Radiological Characteristics

The thermal and radiological characteristics of the HBPP fuel to be stored are summarized below, and constitute limiting values for storage of fuel assemblies at the Humboldt Bay ISFSI. The values listed below were used in the analyses supporting the design and bound the actual values for these parameters for the entire HBPP spent fuel inventory to provide margin. All loose fuel debris is bounded by these parameters.

<u>*Question 4-2*</u>

Given the need for the underground vault to withstand seismic forces, what plans are there to ensure the continued integrity of the reinforcing steel (rebar) of the vault in the presence of groundwater that will intrude into the inevitable shrinkage cracks in the overlying concrete?

The underground vault, by its nature, is not readily observable over its exterior surfaces for detection of degraded conditions (cracks, rust streaks, spalling concrete). It must be assumed that shrinkage cracks will form in the vault concrete and that many of these *cracks will penetrate to the rebar inside. This provides a path for groundwater intrusion to the rebar. Since the ISFSI site is coastal, it is assumed that the salt content of the soil, and consequently the groundwater, will be significant and pose a corrosion threat to the rebar. Absent a practical inspection program (surveillance), it appears that a mitigation plan is required to assure continued structural capability, by minimizing corrosion.*

The staff recognizes the use of coated rebar as an aid toward reducing this problem. However, it does not prevent corrosion at the numerous, and ever-present coating holidays (breaks). The only way to prevent this form of degradation is by use of a cathodic protection system. Marine, bridge, and other structures with buried structural components subjected to other than just compressive loads are now built with coated rebar in conjunction with a cathodic protection system (the coating serves to minimize the size of the cathodic protection system while enhancing its ability to protect the steel).

This information is needed in accordance with 10 CFR 72.24(d)(2).

PG&E Response to Question 4-2

PG&E continues to feel that epoxy coated reinforcing or cathodic protection for the reinforcement is not warranted for this application. The vault is not directly exposed to saltwater. De-icing salts will not be used near the vault. The vault is below grade, but at elevations well above any saline groundwater intrusion areas. Seawater spray and mist at this elevation and distance from the ocean will be negligible. ACI 201.2R-01, "Guide to Durable Concrete", Section 2.4 provides several recommendations for concrete structures exposed to seawater environments. A review of these recommendations indicates that the construction practices to be used to conform to ACI 349 standards also meet the suggested practices, where appropriate. The concrete water-to-cement ratio will meet or exceed 0.45 to limit any possible chemical attack on the cement paste. The concrete cover (3 inches minimum on all surfaces, except 2 inches on the top surface) for the reinforcement will limit any aggravated corrosion of the reinforcement. Providing concrete compressive strength in excess of 4,000 psi (as discussed in ACI 201.2) is not considered warranted because the structure is not submerged in saltwater and salt spray on the upper surface will be insignificant.

Existing HBPP structures in more severe environments (located in the groundwater zone) have evidenced adequate corrosion resistance for a nearly 50-year period.

Question 5-1

A. *The applicant should address the following specific comments regarding the fuel spacers design (Supplement 2 of Holtec International report, HI-2033035):*

A.1 *Buckling load calculation:*

- *PG&E computes the critical load using the Euler equation for a column with pin restrains on both ends, which is not representative of the fuel spacer boundary conditions.*
- *The critical load derived from Euler equation is only valid for components with a large slenderness ratio (KL/r). For sections with a small or medium slenderness ratio, as in the case of the fuel spacers, empirical buckling equations or more refined structural analyses substitute Euler equation. For sections with small slenderness ratio, buckling limits the allowable stress that the components can withstand. ASME code (2004) addresses this situation in Appendix F for linear-type elements with Level D service limits.*

A.2 *PG&E should provide the reasoning for using a permissible stress for the design of fuel basket spacers that exceeds the limits provided in Appendix F of ASME Code (2004)*

B. *Regarding the design of the cell walls (Supplement 2 of Holtec International report, HI-2033035), the applicant should address the following:*

B.1 *PG&E claims that a conservative approach has been carried out by distributing the compressive load of a single basket spacer over two intersecting cell walls. NRC staff considers, however, that most of the load must be resisted by the sections of the wall that are in the immediate surroundings of the spacer, which is where the largest stresses will take place. The staff recognizes that some resistance will be provided by adjacent cell walls, but it is not possible to account for them in such a simplified analysis. Therefore, if PG&E intends to take credit for areas of the cell walls that are not in direct contact with the fuel spacers, they need to provide an analysis that includes non-uniform compression of the cell walls.*

B.2 *The allowable stress under accident conditions is the same stress used for the design of the fuel spacers, i.e., Sallow_ac = 36.9 ksi, which is equivalent to the primary membrane stress intensity, Pm, as defined in HI-STAR 100 in Tables 3.1.6 and 3.1.17. Section F-1332 (Criteria for Plate and Shell Type Supports) Appendix F of ASME Code (ASME, 2004) indicates that the primary membrane stress intensity is limited to the greater of 1.2Sy and 1.5Sm, but may not exceed*

Su. Table 3.3.1 of HI-STAR 100 indicates that at 725°C, Sy = 17.1 ksi (thus, Sallow_ac = Pm = 2.2Sy) and Su = 62.3 ksi (Sallow_ac = Pm = 0.6Su).

B.3 *PG&E performs an additional eigenvalue buckling analysis to show that the fuel basket can withstand the design basis deceleration of 60 g in the lateral direction. The analysis, however, is inadequate for analyzing the load transferred from the spacers to the cell walls, because it is performed in the plane perpendicular to the direction in which the basket spacers transmit the load to the cell walls.*

PG&E Response to Question 5-1

A.1 The Euler equation will no longer be used to determine the buckling load for the fuel basket spacer. Instead the maximum load in axial compression will be determined in accordance with the rules of F-1334 (Criteria for Linear Type Supports) of the ASME Code. In addition, the boundary conditions for the fuel basket spacer will be treated as one end free and the opposite end clamped. The spacer design will be modified as necessary to accommodate these analysis changes.

A.2 The fuel basket spacer will be redesigned to comply with the limits provided in Appendix F of the ASME Code. Moreover, the governing code for the fuel basket spacers will be changed to ASME Code Section III, Subsection NF in Section 4.2.3.3 of the Humboldt Bay ISFSI SAR.

B.1 The revised spacer design and supporting analysis will only take credit for the basket cell walls in direct contact with the fuel basket spacers.

B.2 The stress intensity limits for the Multi Purpose Canister (MPC) fuel basket under various loading conditions are clearly stated in Table 2.2.11 of the HI-STAR Final Safety Analysis Report (FSAR) (Holtec Report HI-2012610, all approved revisions). For Level D service conditions, the primary membrane stress intensity limit (Pm) is stated as governed by Appendix F, Paragraph F-1331 of the ASME Code, which states Pm shall not exceed the lesser of 2.4Sm and 0.7Su. At 725°F, Sm = 15.4 ksi (Table 3.1.15) and Su = 62.3 ksi (Table 3.3.1) for Alloy X material. Thus, Pm = 2.4Sm = 36.9 ksi.

B.3 The longitudinal stability calculation in Subsection 3.4.4.3.1.3 of the generic HI-STAR 100 FSAR (that follows NUREG/CR-6322) will be submitted using the geometry of the MPC-HB.

Holtec Report HI-2033035, Revision 2, has been developed to address the applicable concerns above, and will be submitted to the NRC at a later date.

Question 5-2

In response to NRC RAI 5.2 (Pacific Gas and Electric, 2004b), PG&E has updated sheet 3 of drawing 4082 (Figure 3.3-3 of the SAR). The geometry of the top flange and the trunnion presented in this drawing is different from that used for the structural calculations of Section 3.4.3.1, Appendix 3.D and Appendix 3.Y of HI-STAR 100 FSAR (Holtec International, 2002a). Thus, PG&E needs to provide structural calculations consistent with the final drawing of the top flange and trunnion for the HI-STAR HB, in which the single failure proof criterion is satisfied in accordance to NUREG-0612 (USNRC, 1980) and ANSI-14.6 (American National Standards Institute, 1993).

PG&E Response to Question 5-2

Sheet 3 of drawing 4082 (Figure 3.3-3 of the HB ISFSI SAR) as revised is in agreement with drawing 3913 in section 1.5 of the HI-STAR 100 FSAR for the same component. The calculations presented in Appendix 3.D of the HI-STAR 100 FSAR (to demonstrate compliance with ANSI-N14.6 for the lifting trunnion itself) are in agreement with the HI-STAR 100 FSAR and HB SAR drawings with one exception. A smaller, more conservative value for the mean thread diameter was used at one point in the calculation. Therefore, the results from Appendix 3.D remain acceptable, apply directly to the Humboldt Bay HI-STAR, and demonstrate compliance with ANSI-N14.6.

Differences between the top flange configuration analyzed by the finite element method in Appendix 3.Y and the top flange configurations shown in the Drawing 4082 and Drawing 3913 are explained as follows:

The calculations in Appendix 3.Y solely address the evaluation of primary stresses in the top flange during lifting. These calculations demonstrate that any load totaling less than 3 times the lifted load, result in no primary stress exceeding the material yield strength per Regulatory Guide (RG) 3.61. The calculations in Appendix 3.Y were originally performed using a configuration with an opening for a smaller diameter trunnion and a shorter thread engagement and used a conservative assumption of a compression-only, frictionless surface contact between the trunnion body and the inside surface of the trunnion cavity in the top flange. Thus, the trunnion body itself transferred only normal surface compressive stress to the cavity and provided no shear support to the top flange. As part of the generic HI-STAR licensing process, the trunnion was increased in diameter to ensure a larger safety factor for the calculation in Appendix 3.D. The analysis in Appendix 3.Y did not need to be modified since the insertion of a larger trunnion (an insert made from a material with larger yield strength than the top flange) coupled with more realistic interface conditions resulted in safety factors greater than the values reported in the HI-STAR 100 FSAR, and therefore continued to be in compliance with RG 3.61. In addition, there is additional safety margin for the HI-STAR HB since the same trunnion and upper cask flange dimensions exist, and the overall weight of the HB is significantly less than the generic HI-STAR.

Question 5-3

A. *The applicant needs to provide the validation report HI- 2022896 (Holtec International, 2002b), and any other documentation necessary for validating the use of Visual Nastran program for performing dynamic analyses of the casks, including cask drop and tip-over analyses.*

B. *Staff review of Holtec International (2003a, HI-2033046) determined that the properties of the impact surface used in the analyses are representative of the soil material only, and do not include reinforced concrete (RC) pad properties. Thus, the applicant should justify the use of soil as a bounding impact surface, or modify the analyses to include the RC pad.*

PG&E Response to Question 5-3

A. Holtec Report HI-2022896 will be submitted to the NRC at a later date. This report demonstrates that the Visual Nastran program can be validated for use to model dynamic analyses of the casks, including cask drop, and tip-over analyses.

B. Holtec Report HI-2033046 evaluated the configuration of the HI-STAR HB cask on the rail dolly in the Humboldt Bay Refueling Building (RFB) and approximately 100 feet in a southerly direction from the RFB on a level gravel surface until it is placed on the transporter. As discussed in PG&E Letter HBL-04-016, dated July 9, 2004, the licensing basis for the Humboldt Bay RFB was consequence-based; i.e., the RFB was assumed to fail since it was designed to 0.5g and the new design basis earthquake would exceed this acceleration. The offsite consequences were evaluated and determined to result in offsite radiological consequences that were a small fraction of the 10 CFR 20 and 100 limits. In developing its design approach for the seismic aspects in the RFB, PG&E decided to qualify all new equipment to be installed in the RFB to 0.5g since the structural integrity of the building had not been analyzed for accelerations exceeding these levels. In addition, in order to maintain the existing licensing basis, PG&E designed a cask lid restraint to maintain the fuel within the cask during cask handling activities until such time that the lid was welded. SAR Section 8.2.1.2.1 indicates that the HI-STAR HB cask while on the dolly in the RFB would not slide off the dolly or tip over during a 0.5g acceleration earthquake.

Therefore, the analysis contained in Holtec Report HI-2033046 results in a requirement for the impact surface at Humboldt Bay to have an effective Young's Modulus of the substrate plus any concrete or asphalt overlay less than or equal to 28,000 ksi. This is a condition on the travel path of the dolly, prior to the cask being loaded onto the transporter and moved to the ISFSI.

<u>*Question 5-4*</u>

The seismic analysis of the RC vault does not consider potential amplifications of acceleration forces due to soil-structure interaction, SSI (Holtec International 2003b, HI-2033013). Staff concurs that the storage vault structure can be considered "rigid" but that the response of this "rigid" structure on the supporting soil (soil-vault-cask system) has frequencies in the amplified region of the uniform hazard spectra (UHS) provided by PG&E (SAR).

In its response to RAI 5.4 (PG&E, 2004b), PG&E utilizes available literature, mainly Stewart et al (1998), to claim that the amplifications of accelerations due to SSI are not feasible, or that several factors may mitigate their impact on the response. PG&E sources, however, only provide statistical information derived from structural systems identification. The geometry of the storage vault, as well as the soil conditions and the extreme earthquake hazards of the HB site, include particularities that cannot be addressed based on general trends presented in Stewart et al (1998). Nevertheless, NRC staff agrees with PG&E in that some factors may mitigate the potential of amplifications of accelerations, particularly, the fact that the soil surrounding the vault will provide confinement in the horizontal direction. The top of the vault, however, is not restrained and the soil embedment effect does not take place in the vertical direction, which according to the UHS (SAR) is the direction with the largest spectral accelerations. Thus, NRC considers that PG&E still has to address the potential of amplifications in, at least, the vertical direction. The applicant has several options:

i) *show quantitatively that amplifications are not feasible based on PG&E UHS and the vibrational periods of the soil-vault-cask system*

ii) *carry out the pseudo-static seismic analysis using spectral accelerations of PG&E UHS corresponding to the estimated first natural periods of vibration of the soil-vault-cask system (considering reasonable uncertainties). The analysis should also address the dynamic response of partially loaded RC vault, which will lead to non-uniform mass distribution. The vault-cask interaction analysis (Holtec International, 2003c; HI-2033014) should also be based on the above spectral accelerations.*

iii) *perform a dynamic seismic analysis including SSI for the soil-vault-cask system.*

Any of these options will satisfy 10 CFR 72.122(b)

<u>PG&E Response to Question 5-4</u>

PG&E's response to NRC Question 5-4 will be submitted at a later date.

<u>*Question 6-1*</u>

Using the Holtec International (2003, Holtec Report HI-2033033) thermal modeling methodology for the reinforced concrete storage vault it was established that the thermosyphon effect within the cask will affect predicted storage vault temperatures. The Humboldt Bay SAR (Pacific Gas and Electric Company, 2004a) does not explicitly indicate that credit is being taken for this phenomena. Furthermore, neither Holtec International (2003, Holtec Report HI-2033033) nor the responses to the applicable NRC requests for additional information (Pacific Gas and Electric Company, 2004b) provided a definitive explanation, either directly or by citation, as to how the effective thermal conductivity and decay heat distribution within the cask was determined. Lastly, the HI-STAR 100 FSAR (Holtec International, 2002), which was cited as the basis for the HI-STAR 100 HB thermal analysis, clearly indicated that the thermosyphon effect was not being credited.

<u>PG&E Response to Question 6-1</u>

The thermosiphon heat transfer mechanism has been reviewed and approved for Holtec MPCs in the HI-STORM System (Docket 72-1014). Page 1 of Holtec thermal analysis report, HI-2033033 Revision 0, states that the methodologies used in the HB thermal analysis "are in full accord with prior work done by Holtec in licensing the HI-STAR and HI-STORM Systems." The construction of the HI-STAR MPC-HB is essentially the same as the MPCs used in the HI-STORM System, which have interconnected flow paths that allow helium to circulate within the MPC. Credit for internal convection flow through the MPC-HB fuel basket is mentioned on pages 5 and 6 of Holtec Report, HI-2033033 Revision 0, and other statements relating to flow through the fuel assemblies are on page 9 (items v, vi and viii). Page 26 of this report describes the thermosiphon mechanism, albeit without using the actual term.

SAR Section 4.2.3.3.5, paragraph two of Amendment 1 will be revised to read as follows:

The HI-STAR HB System is designed to transfer decay heat from the spent fuel assemblies to the environment. The MPC design, which includes the all-welded honeycomb basket structure, provides for heat transfer by conduction, convection, and radiation away from the fuel assemblies, through the MPC basket structure and internal region, to the MPC shell. The internal MPC design incorporates top and bottom plenums, with interconnected downcomer paths, to accomplish convective heat transfer via the thermosiphon effect. The thermosiphon heat transfer mechanism is credited in the HB thermal calculations contained in Holtec Report HI-2033033, Revision 0, and is consistent with prior HI-STORM licensing of heat transfer via the themosiphon effect. The MPC is pressurized with helium, which assists in transferring heat from the fuel rods to the MPC shell by conduction and convection. The stainless steel basket conducts heat from the individual spaces for storing fuel assemblies out to the MPC shell.

Question 6-2

The Holtec International (2003, Holtec Report HI-2033033) axisymmetric thermal modeling methodology for the cask and vault overestimated the surface area conducting heat to the surrounding soil (causing the calculated temperatures to be underestimated). The potential increase in temperatures that could occur after correcting the surface areas may be enough to exceed the 93°C [200°F] concrete temperature limit delineated in Table 4.2-10 of the SAR (Pacific Gas and Electric Company, 2004a). See Item 6-3 for additional information relevant to this item.

PG&E Response to Question 6-2

The Humboldt Bay axisymmetric model uses an effective surface heat transfer coefficient, h_i, to model heat dissipation from the vault to the surrounding soil. This approach employs an analytical result for the thermal resistance, R_i, of soil defined as the ratio of vault-to-soil temperature difference (ΔT) and heat dissipation to soil per cask (Q_o). Using R_i, an equivalent h_i is determined to ensure that the axisymmetric model will yield the same Q_o for a given ΔT. The details of this equivalencing procedure are given in Holtec Report HI-2033033, Revision 0, Sections 7.2 and 7.3, which was submitted to the NRC in PG&E Letter HIL-03-004, dated December 15, 2003. The axisymmetric modeling properly accounts for the overstated surface area and the results obtained from the model reflect vault surface temperatures.

To allow greater margin for the vault temperature, SAR Table 4.2-10 will be revised as indicated in PG&E Response to Question 6-3 to reflect the higher allowable concrete temperature of 300°F for the Type II materials being used to construct the concrete storage vault. The Type II materials satisfy the requirements of ASTM C33-03.

Question 6-3

The applicant indicated in response to requests for additional information that the reinforced concrete storage vault would be constructed using Type II cement and fine and coarse aggregates that satisfy the requirements of ASTM C33-03. Therefore, if the reinforced concrete storage vault temperatures of general or local areas exceed 93°C [200°F] but would not exceed 149°C [300°F], no tests to prove capability for elevated temperatures and no reduction of concrete strength are required (per NUREG-1567). The reinforced concrete storage vault temperature limit delineated in Table 4.2-10 can be updated to reflect the use of Type II cement and fine and coarse aggregates that satisfy the requirements of ASTM C33-03.

PG&E Response to Question 6-3

SAR Table 4.2-10 will be revised as indicated below to reflect the higher allowable concrete temperature of 300°F for the Type II materials being used to construct the concrete storage vault.

Component	Calculated Temperature (°F)	Normal Condition Limit (°F)
Fuel Cladding	373	752
Holtite-A Neutron Shield	195	300
MPC Shell	203	450
Local Concrete	175	300
Bulk Concrete	145	150

Question 6-4

Construction specifications should be provided that identify the storage vault backfill soil or aggregate. This backfill should have a thermal conductivity that is greater than or equal to the native soil assumed in the decay heat removal assessment analyses. Section 3.3.1.5.2 of the SAR states that soil will be used as backfill around the exterior of the vault. Figure 3.2-1, Sheet 4 of 5, of the SAR indicates, however, that an unidentified material other than the native soil could be used for the backfill.

PG&E Response to Question 6-4

The material used to backfill around the concrete vault structure will be composed of the native soil that was excavated to allow construction of the underground storage vault. If the quantity of native soil is not sufficient to completely backfill the excavated area, then material with a thermal conductivity greater than or equal to the native soil will be used to complete the backfill process.

SAR Section 3.3.1.5.2, paragraph 2, Amendment 1, will be revised as follows:

The below-grade vault, shown in Figure 3.2-1 provides both man-made and natural shielding for the loaded HI-STAR HB casks during storage operations. The vault walls are constructed of reinforced concrete with native soil backfilled around the exterior. If the quantity of native soil is not sufficient to completely backfill the excavated area, a material with thermal conductivity greater than or equal to the native soil will be used to complete the backfill. Each vault employs a bolted lid made of steel-encased concrete.

<u>*Question 6-5*</u>

A commitment was made to monitor the temperature of the vault air space for a time period of 6 months to validate the actual heat rejection performance of the cask system (see Section 3.3.1.3.2 of the SAR). Based on the information provided, however, it is not clear whether this monitoring will commence when the first storage cask is emplaced within the vault and will continue for 6 months from that time onward or for 6 months after all of the casks have been emplaced.

<u>PG&E Response to Question 6-5</u>

Vault temperature monitoring will commence when the first cask with spent fuel is placed into the vault and will continue for 6 months after the last cask with spent fuel has been placed into the vault

SAR Section 3.3.1.3.2, Amendment 1, will be revised as follows to clarify the duration of time that the vault air space temperature will be monitored:

No instrumentation is required for storage of spent nuclear fuel and Greater Than Class C (GTCC) waste at the Humboldt Bay ISFSI. Due to the welded closure of the MPC, and the passively cooled storage cask design, the loaded overpacks do not require continuous surveillance, monitoring, or operator actions to ensure the safety functions are performed during normal, off normal, and postulated accident conditions. Although not required for safe ISFSI operation, temperature monitoring of the vault air space will be temporarily performed for a sufficient time period (6 months) to validate information as to the actual heat rejection performance of the cask system. Monitoring the temperature of the vault air space will commence when the first cask with spent fuel is placed into the vault and will continue for 6 months after the last cask with spent fuel has been placed into the vault.

<u>*Question 6-6*</u>

For the purpose of conservatism, a 30-minute cask fire exposure duration was assumed consistent with the requirements of 10 CFR §71.73(c)(4). The resulting thermal analysis indicated a cladding temperature of approximately 434°C [814°F] (see Holtec Report HI-2033006). This temperature is below the allowable accident temperature threshold for the SNF cladding. Table 8.2-11 of the SAR, however, needs to be updated to be consistent with this result.

<u>PG&E Response to Question 6-6</u>

SAR Table 8.2-11 will be revised as shown below to match the results of Holtec Report HI-2033006, Revision 3 as submitted in PG&E Letter HIL-04-009, dated

October 1, 2004. The revised maximum temperatures remain well within the accident allowables for the subject materials.

Cask System Component	Maximum Temperature (°F)
Fuel Cladding	813.6
Fuel Basket Periphery	587.7
MPC-HB Enclosure Vessel Inner Surface	259.2
MPC-HB Enclosure Vessel Outer Surface	258.9
Overpack Inner Shell Inner Surface	251.1
Overpack Inner Shell Outer Surface/ Intermediate Shells Inner Surface	250.5
Overpack Intermediate Shells Outer Surface/Neutron Absorber Inner Surface	249.5
Overpack Neutron Absorber Outer Surface/Enclosure Shell Inner Surface	1189.4
Overpack Enclosure Shell Outer Surface	1208.7

Question 7-1

Describe the method for estimating the amount of crud on a fuel assembly cited in RAI response 7-3 and in SAR 7.2.1.4, Amendment 1.

In response to RAI 7-3, the applicant stated that a conservative estimate of the crud deposition on a fuel assembly was determined to be 112 mg/cm2. Describe how this estimate was made and include justification for why this estimate is conservative.

This information is needed to confirm compliance with 10 CFR 72.104(a) and 72.106(b).

PG&E Response to Question 7-1

The estimate is based on data that was collected early in the plant operating history, by Radiochemists from General Electric Co (GE). Although most of the interest was focused on the chemical and radiochemical composition of the deposits on the fuel, some deposition density sampling was performed in order to assess the effects of the plant design (carbon steel piping) and feedwater tube material (brass, replaced with stainless steel in 1965).

A GE "Trip Report" dated December 12, 1965, includes a description of sampling the crud on two assemblies, by abrading the surface and using suction to collect the abraded material. The observed crud deposition (milli-gram per square centimeter) was as follows:

Assembly A-59 (From the periphery of the core, 6200 MWD/T)							
Inches From Top Tie Plate of Bundle	Rod #1 mg/cm^2	Rod #2 mg/cm^2	Rod #3 mg/cm^2	Rod #4 mg/cm^2	Rod #5 mg/cm^2	Rod #6 mg/cm^2	Rod #7 mg/cm^2
4.5	14						15
17	27						25
32	25	18	19	20	18	12	16
47	27						13
58	28						13
70							18
81						112	68

Assembly A-147 (From the center of the core, 7700 MWD/T)							
Inches From Top Tie Plate of Bundle	Rod #1 mg/cm^2	Rod #2 mg/cm^2	Rod #3 mg/cm^2	Rod #4 mg/cm^2	Rod #5 mg/cm^2	Rod #6 mg/cm^2	Rod #7 mg/cm^2
4.5							10
17							15
32	48	21	21	14	30	9	14
47	21						28
58	26						16
70							33
81						47	30

Results for sampling two more assemblies are discussed in a letter from R. N. Osborne (GE Reactor Chemistry) to J. C. Carroll (PG&E General Office), dated June 5, 1967. The sampling of assembly A-58 (with exposure of 10,372 MWD/T), and of assembly A-113 (with exposure of 5,838 MWD/T), was performed on February 9, 1967. The discussion related to the amount of observed deposit is quoted below:

> "The amount of deposit also varied along the length of both fuel assemblies. The top three-quarters of the two assemblies had a fairly uniform deposit of from 8 to 20 mg/cm^2. Then there was a definite increase in the amount of deposit over the bottom quarter, which reached a maximum between 30 and 112 mg/cm^2 at the bottom samples that were taken approximately 3 inches above the tie plate. A similar trend was noted in the 1965 samples; however, it wasn't as clearly defined."

A GE "Memo Report" prepared by M.E. Meek (GE Reactor Chemistry), dated October 1967, includes data for assembly A-58 (with an exposure of 10,400 MWD/t). The following table presents the results reported for sampling on February 9, 1967. The results were presented as the average of multiple rods sampled, after correcting previously determined results for a more accurate determination of the sampling area.

Sample Location, Inches from Top of Fuel	Average Deposit at Sample Elevation, mg/cm^2
3	24
11	15
23	15
31	15
43	24
51	15
67	35
76	105

The October 1967 GE "Memo Report" also described sampling crud from assemblies B-76 (with exposure of 4,300 MWD/T) and C-29 (with exposure of 3,400 MWD/T), with the following results:

Sample Location, Inches from Top of Fuel	Average Deposit at Sample Elevation, mg/cm^2	
	Assembly C-29	Assembly B-76
2	0.8	3.2
9	1.0	3.6
21	3.6	6.2
29	3.4	9.5
43	4.3	12.0
54	4.6	17.9
66	16.0	34.6
79	15.9	11.4

The conclusions of the October 1967 "Memo Report" related to the amount of deposition are quoted below:

> The total deposit on the three fuel elements C-29, B-76, and A-58 were heaviest at or near the bottom (inlet).
>
> Total deposits of iron, copper, nickel, and zinc individually were heaviest at or near the bottom.
>
> Replacement of the copper-zinc feedwater heaters with stainless steel feedwater heaters decreased but did not eliminate deposition of copper and zinc.
>
> Deposition of iron may have decreased between cycles II and III independent of the feedwater heater replacement.

A GE "Memo Report" prepared by W. Lim and H.C. Sebastian (GE Reactor Chemistry), dated October 1968, discusses sampling assembly C-85 (with approximately 10,000 EFPH at 165 MWt) and assembly B-67 (with approximately 19,000 EFPH at 165 MWt). Assembly C-85 had an average deposit of 27.8 mg/cm^2 on its bottom portion, with an average deposit of 4.1 mg/cm^2 on the center portion of its rods. The average deposition for assembly B-67 was not explicitly stated in units of mg/cm^2, but it can be calculated from the data presented to have been about 15.0 mg/cm^2 on the middle of the rods, and about 27.4 mg/cm^2 on the bottom of the rods.

The corrosion rate of the plant carbon steel piping would be expected to be highest at the beginning of unit operation, and then to subsequently decrease. The deposition of material on the heat transfer surface of the fuel is expected to be directly related to the amount of corrosion products entering the reactor, and the length of time (exposure) for deposition to occur. The available data shows that the deposition is higher with higher exposures, and that the deposition rate (i.e., mg/cm^2 per MWD/t) decreased subsequent to the replacement of the brass feedwater heater tubing material, and subsequent to the change from the stainless steel clad fuel ("A" Series assemblies, with a 7x7 array) to Zirconium clad fuel ("B" Series, "C" Series, etc., with a 7x7 array). The table below illustrates this:

Assembly	Exposure (MWD/t)	Available sample results, mg/cm^2		
		Upper Third (0 – 25 inches from top of fuel)	Middle (25 – 50 inches from top of fuel)	Bottom (50+ inches from top of fuel)
A-113	5,838	8 to 20	8 to 20	30 to 112
A-59	6,200	14 to 27	12 to 27	13 to 112
A-147	7,700	10 to 15	9 to 48	16 to 47
A-58	10,400	15 to 24	15 to 24	15 to 105
B-76	4,300	3.2 to 6.2	9.5 to 12.0	11.4 to 34.6
C-29	3,400	0.8 to 3.6	3.4 to 4.3	4.6 to 16
C-85	5,000 (approx)	N/A	4.1 (average)	27.8 (average)
B-67	9,500 (approx)	N/A	15.0 (average)	27.4 (average)

Assuming deposition is proportional to exposure, the deposition on the highest exposure assembly (about 23,000 MWD/t) would be about 6 times the results observed for assemblies B-76 and C-29, about 5 times the results observed for C-85, or about 2.5 times the results observed for assembly B-67. Then, the highest ISFSI assembly deposition would be about 5 to 37 mg/cm^2 on the top third, about 20 to 72 mg/cm^2 on the middle third, and about 28 to 208 mg/cm^2 on the bottom third, with an overall average (of these ranges) of about 18 to 106 mg/cm^2.

The highest recorded value (112 mg/cm^2) was used to estimate the deposition for fuel in the ISFSI. As discussed in PG&E's response to Question 7-3 (Reference PG&E Letter HIL-04-007, dated October 1, 2004), including the source term from crud deposition equal to 112 mg/cm^2 adds less than 1.2 percent to the gamma source term for fuel and does not change the neutron source term. If the estimated crud deposition source term were to be tripled, the resulting offsite dose using the maximum fuel assembly burnup would still be within NRC limits. SAR Table 7.5-3 shows the calculated site boundary dose for the fuel source term, is 21.5 mr/year. Increasing this by 3.6 percent does not cause the result to exceed 25 mr/year limit.

The potential increased source term from crud is insignificant in comparison to the conservatism that results from basing the MPC source term on the assumption that all fuel assemblies are at the maximum burnup for any single assembly rather than the average burnup for the assembly.

Question 7-2

Describe the operation procedures for loading, transfer, and storage of the GTCC cask. Also describe how the proposed Technical Specifications (TS) cover the operations involving the GTCC cask.

Chapter 5 of the SAR describes the operation procedures for the spent fuel casks. However, the description related to GTCC waste only states that specific procedures will be used to identify and select GTCC waste for loading in a GTCC waste only cask. However, the chapter doesn't describe the procedures for loading, transfer, or storage of the GTCC cask. A description of these procedures should be provided and include verification of cask integrity and dose rate assumptions that are relied upon in the shielding and radiation protection analyses. Description of the GTCC cask procedures as they compare to those for the spent fuel casks, explaining differences in the procedures, would be sufficient.

Also, the TS should cover not only the procedures and programs for the spent fuel casks but should include the relevant procedures and programs for the GTCC cask as well. Describe the portions of the current proposed TS that apply to the GTCC cask and how they cover the GTCC cask procedures.

This information is needed to confirm compliance with 10 CFR 72.104, 20.1201, and 20.1301.

PG&E Response to Question 7-2

At the Humboldt Bay ISFSI, GTCC waste will not be stored in the same cask as spent fuel. The spent fuel cask specific limits regarding heat load issues are not applicable for the GTCC waste as there is negligible heat load and no criticality concern. As a result, a separate GTCC cask will be provided. Per SAR Section 3.3.1.1, the GTCC cask will be certified as being designed to safely store the type of waste to be loaded (i.e., nonfuel material not requiring neutron poisons or fuel assembly-sized storage locations) while being compatible with the spent fuel cask lift devices and cask transporter.

Operations to load and place the GTCC cask at the storage location in the ISFSI vault are performed both inside and outside the Humboldt Bay Refueling Building (RFB). GTCC cask loading and handling operations are performed inside the RFB using existing HBPP systems and equipment for radiation monitoring, decontamination, and auxiliary support, augmented as necessary by ancillary equipment specifically designed for spent fuel cask loading and handling operations as described in the Humboldt Bay ISFSI SAR Chapter 5. This includes the use of the davit crane and cask transfer dolly for heavy lifts and other cask movements. The implementing procedures used for GTCC cask loading and handling will incorporate applicable 10 CFR 50 license conditions and commitments, such as those governing heavy loads. Loading of GTCC waste into the cask and transfer of the GTCC cask to the storage location will be performed using procedures developed specifically for these operations. Specific procedures will identify and control the selection of GTCC waste for loading into a GTCC qualified cask.

The radioactive material in the GTCC cask will primarily be segments of reactor core structure and other reactor internals (neutron activated Type 304 stainless steel), with gamma radiation primarily from Cobalt-60, rather than from Cs-137 (as is the case for the spent fuel), and with essentially no neutron radiation (see SAR Table 3.1-3). Planning the GTCC cask packaging and loading will involve a determination of the isotopic radioactivity content of the materials to be placed in the cask, followed by calculating the dose rates expected at the surface of the cask in a manner similar to that for nonfuel sources, as described in SAR Section 7.2. If the expected GTCC cask surface dose rate could exceed the calculated SAR surface dose rate determined for the spent fuel casks, additional shielding will be incorporated within the GTCC cask until the predicted dose rates are bounded by the analyzed spent fuel cask dose rates. The validity of the calculations will be confirmed by measuring the loaded GTCC cask surface dose rate prior to transferring the cask from the RFB to the ISFSI. These requirements will be incorporated into the GTCC loading procedures. The GTCC packaging and loading procedures will contain suitable methods to assure the material is maintained encapsulated such that any loose particulate matter does not become airborne in storage. These methods could include a welded container, encapsulation polymer or cementicious material, or other equivalent processes.

All of the GTCC packaging, loading, and transport procedures will be controlled under the current proposed Humboldt Bay ISFSI TS Sections 2.1.4, 5.1.2, 5.1.5, and a new program Section 5.1.6. The contents of the GTCC cask are controlled by TS section 2.1.2, which references SAR Section 3.1 for a detailed list of contents. Section 5.1.2.b will be amended to address surface contamination limits on the GTCC cask and will read as follows:

> Provide limits on surface contamination of the OVERPACK and GTCC cask and verification of meeting those limits prior to removal of a loaded OVERPACK or GTCC cask from the refueling building.

The transport of the GTCC cask is controlled under Section 5.1.5.

A new TS section will be added, requiring the establishment and maintenance of a GTCC Cask Loading and Preparation Program. This new TS Section will read as follows:

5.1.6 GTCC Cask Loading and Preparation Program

This program shall be established and maintained to implement Humboldt Bay ISFSI SAR Section 3.1 requirements for loading a GTCC cask and preparing the GTCC cask for storage in the ISFSI. The requirements of the program for loading and preparing the GTCC cask shall be complete prior to removing the GTCC cask from the refueling building. The program provides for evaluation and control of the following requirements during the applicable operation:

a. Verify surface dose rates on the GTCC cask are consistent with the offsite dose analysis.

b. Verify that any effluents from the GTCC cask comply with 10 CFR 20 requirements.

<u>Question 7-3</u>

Clarify that the gates along the public trail will be locked, that the public will be kept at a distance of no less than 100 m, during cask transfer, including a transporter fire accident and while corrective actions are taken.

The SAR and Emergency Plan (EP) appear to be inconsistent regarding the condition of the gates along the public trail and whether the public will be kept at a distance of at least 100 m. EP Sections 2.3 and 5.2.2.2 state that the gates MAY be locked for a transporter fire accident and remain so until corrective actions are taken and radiation levels are reduced. However, SAR Sections 2.1.2 and 8.2.5.3 indicate that the gates WILL be closed. Also, the accident analysis relies on the gates being locked, the public being kept at a 100 m distance. Further, the SAR indicates the gates should already be locked during cask transport operations. The affected sections of the SAR, including the accident analysis, and EP should also be updated to accurately reflect the actual *distances of the public, the condition of the gates. This information is needed to confirm compliance with 10 CFR 72.106(b).*

PG&E Response to Question 7-3

The following describes how the gates of the 100-meter area will be controlled during times of cask transportation and normal ISFSI storage operations.

Transportation:

During loaded cask movements or handling evolutions, the gates will be locked to prevent public access within the 100-meter area until the cask transfer activities and any corrective actions are completed.

ISFSI Storage Operations:

The gates will normally be open to allow access to the public trail during ISFSI operation. If an accident should occur during normal ISFSI operation, PG&E will assess radiological conditions. If radiation levels exceed the allowable levels for public health and safety, PG&E will close and lock the gates to prevent public access within the 100-meter area until actions are completed to return radiation levels to allowable levels.

SAR Sections 2.1.2 and 8.2.2.4 and EP Sections 2.3 and 5.2.2.2 will be revised as follows:

SAR Section 2.1.2, paragraph 4, Amendment 1, will be revised and incorporated into the HB ISFSI FSAR, which will read as follows:

2.1.2 SITE DESCRIPTION

In accordance with 10 CFR 72.106, a 100-meter area will be established around the ISFSI, as shown in Figures 2.1-2 and 2.2-2. A public trail to access a breakwater for fishing traverses the ISFSI 100-meter area (see Figures 2.1-2 and 2.2-2), a condition allowed by 10 CFR 72.106, so long as appropriate and effective arrangements are made to control traffic and to protect public health and safety. The public trail crossing the PG&E property to the north of the ISFSI is controlled by fencing and gates. Figure 2.2-2 indicates the approximate location of the gates. The gates will normally be open to allow access to the public trail during normal ISFSI operation. If an accident should occur within the 100-meter area during normal ISFSI operation, PG&E will assess radiological conditions. If radiation levels exceed the allowable levels for public health and safety, PG&E will close and lock the gates to prevent public access within the 100-meter area until actions are completed to return radiation levels to allowable levels. During loaded cask movements or handling evolutions, the gates will be locked to prevent public access within the 100-meter area until the cask transfer activities and any corrective actions are completed. Loaded cask movements or handling evolutions will occur primarily during the initial transport of storage casks to the ISFSI and potentially not again until the casks are transported offsite to the U.S. Department of Energy permanent storage repository.

SAR Section 8.2.4, paragraph 1, Amendment 1 will be revised and incorporated into the Humboldt Bay ISFSI FSAR, which will read as follows:

8.2.2.4 Accident Dose Calculations

Extreme winds in combination with tornado missiles are not capable of damaging an MPC located within a HI-STAR HB overpack. Therefore, no radioactivity would be released due to tornado effects on the HI-STAR HB cask. If there is local reduction of shielding provided by the vault lid due to denting and/or loss of concrete, the local dose rates could increase. However, there should be no noticeable increase in the ISFSI site or controlled-area boundary dose rate because the affected area will likely be small. In the event of such an accident occurring, the 100-meter controlled area will be maintained as described in Section 2.1.2, and an ISFSI operator will first perform a radiological and visual inspection to determine the extent of the damage to the vault. As appropriate, temporary shielding would be placed around the affected area to reduce dose rates.

EP Section 2.3, paragraph 7, Amendment 1, will be revised to read as follows:

2.3 DESCRIPTION OF AREA NEAR THE SITE

10 CFR 72.106 requires a minimum distance from the spent fuel handling and storage facilities to the nearest boundary of the controlled area to be at least 100 meters.
10 CFR 72.106 allows the controlled area to be traversed by a highway, railroad, or waterway, so long as appropriate and effective arrangements are made to control traffic and to protect public health and safety. A public trail to access a breakwater for fishing transects the PG&E property and transverses the 100-meter ISFSI controlled area as shown on Figure 2.2-2. The public trail crossing the PG&E property to the north of the ISFSI is controlled by fencing and gates. The gates will normally be open to allow public access during ISFSI operation. If an accident should occur during normal ISFSI operation, PG&E will assess radiological conditions. If radiation levels exceed the allowable levels for public health and safety, PG&E will close and lock the gates to prevent public access within the 100-meter area until actions are completed to return radiation levels to allowable levels. During loaded cask movements or handling evolutions, the gates will be locked to prevent public access within the 100-meter area until the cask transfer activities are completed. Loaded cask movements or handling evolutions will occur during the initial transport of storage casks to the ISFSI and are not planned again until the casks are transported offsite to the U.S. Department of Energy permanent storage repository.

EP Section 5.2.2.2, Amendment 1, will be revised to delete Item E, Public Trail.

<u>*Question 8-2*</u>

PG&E provided a criticality safety analysis for 2 specific loading patterns: (1) intact fuel in the center baskets with damaged fuel around the edges and (2) a checkerboard loading pattern for damaged and intact fuel. Specific to this license is the unique request to load both damaged fuel and intact fuel into damaged fuel canisters (DFCs). In the staff's view, this fact and the unique loading patterns increase the likelihood for potential misloading events.

PG&E has described, in their RAI response dated October 1, 2004, procedures in place to prevent misloadings and, because of these procedures, has concluded that a misloading is considered "not credible." Because of the unique loading patterns, the limited criticality safety analysis provided by PG&E (not bounding), and the loading of intact fuel into DFCs, the staff believes that some of these administrative controls should be included in the technical specifications to provide reasonable assurance that an inadvertent criticality will not occur during fuel loading.

Please provide PG&E's TS controls for assuring assurance that an inadvertent criticality due to a cask-misloading event will not occur.

<u>PG&E Response to Question 8-2</u>

PG&E will revise TS 5.1.3 to add the following administrative control to prevent misloading events:

5.1.3 MPC-HB and SFSC Loading, Unloading, and Preparation Program

f. Loading is to be independently verified by a cognizant engineer to ensure that the fuel assemblies in the MPCs are placed in accordance with the original loading plan.

<u>*Question 11-1*</u>

Provide a description of the Health Physics program's organizational structure following decommissioning of the Humboldt Bay reactor.

In the revised SAR Section 7.6.1, the applicant states the Health Physics (HP) program's organization will change upon completion of reactor decommissioning. The final organizational structure is stated to depend upon the organizational change within PG&E that will result from decommissioning. However, no description is provided of what the anticipated, or potential, organization could be. While there may be some *changes that cannot be foreseen, a basic structure with reporting chains and designations of authorities and responsibilities should be provided.*

This information is needed to confirm compliance with 10 CFR 20.1101(a) and (b).

PG&E Response to Question 11-1

This response also addresses NRC Question 11-2.

The Director and Plant Manager, HBPP, is responsible for assuring appropriate radiation protection (RP) support services are available throughout the lifetime of the Humboldt Bay ISFSI.

Following termination of the 10 CFR 50 license, the Director and Plant Manager, HBPP, will be responsible for day-to-day management and overall safety of ISFSI activities. RP functions and operations functions will report directly to the Director and Plant Manager, HBPP. RP functions will remain organizationally independent of ISFSI operations. The RP function will be maintained for the life of the ISFSI, including decontamination and decommissioning operations.

ISFSI RP services are required to comply with the Humboldt Bay ISFSI Quality Assurance Program (QAP) as stated in the Humboldt Bay ISFSI QA Program, Chapter 17. Personnel providing ISFSI RP functions have the authority and responsibility to halt operations deemed to be unsafe, and to perform any required radiological sampling analyses and radiation and contamination surveys. In addition, they implement the personnel radiation-monitoring program, maintain RP records, and provide monitoring for work in radiologically controlled areas. Appropriate portable and laboratory equipment and instrumentation will be provided for the ISFSI as required by the health physics program. This equipment includes, but is not limited to, personal monitoring devices, portable radiation meters, air sampling equipment, decontamination equipment, equipment and/or contracts for internal radiation monitoring, and personal protective equipment.

As required by SAR Chapter 11, vendors and contractors that provide important-to-safety items and services, such as RP services, will work to a PG&E approved QAP that meets the requirements of 10 CFR 72.140. Conformance with the Humboldt Bay ISFSI QAP and procedures will ensure that any offsite facilities equipment, training and qualifications meet the requirements for having qualified RP personnel and equipment and will ensure that the offsite facilities services meet regulations and Humboldt Bay ISFSI QAP requirements.

Question 11-2

Describe how appropriate radiation protection services will be continuously assured.

In its revised SAR Section 7.6, "Health Physics Program," the applicant states that radiation protection services will be provided by an offsite facility under contract. These services include providing various radiation protection instrumentation and equipment, maintaining and calibrating of the instrumentation and equipment, performing radiation and contamination surveys, and so forth. The applicant also states that this facility will provide its own instruments and personnel and will be responsible for its own training and qualification.

Provide a description of the offsite facility. This description should include its capabilities and qualifications to provide the required equipment and services. The description should also include how the applicant intends to ensure that the offsite's training and qualifications and so forth meet the requirements for having qualified radiation protection personnel and how the applicant will ensure that the offsite facility's services and related programs meet regulations and the applicant's QA requirements.

This information is needed to confirm compliance with 10 CFR 20.1101(a) and (b), 20.1302(a), and 20.1501.

PG&E Response to Question 11-2

Refer to PG&E response to NRC Question 11-1.

Question 11-3

Describe the changes to records system and procedures related to the HP program that will result from reactor decommissioning.

A system for managing records of radiation surveys, access control, reporting/recordkeeping of exposure monitoring results, respiratory protection, and so forth seems to be currently in place. However, the procedures/system appears to be tied to the reactor's HP program. Provide a description of how the system and procedures will change with respect to these records upon decommissioning of the reactor.

This information is needed to confirm compliance with 10 CFR 20.2102, 20.2103, 20.2106, and 20.2107.

PG&E Response to Question 11-3

As stated in Humboldt Bay ISFSI SAR Section 9.4.2, Humboldt Bay ISFSI records will be maintained in accordance with the Humboldt Bay ISFSI QA Program contained in Attachment E to the ISFSI License Application. Humboldt Bay ISFSI QA Program Section 17.17.3 identifies the important-to-safety records that are to be classified as lifetime or nonpermanent and identifies the types of records that shall be maintained for the Humboldt Bay ISFSI, including radiation protection program records. QA Program Section 17.5 establishes the requirements to establish and control procedures. SAR Section 7.6 indicates that the policy and procedures (described in Sections 7.6.3) will comprise the health physics program during ISFSI operations phase. SAR Section 7.6.3 indicates that it is the policy of PG&E to operate and maintain the ISFSI in accordance with the program and implementing procedures.

Question 11-4

Submit the revised pages for SAR Section 7.5.2.

The applicant, in response to a previous RAI, revised SAR Section 7.5.2. This revision was not provided with the other revised pages. Therefore, submittal of these pages is requested.

PG&E Response to Question 11-4

A markup of Humboldt Bay ISFSI SAR Section 7.5.2 is provided below. SAR Section 7.5.2 makes reference to SAR Section 2.1.2 for additional information regarding control of the public access trail to the 100-meter controlled area. SAR Section 7.5.2 was inadvertently omitted from the Humboldt Bay ISFSI SAR Amendment 1 submittal. Information from this Attachment will be incorporated into the Humboldt Bay ISFSI SAR, Amendment 2 or the Humboldt Bay ISFSI FSAR.

7.5.2 OFFSITE DOSE FROM OVERPACK LOADING OPERATIONS

The transfer of the HI-STAR HB into the final storage configuration at the ISFSI vault will occur outside the RFB. As a result, the offsite dose for transferring the HI-STAR HB from the RFB to the ISFSI was considered. Table 7.5-2 presents the results of this analysis. The analysis assumes that access to the public trail will be controlled to keep members of the public beyond a 100-meter boundary during cask transport and vault loading operations. Refer to SAR Section 2.1.2 for additional information regarding control of the public access trail to the 100-meter controlled area.

Revised Question 15-6

Provide adequate basis for assuming 95 percent of the aircraft approaching or departing Eureka-Arcata Airport would use the V 607 route and need not be considered in estimating the crash hazard frequency. Alternatively, provide an estimated annual crash frequency of aircraft using the airway V 607.

Basis for this assumption, as given in Calculation File PRA-03-14, revision 1, is that all commercial aircraft currently are required to fly ILR flight plans. This would bring them on V 607 because of the instrument landing capabilities. However, as stated in

Calculation File PRA-03-14, revision 1, general aviation and military aircraft flights are less regulated. Consequently, they may not always use V 607 airway. http://airnav.com/airports/kcav show that on average 207 daily operations take place at Eureka-Arcata Airport. Out of these, 83 percent or 172 operations are by general aviation and military aircraft. Calculation File PRA-03-14, revision 1, has estimated the daily traffic on air corridor V 607 to be approximately 207 flights as V 607 is the main approach corridor for flights into the Eureka-Arcata Airport. Therefore, it has not been established that the assumption of 95 percent of the daily flights will use the airway V 607, as they are commercial aircraft, is conservative. It is not clear what would be the estimated number of flights in V 607 and other secondary approach and departure patterns.

Additionally, Section 3.5.1.6 of NUREG–0800 provides three proximity criteria to determine by inspection whether the annual crash hazard to a given site is less than 10^{-7} if the site meets all three criteria; otherwise, a detailed review of aircraft hazards must be performed. Note that NUREG–0800 provides a quick way to estimate whether the annual crash hazard to a particular site is below 10^{-7} or not. It does not state that the hazard is zero and need not be considered further in estimating the cumulative crash hazard. This is also true for other flights activities assumed to zero based on NUREG–0800 proximity criteria.

<u>PG&E Response to Revised Question 15-6</u>

PG&E's response to revised NRC Question 15-6 will be submitted at a later date.

<u>Revised Question 15-16</u>

Clarify the capacity of the barge. Is it 65,000 barrels or 65,000 gallons?

PG&E Response to Question 15-16 indicates the barge can hold a maximum of 65,000 gallons. Same information is in page 2.2-4 of the SAR. However, in page 2.2-4 of the SAR also states the capacity of the barge is 65,000 barrels. Other part of the SAR similar contradictory information. Calculation File No. PRA03-13, Revision 1, assumes the capacity to be 65,000 barrels.

<u>PG&E Response to Revised Question 15-16</u>

In PG&E's response to Question 15-16 the barge capacity was incorrectly stated at 65,000 gallons. It should have been 65,000 barrels as stated in the SAR and provided in the Probabilistic Risk Analysis (PRA) calculation file. The barge capacity of 65,000 barrels is the amount assumed in the PRA calculation and is the basis for all evaluations of this hazard. PG&E will assure that all references in the SAR are consistent. Any revised material will be included in the next revision of the SAR.