September 15, 2011

PG&E Letter DIL-11-0XX

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

Materials License No. SNM-2511, Docket No. 72-26
Diablo Canyon Independent Spent Fuel Storage Installation
Response to NRC Request for Information Pertaining to Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI) License Amendment Request 11-001

References:

1. PG&E Letter DIL-11-001, "License Amendment Request 11-001, Revision to Technical Specifications 1.1, 2.0, 3.1.1, 3.1.2, 4.1.2, and 5.1.3; Addition of Technical Specifications 2.3 and 3.1.4; and, Request for an Exemption from the Requirements of 10 CFR 72.236(f)," dated January 31 2011

2. PG&E Letter DIL-11-003, "Supplement to License Amendment Request 11-001, Revision to Technical Specifications 1.1, 2.0, 3.1.1, 3.1.2, 4.1.2, and 5.1.3; Addition of Technical Specifications 2.3 and 3.1.4; and, Request for an Exemption from the Requirements of 10 CFR 72.236(f)," dated June 8, 2011

3. PG&E Letter DIL-11-004, "Supplement No. 2 to License Amendment Request 11-001, Revision to Technical Specifications 1.1, 2.0, 3.1.1, 3.1.2, 4.1.2, and 5.1.3; Addition of Technical Specifications 2.3 and 3.1.4; and, Request for an Exemption from the Requirements of 10 CFR 72.236(f)," dated July 28, 2011

Dear Commissioners and Staff:

In Reference 1, Pacific Gas and Electric Company (PG&E) submitted License Amendment Request (LAR) 11-001 to the Nuclear Regulatory Commission (NRC), which proposed to revise Technical Specifications (TS) 1.1, "Definitions," TS 2.0, "Approved Contents," TS 3.1.1, "Multi-purpose Canister (MPC)," TS 3.1.2, "Spent Fuel Storage Cask (SFSC) Heat Removal System," TS 3.1.4 (new), "Supplemental Cooling System," TS 4.1.2, "Design Features Important to Criticality control," and TS 5.1.3, "MPC and SFSC Loading, Unloading, and Preparation Program."

By NRC letter dated April 14, 2011, "Diablo Canyon Independent Spent Fuel Storage Installation Materials License No. SNM-2511, Amendment Request No. 2 – Acceptance Review (TAC NO. L24515)," the NRC staff performed an acceptance review of the application to determine if the application contains sufficient technical information in scope and depth to allow the staff to complete the detailed technical review. The letter acknowledged acceptance of PG&E's application.

The NRC staff included observations that may be asked at a later date noting that responses to observations are not required for the staff to begin a detailed technical review. In Reference 2, PG&E submitted a supplement to the application addressing the observations and committing to a second supplement of the application to submit a revised thermal report addressing a FLUENT computer code issue by July 30, 2011. In Reference 3, PG&E submitted the second supplement to the application in accordance with the commitment.

On August 16, 2011, the NRC staff requested additional information required to complete its review of LAR 11-001. Enclosed is PG&E's response to the August 16, 2011 NRC request for additional information, including a removal of the Request for an Exemption from the Requirements of 10 CFR 72.236(f) transmitted in Reference 1.

The revised Holtec thermal report HI-2104625, Revision 6 is attached to the Enclosure. The revision added sensitivity analyses to Appendix B for resolution of RAI Questions 6.10 and 6.14.
The Enclosure to this letter provides three attachments. Attachment 1 of the Enclosure contains Holtec International Report HI-2104625, "Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI STORM System Design," Revision 6 – Proprietary Version, with one disc of proprietary data files on optical storage medium (OSM) DVD-ROM. Attachment 2 of the Enclosure contains an affidavit signed by Holtec, the owner of the proprietary information. The affidavit sets forth the basis on which the Holtec information contained in Attachment 1 of the Enclosure, may be withheld from public disclosure by the Commission consistent with the Freedom of Information Action ("FOIA"), 5 USC Section 552(b)(4) and the Trade Secrets Act, 18 USC Section 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.390(a)(4), and 2.390(b)(1). PG&E requests that the Holtec proprietary information be withheld from public disclosure in accordance with these laws and regulations. Attachment 3 of the Enclosure contains Holtec International Report HI-2104625, "Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI STORM System Design," Revision 6 (non-proprietary version).

Correspondence with respect to the proprietary aspects of the application or the Holtec affidavit provided in Attachments 1 and 2 of the Enclosure should be addressed to Ms. Kelly Kozink, Holtec International, 555 Lincoln Drive West, Marlton, New Jersey 08053.

PG&E makes no regulatory commitments (as defined by NEI 99-04) in this letter. This letter includes no revisions to existing regulatory commitments.

This information does not affect the results of the technical evaluation or the no significant hazards consideration determination previously transmitted in Reference 1.

If you have any questions regarding this response, please contact Mr. Lawrence Pulley at (805) 545-6165.

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

Executed on September 15, 2011.

Sincerely,

James R. Becker
Site Vice President

mjr/50249568
Enclosure
cc: Diablo Distribution
cc/enc: John Goshen, NRC Project Manager, Division of Spent Fuel Storage and Transportation
Kelly Kozink, Holtec International Project Manager
Michael S. Peck, NRC Senior Resident Inspector

PG&E Response to August 16, 2011 Request for Additional Information For Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI) License Amendment Request 11-001

This enclosure provides PG&E's response to the August 16, 2011 NRC request for additional information, including a removal of the Request for an Exemption from the Requirements of 10 CFR 72.236(f) transmitted in Reference 1.

References:

1. PG&E Letter DIL-11-001, "License Amendment Request 11-001, Revision to Technical Specifications 1.1, 2.0, 3.1.1, 3.1.2, 4.1.2, and 5.1.3; Addition of Technical Specifications 2.3 and 3.1.4; and, Request for an Exemption from the Requirements of 10 CFR 72.236(f)," dated January 31 2011.

2. PG&E Letter DIL-11-003, "Supplement to License Amendment Request 11-001, Revision to Technical Specifications 1.1, 2.0, 3.1.1, 3.1.2, 4.1.2, and 5.1.3; Addition of Technical Specifications 2.3 and 3.1.4; and, Request for an Exemption from the Requirements of 10 CFR 72.236(f)," dated June 8, 2011.

3. A. Zigh and J. Solis, "Computational Fluid Dynamics Best Practice Guidelines in the Analysis of Storage Dry Cask", WM2008 Conference, Phoenix, AZ, 2008.

<u>*NRC Question 6.1*</u>

Explain the reasons why the design ambient temperatures may be exceeded without exceeding the fuel temperature limits.

A note in Table 1.2 of Holtec Report HI-2104625 states that the design ambient temperatures may be minimally exceeded without exceeding the fuel temperature limits.

Provide site information to justify the design ambient temperatures. This information should include maximum average values and length. If the duration at ambient temperatures higher than design limits is long enough to achieve thermal equilibrium, then the higher temperature should be considered the design ambient temperatures.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

PG&E Response to Question 6.1

The average annual temperature at the Diablo Canyon ISFSI site is 55°F as specified in Section 2.3.2 of latest revision of the Diablo Canyon ISFSI UFSAR. All the thermal evaluations of HI-STORM 100 System were performed with an annual average temperature of 80°F which bounds the Diablo Canyon site annual average temperature of 55°F. The note in Table 1.2 of the Holtec report HI-2104625 Revision 5 has been deleted in Revision 6.

NRC Question 6.2

Clarify the maximum heat load and operating pressure being requested in this license amendment. Holtec Report HI-2104625 includes two different scenarios with thermal evaluations performed based on these two scenarios. The thermal calculations performed for two different scenarios are not clearly described, and the staff can't make a safety determination based on these calculations. The thermal evaluation should be modified to include only the thermal calculations for what is being requested in this amendment.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

PG&E Response to Question 6.2

PG&E is not requesting any change to the presently authorized maximum heat load of 28.74 kW at an operating pressure of 5 atmospheres absolute in this license amendment. However, to maintain consistency with the Holtec CoC 72-1014, Amendment 5, Holtec's calculation package HI-2104625, Revision 6, documents two different scenarios:

1. Scenario 1 – Heat load of 28.74 kW and Operating pressure of 5 atm absolute
2. Scenario 2 – Heat load of 36.9 kW and Operating pressure of 7 atm absolute

The peak cladding temperature and all other MPC and cask component temperatures in Scenario 2 bound Scenario 1 during normal long-term storage conditions (See Table B.5.13 of Holtec Report HI-2104625, Revision 6). Also, the MPC cavity pressures during normal, off-normal and accident conditions for Scenario 2 bound Scenario 1 (see Tables B.5.11 and B.5.14 of Holtec Report HI-2104625, Revision 6). Since the initial condition for the accident evaluations (e.g.,100 percent vents blockage and hypothetical fire accident) is the normal long term storage temperature field, the temperatures and pressures reported for accidents based on Scenario 2 initial temperature will bound the accident temperatures for Scenario 1. Therefore, the component temperatures and MPC pressure reported for all accident conditions are conservative.

<u>*NRC Question 6.3*</u>

Clarify if the multi-purpose canister (MPC) gas and fuel effective thermal conductivity included the effect of gas dilution during the 100% rod rupture accident.

Holtec Report HI-2104625 states that the 100% rod rupture accident is evaluated with due credit for increased heat dissipation under increased molecular weight of the cavity gases. However, it is not clear how the gas dilution affects the heat dissipation.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

<u>PG&E Response to Question 6.3</u>

The MPC maximum gas pressure is computed for a postulated release of fission product gases from fuel rods into the MPC cavity free space. For this scenario, the amounts of each of the release gas constituents in the MPC cavity are summed and the resulting total pressures determined from the ideal gas law. A concomitant effect of rod ruptures is the increased pressure and molecular weight of the cavity gases with enhanced rate of heat dissipation by internal gas convection and lower cavity temperatures. As these effects are substantial under large rod ruptures the 100 percent rod rupture accident is evaluated with due credit for increased heat dissipation under increased molecular weight of the cavity gases. This methodology has been previously approved by USNRC in the HI-STORM FW System (Docket No. 72-1032).

The heat transfer from the fuel region to MPC is primarily due to convection of gas inside the MPC. Heat transfer due to conduction has only a second order effect on temperatures. Therefore, an increased molecular weight and increased internal pressure due to release of gases into the MPC have a more significant effect on the heat transfer than the MPC gas thermal conductivity during the 100 percent rod rupture accident. Henceforth, the effective thermal conductivities of the fuel are based on there being 100 percent helium present even during the 100 percent rod rupture accident since conduction has a second order effect on the peak temperatures.

A credit for higher molecular weight of mixtures of gases inside the MPC is taken only for Scenario 2. Please note that <u>no</u> credit for either higher molecular weight of mixture of gases or increased cavity pressure inside the MPC is considered in the calculation of MPC pressure due to 100 percent rods rupture in Scenario 1, which is the licensing basis calculation for Diablo.

NRC Question 6.4

Clarify if the temperatures with fuel spacers reported in Table B.5.10 of Holtec Report HI-2104625 were obtained based on the finest mesh. Note 18 on this table states that all temperatures tabulated herein include the temperature adder reported in Table B.5.2 for all the components. The calculation should be performed at the design basis heat load and operating pressure for the finest mesh and the resulting temperature should be reported in the thermal evaluation.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

PG&E Response to Question 6.4

The temperatures reported in Table B.5.10 of Holtec Report HI-2104625, Revision 6, are a result of a temperature adder reported in Table B.5.2 to the baseline mesh results for the scenario with fuel spacers. The calculations reported in Table B.5.10 were not explicitly performed for the finest mesh since the effect of grid refinement in the annulus region between the MPC and HI-STORM is included by the temperature adder. It is also to be noted that the calculations are conservative because of the following reasons in addition to those already specified for the HI-STORM System in Holtec report HI-2104625, Revision 6:

1. No conduction through the fuel spacers is credited
2. The open space between the fuel spacers which facilitates natural convection of helium inside the MPC is ignored

Therefore, the calculations reported in Table B.5.10 are conservative.

NRC Question 6.5

Explain the reasons why for the same storage system Table B.5.13 of report HI-2104625 includes different temperature limits for the two scenarios described in this report. Specifically for the MPC shell, lid bottom plate, and lid top plate, the temperature limits are different. The amendment should only include results from the thermal evaluation of what is being requested. See also RAI 6-2.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

PG&E Response to Question 6.5

As mentioned in the response to RAI 6-2, two scenarios were evaluated in the Holtec Report HI-2104625, Revision 6. The temperature limits specified in the table are different for the two different scenarios. The temperature limits for Scenario 2

are based on Revision 7 of the Holtec International HI-STORM 100 FSAR. The thermal acceptance criteria for Scenario 1 are Diablo specific that must satisfy the requirements of the Diablo Canyon ISFSI UFSAR with exceptions discussed in Section 4.0 of Holtec Report HI-2104625, Revision 6. The PG&E request in this license amendment is only for Scenario 1 specified in Table B.5.13 of the Holtec Report HI-2104625, Revision 6.

NRC Question 6.6

Demonstrate that multiplying the thermal conductivity of air by 1.4 is equivalent to using equation C.2.1 of report HI-2104625 to calculate the effective radial thermal conductivity of air based on this equation without modifying the air conductivity and considering the material expansion (and therefore gap reduction.)

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

PG&E Response to Question 6.6

Equation C.2.1 in the Holtec Report HI-2104625, Revision 6, is used to compute an equivalent thermal conductivity for the annulus air gap between the MPC and the HI-TRAC. The effective radial air gap thermal conductivity includes the effects of radiation heat transfer across the annulus gap. The annulus gap between the MPC and HI-TRAC reduces by 40 percent due to radial thermal expansion of the MPC with respect to the HI-TRAC, as shown in sub-section C.5.6 of Holtec Report HI-2104625, Revision 6. However, a conservatively lower gap reduction of 30 percent is used in the calculations of the effective radial thermal conductivity of air in the annulus gap. The thermal resistance due to air in the annulus reduces due to the decrease in the annulus air gap. Since the annulus gap is explicitly modeled without considering the radial thermal expansion, the thermal conductivity is multiplied by a factor of 1.4 due to a conservative gap reduction of 30 percent. This methodology is consistent with the USNRC approved methodology in HI-STORM 100 CoC 1014 Amendment 5.

NRC Question 6.7

Clarify why a helium absolute pressure of 7 atm is conservative for MPC internal convection heat transfer during on-site transfer of the MPC in the HI-TRAC. Clarify what is the absolute operating pressure of the MPC. It appears from the amendment request that the design operating pressure is 5 atm absolute. Assuming a higher operating pressure is non-conservative because it overstates internal convection heat transfer.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

<u>PG&E Response to Question 6.7</u>

As mentioned in response to RAI 6-2, PG&E is maintaining a maximum decay heat of 28.74 kW at an operating pressure of 5 atm absolute in this license amendment request. The normal long-term storage condition is evaluated for Scenario 1 and reported in Table B.5.13 of the Holtec Report HI-2104625, Revision 6. Since the MPC cavity pressures and component temperatures are lower for Scenario 1 as compared to Scenario 2, the results reported for off-normal and accident conditions (which are based on Scenario 2) are conservative. The higher helium absolute pressure of 7 atm is used in the thermal analysis of Scenario 2 only.

<u>NRC Question 6.8</u>

Perform the on-site transfer thermal evaluation for the thermal-hydraulic conditions being requested in this amendment application. The staff can't make a safety determination if adequate supporting analyses are not provided. Most of the thermal analyses provided in the amendment request correspond to conditions which are not being requested. See also RAI 6-2.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

<u>PG&E Response to Question 6.8</u>

As mentioned in the response to RAI 6-2, two different scenarios were evaluated in the Holtec Report HI-2104625, Revision 6. Thermal evaluations of on-site transfer are reported in Appendix C of the Holtec Report HI-2104625, Revision 6. The thermal evaluations supporting Diablo ISFSI license (Scenario 1 as mentioned in the response to RAI 6-2) are discussed in sub-section C.5.2 of the report. The results of the evaluations are presented in Table C.6 of the report. It can be noted from Table C.6 that the cladding and cask component temperatures are higher for Scenario 2. Therefore, accident evaluations like loss of water from the water jacket and fire accident are conservatively performed for Scenario 2 and are respectively reported in Tables C.2 and C.3 of the report.

<u>NRC Question 6.9</u>

Provide a definition for "operable" for LCO 3.1.4. As part of surveillance requirement SR 3.1.4.1, provide monitored parameters for the system user to conclude that the supplemental cooling system is operable.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

PG&E Response to Question 6.9

The SCS is operable when the transfer cask annulus is full of demineralized water, the make-up tank is connected to the pool lid annulus drain port and equalized with the annulus water.

For surveillance of the system, the user visually inspects the transfer cask outer surfaces and connection to the make-up tank for indications of leakage, and visually confirms the water level in the make-up system and annulus. If the transfer cask top lid is installed, the water in the annulus can be validated by raising the level in the make-up system slightly until the water is visible through the downloading hole in the top lid or by using a calibrated, temperature instrument inserted into the annulus water through the top lid and reading at or below the saturation temperature of the water.

NRC Question 6.10

Clarify if the cask transfer facility (CTF) thermal analysis includes the effect of wedge assemblies in the convective heat transfer. Page 4.4-14 of the Final Safety Analysis Report (FSAR) states that with the CTF wedge assemblies in place between the loaded overpack and the CTF walls, there is still some convective heat transfer through the overpack, albeit not at a rate commensurate with the conditions on the ISFSI pad. Since this hardware affects the convective air flow, its effect should be included in the calculation of peak cladding temperature for the CTF configuration.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

PG&E Response to Question 6.10

The HI-STORM System in the CTF is modeled without the CTF wedge assemblies. The reason for ignoring all the four CTF wedge assemblies is that the area occupied by them is less than 3 percent of the annulus flow area between the HI-STORM and CTF wall. This will have no significant impact on the peak fuel and other cask component temperatures. It is also to be noted that, the thermal analysis of HI-STORM in CTF configuration is conservative since it assumes that the CTF walls are thermally insulated.

However, to provide an additional assurance to the staff, a steady state evaluation was performed to include all the four CTF wedges in the annulus region (Refer to Section B.5.5 of HI-2104625, Revision 6). All the wedge assemblies are explicitly modeled and no other changes to the existing thermal model for this configuration were made. The CTF walls are thermally insulated thereby ignoring any heat

transfer through conduction from the wedge assemblies to the CTF walls. The principal temperature results of the analysis are presented below:

Table 6-10.1: EFFECT OF CTF WEDGE ASSMEBLIES ON HI-STORM 100SA NORMAL STORAGE MPC AND OVERPACK TEMPERATURES IN THE CASK TRANSFER FACILITY (CTF)

Component	No CTF Wedge Assemblies Temperature °C	With CTF Wedge Assemblies Temperature °C
Fuel Cladding	391.9	392.6
MPC Basket	388.6	389.2
MPC Shell	221.6	223.0
Overpack Inner Shell	147.3	149.8

The effect of including the CTF wedge assemblies in the thermal model is small and therefore, it is safe to ignore them in the analysis.

NRC Question 6.11

Explain the reasons why during the 100 percent vent blockage the peak cladding temperature is the same for either the thermo-siphon enabled or thermo-siphon suppressed solutions.

Page 8.2.58 of the FSAR states that both the thermosiphon-enabled solution and the thermosiphon-suppressed solution compute approximately the same peak cladding temperature.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

PG&E Response to Question 6.11

The decay heat load approved in Revision 0 of the Holtec International HI-STORM 100 FSAR is 22.25 kW. The evaluations in Revision 0 were performed with thermosiphon being suppressed inside the MPC. However, incorporation of the thermosiphon internal convection inside the MPC enables the maximum decay heat load to rise to 28.74 kW in Revision 1 of the Holtec International HI-STORM 100 FSAR with the peak cladding temperatures being approximately the same as

thermosiphon suppressed solution. The statement in the Diablo Canyon ISFSI UFSAR reflects that the peak cladding temperature for thermosiphon enabled solution with a decay heat of 28.74 kW is approximately the same for the thermosiphon suppressed solution with a decay heat of 22.25 kW.

Excerpts from the Diablo Canyon ISFSI UFSAR on the 100 percent vent blockage are mentioned below that explains the procedure adopted for 100 percent vent blockage condition:

> *"Incorporation of the MPC thermosiphon internal convection phenomenon, as described in Chapter 4 of the HI-STORM 100 System FSAR enables the maximum, design-basis, PWR-decay-heat load to rise to about 29 kW. The thermosiphon effect also shifts the highest temperatures in the MPC enclosure vessel toward the top of the MPC. The peak, MPC lid, outer-surface temperature, for example, is computed to be about 450°F in the thermosiphon-enabled solution compared with about 210°F in the thermosiphon-suppressed solution, with both solutions computing approximately the same peak cladding temperature. In the 100 percent, inlet-duct-blockage condition, the heated MPC lid and MPC shell become effective heat dissipaters because of their proximity to the overpack outlet ducts and because the thermal radiation heat transfer rises at the fourth power of absolute temperature. As a result of this increased heat rejection from the upper region of the MPC, the time limits for reaching the short-term peak fuel-cladding temperature limits calculated without thermosiphon (72 hours)remains bounding."*

These statements will no longer have importance once the proposed license amendment is approved since the transient thermal evaluations in the current submittal are explicitly performed on a 3-D thermal model of HI-STORM 100SA System.

In implementing the approved Diablo Canyon ISFSI license amendment, PG&E administrative controls require that the DCPP and Diablo Canyon ISFSI design and licensing bases documents are updated to reflect these changes.

NRC Question 6.12

PG&E should remove all references to vacuum drying in the FSAR, since the MPC is dried using the forced helium dehydration system only. The FSAR contains statements in several places which may imply that vacuum could also be used to dry the MPC.

This is requested to provide accuracy, clarity, and consistency within the FSAR.

PG&E Response to Question 6.12

PG&E concurs. In implementing the approved DC ISFSI license amendment, PG&E administrative controls require that the DCPP and Diablo Canyon ISFSI design and licensing bases documents are updated to reflect these changes.

NRC Question 6.13

Please refer to Holtec Report HI-2104625. It is not clear if computational fluid dynamics (CFD) best practice guidelines (BPG) were used to perform the thermal evaluation of the HI-STORM 100 in the CTF configuration for design basis heat load and ambient conditions and to obtain the discretization error. It is not clear that the thermal analysis results provided in Table B.5.9 of Holtec Report HI-2104625 include adequate margins.

In order to facilitate the review, the analysis results should include an estimate of the numerical uncertainty, grid convergence, and sensitivity of the performed CFD analyses. To assist in the technical review, please provide an estimate of the numerical uncertainty and provide a response to the following questions:

a) Has a sensitivity analysis been performed concerning turbulence modeling, boundary conditions, grid independence and grid convergence?

b) Was grid convergence index (GCI) used to assess uncertainty of the predicted results?

Provide results such as percentage of the calculation discretization error and analysis files used to obtain the GCI. The applicant may consult the following documents for further information on CFD BPG: (1) Best Practice Guidelines for the use of CFD in Nuclear Reactor Safety Applications, NEA/CSNI/R(2007)5, (ADAMS accession number ML071581053); and (2) Policy of Journal of Fluid Engineering of ASME about CFD analyses.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

PG&E Response to Question 6.13

PG&E previously provided a response to this question in Reference 2 (i.e., LAR Supplement 1). The following response has been updated to include reference to the latest revision of the Holtec thermal report where in context.

In accordance with the above referenced ASME Journal procedure a grid sensitivity study had been performed for the normal long term storage condition (i.e.,

HI-STORM internal) and reported in Subsection B.5.1.2 of Holtec Report HI-2104625, Revision 3. The sensitivity calculations addressed the principal mechanism of heat dissipation, namely annulus cooling of the MPC. As required by the ASME Journal procedure, grid-refined thermal solutions are computed and results are post-processed to obtain the Grid Convergence Index (GCI) and added to Subsection B.5.1.2 of Holtec Report HI-2104625, Revision 3. For this purpose the following discrete sensitivity analyses were performed using the peak cladding temperature as the telltale output to infer convergence:

a. The number of radial cells in HI-STORM annulus is increased until convergence is established.
b. The mesh density in the axial direction is increased in steps to identify the converged configuration.

The principal result that is used to determine the converged mesh is the Peak Cladding Temperature (PCT). The grid sensitivity studies focus on the PCT computed by the FLUENT model of the HI-STORM system as the target output for checking convergence.

The converged mesh used for the long-term storage condition (as discussed in Subsection B.5.1.2 of Holtec Report HI-2104625, Revision 3) was therefore used for the thermal analysis of HI-STORM in the Cask Transfer Facility (CTF) and results had been reported in Table B.5.9 of Holtec Report HI-2104625, Revision 3. The airflow outside the HI-STORM system was modeled as turbulent flow using the k-ω turbulence model to incorporate the effect of air turbulence on the systems thermal performance. The adequacy of the k-ω model was addressed in the licensing of the Holtec International HI-STORM 100 System in CoC Amendment No. 5 in Docket No. 72-1014 and NRC acceptance obtained. For the k-ω turbulence model, y^+ should be less than 4 or 5 to ensure an adequate level of mesh to resolve the viscosity affected region near the wall (FLUENT Manual, Version 6.3.26). The mesh between the CTF and HI-STORM was constructed with a y^+ less than 2 towards HI-STORM external surface.

However, to provide an additional assurance to the staff, a grid sensitivity study has been performed to evaluate the condition of a HI-STORM 100SA system placed in the CTF. Since the airflow between the CTF and HI-STORM system is critical to the thermal performance of the system, the mesh in this region is modified. Two additional meshes are constructed – one coarser and another mesh finer than that presented in Table B.5.9 of Holtec Report HI-2104625, Revision 3. A brief summary of the different sets of grids evaluated is provided in Section B.5.5 of the report beginning with Revision 4.

The results from the study are post-processed in accordance with the guidance in the ASME Journal procedure cited by the staff (specifically the Grid Convergence Index, GCI) and evaluated in the Holtec Report HI-2104625, Revision 4.

The thermal solution presented in Table B.5.9 of Holtec Report HI-2104625, Revision 4, is on a converged mesh. To provide further assurance of convergence, the sensitivity results are evaluated in accordance with the ASME Journal procedure for control of numerical accuracy. Towards this end the GCI, which is a measure of the solution uncertainty, is computed in Appendix F of this report. The GCI for the fine grid (i.e. 18 radial cells between the CTF and HI-STORM 100SA System) computes to be 0.4 percent which provides further assurance of grid convergence.

<u>*NRC Question 6.14*</u>

Verify that for all thermal calculations an adequate operating density is provided in the analysis models. When reviewing some of the analysis files, the staff noticed that the operating density provided as input does not seem to be adequately calculated for the air side. The air operating density for the air side provided as input in the Fluent thermal models should correspond to the operating conditions of pressure and temperature.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

<u>PG&E Response to Question 6.14</u>

A reference density of 1.225 kg/m^3 was used as operating density parameter in FLUENT for all the thermal evaluations of the HI-STORM 100SA System. The FLUENT user manual indicates that the operating density should be representative of the volumetric average density of the fluid. The following excerpt is from Section 13.2.5 of the FLUENT 6.3 user's manual:

> *"By default,* ***FLUENT*** *will compute the operating density by averaging over all cells. In some cases, you may obtain better results if you explicitly specify the operating density instead of having the solver compute it for you ……Therefore, you should explicitly specify the operating density rather than use the computed average. The specified value should, however, be representative of the average value.*
>
> *In some cases the specification of an operating density will improve convergence behavior, rather than the actual results. For such cases use the approximate bulk density value as the operating density and be sure that the value you choose is appropriate for the characteristic temperature in the domain."*

However, a conference paper by Zigh and Solis (Reference 3) studies the effect of operating density in the analysis of dry storage casks. In accordance with the conference paper (Reference 3), it was validated that an appropriate operating density in the analysis of dry storage casks corresponds to the density evaluated at

the air inlet condition of pressure and temperature. The density of air based on air inlet conditions of pressure at 101325 Pa and temperature at 300K (80^{o}F) is 1.17 kg/m^3. Even though the reference density used in the evaluations is within 5 percent of the value of density based on air inlet conditions, a sensitivity study is performed for normal storage conditions (Scenario 1) with an operating air density corresponding to air inlet conditions. The result of such an evaluation is presented below:

Table 6-14.1: HI-STORM 100SA NORMAL STORAGE MPC AND OVERPACK TEMPERATURES FOR DIFFERENT OPERATING DENSITIES

Component	**Temperature for Scenario 1 (Reference Density) ^{o}C (^{o}F)**	**Temperature for Scenario 1 (Density Based on Air Inlet Condition) ^{o}C (^{o}F)**	**Limit ^{o}C (^{o}F)**
Fuel Cladding	379 (714)	386 (727)	MBF: 570 (1058) HBF: 400 (752)
MPC Basket	376 (709)	383 (721)	385 (725)
MPC Shell	214 (417)	217 (423)	232 (450)
Overpack Inner Shell	137 (279)	143 (289)	177 (350)
Overpack Body Concrete	92 (198)	95 (203)	149 (300)
Average Air Outlet	83 (181)	92 (198)	-
Pressure (in psig)			
MPC Cavity Pressure	67.0	68.0	100.0

It can be concluded from the above table that the predicted fuel cladding, MPC and overpack cask component temperatures changes by a small amount due to the variation in the air operating density used in the analyses and all components still remain well below their respective temperature limits. The safety of the system is not challenged since all the cask component temperatures and MPC cavity pressure remain well below their respective limits. Moreover, the fuel cladding and cask component temperatures for Scenario 2 (see Table B.5.13 of Holtec Report HI-2104625, Revision 6) still bound the corresponding temperatures for Scenario 1 with the new operating density. Therefore, the thermal evaluations of all the accident

and off-normal conditions (except the HI-STORM in CTF configuration which is evaluated below) remain conservative.

A similar evaluation is performed for the HI-STORM in the CTF (see Table B.5.17 of Holtec Report HI-2104625 Revision 6) to study the effect of operating air density on temperatures since it has least margin to temperature limits. The results are tabulated below for reference density of 1.225 kg/m^3 and density based on inlet air conditions (1.17 kg/m^3):

Table 6-14.2: HI-STORM 100SA MPC AND OVERPACK TEMPERATURES IN THE CTF FOR DIFFERENT OPERATING DENSITIES

Component	Temperature for Scenario 1 (Reference Density) ^oC (^oF)	Temperature for Scenario 1 (Density Based on Air Inlet Condition) ^oC (^oF)	Limit ^oC (^oF)
Fuel Cladding	392 (738)	389 (732)	MBF: 570 (1058) HBF: 400 (752)
MPC Basket	389 (732)	386 (727)	510 (950)
MPC Shell	222 (432)	221 (430)	413 (775)
Overpack Inner Shell	147 (297)	146 (295)	427 (800)
Overpack Body Concrete	99 (210)	97 (207)	177 (350)
Average Air Outlet	97 (207)	97 (207)	-
Pressure (in psig)			
MPC Cavity Pressure	68.7	68.3	100.0

It can be noted from the table above that the fuel cladding and cask component temperatures decrease with the change in operating density now based on inlet air conditions. The temperatures and pressures essentially remain unchanged for the HI-STORM in the CTF configuration.

The sensitivity studies show that small variations in the operating density has a relatively small effect on the predicted temperatures and pressures and in these particular thermal evaluations, all the safety conclusions remain unchanged.

<u>*NRC Question 6.15*</u>

Verify that all the thermal models used in the thermal evaluation include adequate insolation values. When reviewing some of the analysis files, the staff noticed that some surfaces that are exposed to solar heating did not include a heat source, as a result of insolation.

This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

<u>PG&E Response to Question 6.15</u>

It has been verified that all the thermal models used in the thermal evaluation of HI-STORM 100 System for all normal conditions include adequate solar insolation on the external cask surfaces.

Thermal analyses of off-normal conditions also include the solar insolation. However, no solar insolation is applied to a part of HI-STORM external cask surface when it is placed inside the cask transfer facility (CTF). The CTF will restrict the solar insolation onto that part of cask surface that is inside the CTF. Solar insolation is applied to the part of the cask external surface which is outside the CTF.

All the accident evaluations except the hypothetical fire accident include solar insolation values. This methodology is consistent with evaluations in HI-STORM FW System (Docket No 72-1032) that has been previously approved by USNRC.

<u>*NRC Question 8.1*</u>

Provide additional information regarding the type and properties for nonfuel hardware such as Neutron Source Assemblies (NSA), Instrument Tube Tie Rods (ITTR) and components of these devices such as individual rods in terms of galvanic/corrosive reactions. The staff recognizes that the ITTRs were approved for storage as non fuel hardware in the HI-STORM 100 Cask System in Amendment No. 6. However, that was for use for general certificate users. The licensee needs to provide information showing the applicability and acceptability for use at the DC site specific ISFSI. Additionally, the staff is concerned with compatibility between component materials and canister interior components, and needs additional information to ensure technical acceptability.

This information is needed to determine compliance with 10 CFR 72.120(d)

<u>PG&E Response to Question 8.1</u>

The following was excerpted from the DCPP UFSAR, Revision 19, Section 4.2.3.2.1.3 mechanical design description on neutron source assemblies as reactor core components.

> "The DCPP Units 1 and 2 reactor cores each employ two primary source assemblies and two secondary source assemblies in the first core. Each primary source assembly contains one primary source rod and between zero and twenty-three burnable absorber rods. Each secondary source assembly contains a symmetrical grouping of four secondary source rods and between zero and twenty burnable absorber rods.
>
> The primary and secondary source rods both utilize slightly cold worked 304 stainless steel (SS) material. The secondary source rods contain about 500 grams of stacked antimony-beryllium pellets, and the rod is internally prepressurized to 650 psig. The primary source rods contain capsules of Californium source material and alumina spacer rods to position the source material within the cladding. The rods in each assembly are permanently fastened at the top end to a hold down assembly, which is identical to that of the burnable absorber assemblies.
>
> The other structural members are fabricated from Type 304 and 308 stainless steel except for the springs exposed to the reactor coolant. They are wound from an age hardened nickel base alloy for corrosion resistance and high strength."

The primary and secondary neutron source assemblies to be stored in the Diablo Canyon ISFSI are standard Westinghouse supplied components common to this reactor type. The materials of these assemblies and their construction are similar to nonfuel hardware currently authorized for storage in the Diablo Canyon ISFSI (i.e., rod cluster control assemblies (RCCAs), burnable poison rod assemblies (BPRAs) and thimble plug devices (TPDs)). The neutron source assemblies are designed for a reactor core service life cycle where design pressures, temperatures and reactor coolant system chemistry and galvanic reactions are more challenging to these components than service conditions during the few days of MPC closure and sealing operations and subsequent inert, interim storage conditions in the MPC at the Diablo Canyon ISFSI.

Instrument Tube Tie Rod (ITTR) modifications were performed on selected nuclear fuel assemblies at DCPP to resolve an industry fuel performance issue with potential for fuel assembly top nozzle separation that adversely affected the conventional handling and retrievability of the fuel assembly. All of the Instrument Tube Tie Rod

(ITTR) hardware used to repair selected nuclear fuel assemblies at DCPP was designed, fabricated, and field-installed by Westinghouse, the Original Equipment Manufacturer (OEM) of the repaired fuel. The ITTR hardware used at DCPP is similar if not identical to Westinghouse hardware implemented at other plants. Westinghouse designed the ITTR hardware to be compatible with service conditions in the spent fuel pool as the ITTR assemblies cannot be placed in the reactor core. The subcomponents comprising the ITTR are in contact only with the materials of the fuel assembly itself and not the MPC basket. Service conditions in the MPC during the few days of closure and sealing operations are comparable to the spent fuel pool environment and do not present any new potential for chemical or galvanic reactions of the repaired assemblies or the MPC prior to being sealed in the inerted atmosphere of the MPC for interim storage.

In summary, there are no Diablo Canyon ISFSI site-specific characteristics that adversely affect storage of neutron source or ITTR-repaired assemblies as presently authorized for general license users in Holtec International HI-STORM 100 CoC 1014.

NRC Question 8.2

Provide additional design and operation details concerning the referenced "keep full system" to be used as a supplemental cooling system (SCS) for high burnup fuel during transfer to the ISFSI site.

The licensee is requesting an exemption to 10 CFR 72.236(f) similarly to that requested by Holtec for use in CoC No. 1014 for its SCS. However, since the DC ISFSI is a site specific license, 10 CFR 72.236(f) is not applicable. However, the requirements of 10 CFR 72.128(a)(5) are applicable. It is difficult from the information supplied by the licensee to determine if this system acceptably meets this requirement. The additional heat load affects the materials evaluation of the canister components.

This information is needed to determine compliance with 10 CFR 72.128(a)(4).

PG&E Response to Question 8.2

The passive supplemental cooling system proposed by PG&E in accordance with the Holtec thermal analysis HI-2104625, Revision 6, Section C.5.7 is designed to use the transfer cask annulus filled with non-contaminated, demineralized water at atmospheric pressure to facilitate increased heat transfer from the MPC to the environment. PG&E is not requesting any change to the presently authorized maximum heat load of 28.74 kW at an operating pressure of 5 atmospheres absolute in this license amendment. As is discussed in the thermal report Section C.5.7, using the SCS for high-burnup fuel (HBF) results in a 220 degree F reduction

in MPC outer surface temperatures thereby keeping the HBF fuel cladding within limits.

The transfer cask is described in the DC ISFSI UFSAR, Revision 3, Section 4.3.2.3.4 and is designed for the service temperatures of the annulus water (approx. 232 degrees F). Components and connections to the transfer cask annulus drain port (ref. DC ISFSI UFSAR Figure 4.2-8 Page 2 Detail A) for the purpose of the draining the annulus water are designed to accommodate this temperature and the hydrostatic pressures achieved in draining the annulus.

The makeup tank is designed of stainless material with two top-mounted, fill openings and one bottom mounted drain port for connection to the transfer cask annulus drain port. The make-up tank is approximately 13-inches square in radial cross section and shaped to conform to the outer radius of the transfer cask top lid flange along a circumference of approx. 64 inches. The tank is seated on the transfer cask water jacket top plate and attaches to three transfer cask top lid bolts. The capacity of the tank when in service is approx. 35 gallons.

To place the system in service, the empty make-up tank is placed atop the transfer cask water jacket top plate and its drain port closed. The transfer cask annulus is first filled with approx. 80 gallons of non-contaminated, demineralized water from the same plant supply used for the transfer cask overpressure system during early preparations for MPC loading operations. For unloading operations, the annulus and make-up water source is supplied from a pre-staged, designated storage tank at the cask transfer facility. Once the annulus and make-up tank are filled, the line is connected first to the tank, purged of any air and then connected to the transfer cask annulus drain port. Next, the branch line to the level indicating standpipe is connected valved in and purged. The valve in the line to the transfer cask annulus is slowly opened and the two levels are equalized. The make-up tank is topped off with additional water until the annulus is full.

For cask loading operations, the annulus is drained into a designated container at the cask transfer facility for sampling and processing in accordance with DCPP Radioactive Effluent Management Program requirements prior to MPC downloading operations. For cask unloading operations, the annulus would be drained in the DCPP Fuel Handling/Auxiliary Building in accordance with radiation protection administrative controls.

Use of the SCS is administratively controlled in the dry fuel storage loading and unloading procedures.

<u>*NRC Question 9.1*</u>

Clarify the information concerning the details of helium leak tests discussed in the DC ISFSI FSAR, Section 4.2.3.3.6, "Confinement Design."

The staff specifically requests additional information on testing of the base material, including the MPC shell, baseplate, lid, port covers, etc., performed by Holtec and PG&E to ensure confinement integrity over the life of the HI-STORM 100 MPC.

This information is needed to determine compliance with 10 CFR 72.122(h) and 10 CFR 72.126(d).

<u>PG&E Response to Question 9.1</u>

PG&E is proposing no changes to the fabrication, closure or testing of the Multi-Purpose Canisters (MPCs) under this LAR. The welding and testing of the MPCs is as described in the DC ISFSI UFSAR, Revision 3, Section 4.2.3.3.6 (excerpt provided below). Overall MPC construction meets SFST-ISG-18 criteria. The fabricated MPC body (shell, baseplate and fuel basket) is subjected to a helium leak test in the shop prior to its delivery to the site. The base metal of the lid and baseplate are tested for flaws using UT and PT methods to assure integrity. For MPC closure, only the port cover plates are subject to a post-weld helium leak test under SFST-ISG-18. Thus, as-completed MPCs meet the leaktight criteria of SFST-ISG-18.

Excerpt from ISFSI FSAR 4.2.3.3.6:

> The MPC is a totally welded pressure vessel designed to meet the stress criteria of ASME Section III, Subsection NB. No bolts or fasteners are used for closure. All factory welds are examined per ASME Section III and helium leak tested to ensure conformance to the offsite dose analysis. All closure welds are examined using the liquid-penetrant method. Two penetrations are provided in the MPC lid for draining, drying, and backfilling during loading operations. Following loading operations, vent and drain port cover plates are welded to the MPC lid and helium leak tested to ensure their integrity. A closure ring, which covers the penetration cover plates and welds, is welded to the MPC lid to provide redundant closure of the MPC vessel. The loading and welding operations are performed inside the DCPP FHB/AB. There are no confinement boundary penetrations required for MPC monitoring or maintenance during storage.

Holtec International Report HI-2104625, "Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System Design," Revision 6 – Proprietary Version, with one disc of proprietary data files on optical storage medium (OSM) DVD-ROM

Holtec Affidavit for Holtec International Report HI-2104625, "Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System Design," Revision 6 – Proprietary Version, with one disc of proprietary data files on optical storage medium (OSM) DVD-ROM

Holtec International Report HI-2104625, "Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System Design," Revision 6
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