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PG&E Letter DCL-05-015

U.S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Request for Additional Information Regarding License Amendment Request 04-07, "Revision to Technical Specifications 3.7.17 and 4.3 for Cycles 14-16 for a Cask Pit Spent Fuel Storage Rack"

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) submitted an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 by PG&E Letter DCL-04-149, dated November 3, 2004. License Amendment Request (LAR) 04-07, submitted for Nuclear Regulatory Commission (NRC) review and approval, proposed Technical Specification (TS) changes to allow installation and use of a temporary cask pit spent fuel storage rack for Units 1 and 2. The cask pit rack would allow the storage of an additional 154 spent fuel assemblies. The total spent fuel pool (SFP) storage capacity for each unit would be increased to 1478 fuel assemblies for Cycles 14-16.

The NRC staff has identified additional information required to complete their evaluation of the criticality analysis associated with LAR 04-07. Enclosed is PG&E's response to the request for additional information. As noted in the enclosure, during Cycles 14-16 when the temporary cask pit rack is being utilized, there will be two applicable criticality analyses for the SFP: (1) the existing criticality analysis for the permanent racks; and (2) the new criticality analysis for the temporary cask pit rack. Following removal of the temporary cask pit rack during Cycle 16, the criticality analysis for the temporary cask pit rack will no longer be applicable (see proposed TS Bases B 3.7.16).

The enclosed additional information does not affect the results of the safety evaluation and no significant hazards determination previously transmitted in PG&E Letter DCL-04-149, "License Amendment Request 04-07, Revision to Technical Specifications 3.7.17 and 4.3 for Cycles 14-16 for a Cask Pit Spent Fuel Storage Rack."

A001

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

Sincerely,

Donna Jacobs
Vice President Nuclear Services

tlg/4160
Enclosure
cc: Edgar Bailey, DHS
Bruce S. Mallett
David L. Proulx
Diablo Distribution
cc/enc: Girija S. Shukla

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT

Donna Jacobs, of lawful age, first being duly sworn upon oath states 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 additional NRC questions regarding License Amendment Request 04-07, "Revision to Technical Specifications 3.7.17 and 4.3 for Cycles 14-16 for a Cask Pit Spent Fuel Storage Rack" 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 24th day of February 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 Regarding License Amendment Request 04-07, "Revision to Technical Specifications 3.7.17 and 4.3 for Cycles 14-16 for a Cask Pit Spent Fuel Storage Rack"

Question 1

In Attachment 5, it is concluded that the bias is independent of the enrichment. (Section 4A.2). However, Figure 4A.3 has very few points in the area near and over the 5 percent enrichment. One would anticipate that the uncertainty in the bias increases in that region. You do not discuss bias uncertainty (indicating that it does not vary with enrichment). Why not?

PG&E Response to Question 1

The bias uncertainty is calculated as a constant value due to its minimal effect on the final calculated maximum k-eff. When calculating the combined uncertainties and tolerance effects, all individual components are statistically combined (see Section 4.5 and Table 4.1.1 of Holtec Report HI-2043162, submitted in PG&E Letter DCL-04-149, dated November 3, 2004). The combined uncertainty is substantially larger than the bias uncertainty itself. This means that variations in the bias uncertainty would have a negligible effect on the combined uncertainty, and therefore a negligible effect on the final k-eff value. Determining the bias uncertainty as a constant value is therefore considered appropriate. Note that the bias and bias uncertainty approach outlined in Holtec Report HI-2043162 has been Holtec's standard approach for many years.

Additionally, as discussed in PG&E's response to Question 2 below, the proposed Technical Specification (TS) Figure 3.7.17-4 for the Cask Pit Rack specifies a maximum initial enrichment of 4.1 percent for fuel that may be stored in the rack. Therefore, Figure 4A.3 contains a sufficient number of data points in the range of interest to support the conclusion that the bias is independent of the enrichment.

Question 2

The proposed TS 3.7.17 Figures 3.7.17-2 and 3.7.17-3 indicate enrichment up to 5 percent. The accident analyses indicate that the calculations were performed for the pool populated by assemblies of lower enrichment. You request that the new calculations become the calculations of record. If in the future the plant operates with enrichments at or higher than 5 percent, will these calculations still be valid?

PG&E Response to Question 2

As part of its conservative approach to licensing the temporary cask pit storage rack, PG&E decided to only allow storage of relatively low enrichment, high burnup spent fuel in the temporary cask pit rack. TS 3.7.17, Figures 3.7.17-2 and 3.7.17-3, specify minimum required burnup as a function of initial enrichment for storing fuel in the

permanent storage racks. The only revisions to these TS figures were to add the term permanent storage rack in the title of the these figures to distinguish between these figures for the existing permanent racks and Figure 3.7.17-4, which was added to specify minimum required discharge burnup as a function of initial enrichment for the temporary cask pit rack. Figures 3.7.17-2 and 3.7.17-3 allow storage of fuel up to 5 percent enrichment in the permanent storage racks, which is the analytical limit of the existing spent fuel pool (SFP) criticality analysis (see TS Bases B3.7.16 references 3 and 5). Figure 3.7.17-4 for the temporary cask pit rack specifies a maximum initial enrichment of 4.1 percent. License Amendment Request (LAR) 04-07, Section 4.6.3, states that the criticality calculations for the temporary cask pit rack normal conditions were performed for spent fuel assemblies with less than or equal to 4.1 percent and greater than or equal to 28.53 GWD/MTU.

With the exception of thermal analyses, all of the analyses performed for LAR 04-07 are only intended to be applicable during Cycles 14-16 when the cask pit rack is installed in the SFP. LAR 04-07, Section 4.3.5, indicates that PG&E updated the SFP thermal-hydraulic analyses as part of the cask pit rack project and requested that the NRC approve the updated thermal-hydraulic analysis as the licensing basis of record for future spent fuel storage requirements, including the temporary supplemental spent fuel storage capacity provided by the cask pit rack.

In summary, during Cycles 14-16 when the temporary cask pit rack is being utilized, there will be two applicable criticality analyses for the SFP: (1) the existing criticality analysis for the permanent racks; and (2) the new criticality analysis for the temporary cask pit rack. Following removal of the temporary cask pit rack during Cycle 16, the criticality analysis for the temporary cask pit rack will no longer be applicable (see proposed TS Bases B 3.7.16). The existing permanent storage rack criticality calculations analyzed enrichments up to 5 percent and fuel with up to 5 percent enrichment may be stored in the permanent storage racks in accordance with existing TS Figures 3.7.17-2 and 3.7.17-3. Should PG&E desire to store fuel with enrichments higher than 5 percent, new criticality analyses for the permanent storage racks would be required. A new LAR would be required for the associated TS changes.

Question 3

In Section 4.6.5, what is the basis for the 5 percent depletion uncertainty?

PG&E Response to Question 3

The 5 percent depletion uncertainty used for the criticality analyses is established to bound the accuracy of the data from the INCORE computer code used to calculate the nuclear fuel burnup for each individual fuel assembly. Each new cycle of operation at Diablo Canyon Power Plant requires the development and verification of a new reload core design that typically consists of approximately 40 to 45 percent new fuel assemblies and 55 to 60 percent previously burned fuel assemblies. As part of

ensuring that each new core reload meets all of the applicable nuclear design requirements, particularly the core power distribution limits, a detailed calculation of the nuclear fuel burnup or depletion for each previously burned fuel assembly is maintained.

This fuel assembly burnup is calculated using input from the Westinghouse INCORE computer code and is based on flux map data obtained using the Movable Incore Detector System. These flux maps are nominally performed every month of full power operation and are required by TS to be performed at least once every 31 Effective Full Power Days. In addition to verifying that the measured core power distribution is within the TS peaking factor limits, the flux map data is also used by the INCORE code to determine a corresponding fuel burnup rate for each fuel assembly.

An overall uncertainty in the calculated burnup or depletion for a given fuel assembly may then be conservatively estimated by evaluating the individual terms and their associated uncertainties, which are used as inputs. The fuel burnup rate as determined by the INCORE computer code is a function of the fuel assembly average power, which is calculated using the fuel assembly FNΔH peaking factor (nuclear enthalpy rise hot channel factor) and the total core thermal power. A fuel assembly FNΔH calculation uncertainty of 4 percent has been established to bound the nuclear measurement uncertainty of approximately 3.8 percent, which is applied to the corresponding plant FNΔH surveillance limit in the plant Core Operating Limits Report. The assumption of a 2 percent uncertainty on the total core thermal power is used in the plant safety analysis to conservatively bound the actual measurement uncertainties. Therefore, the overall statistical uncertainty associated with the calculated fuel depletion or burnup is determined to be about 4.5 percent based on using the square root of the sum of the squares methodology for combining these independent uncertainty components.

In conclusion, the 5 percent depletion uncertainty assumed in the spent fuel criticality analysis is based on a value that bounds the calculational uncertainty for determining the individual fuel assembly burnup.