NRR-PMDAPEm Resource

From: Singal, Balwant
Sent: Friday, December 09, 2016 1:23 PM
To: 'Richardson, Michael'
Cc: DeVaughn, Elijah
Subject: Request for Additional Information (RAI) - Diablo Canyon Power Plant License Amendment Request for Adoption of NEI 94-01 (CAC Nos. MF7731 and MF7732)
Attachments: MF7731-RAI-APLA.docx

By letter dated May 12, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16146A100), Pacific Gas and Electric Company (PG&E, the licensee) submitted an application to adopt NEI 94-01, Revision 2A, "Industry Guidance for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008 (ADAMS Accession No. ML100620847). The U.S. Nuclear Regulatory Commission (NRC) staff needs the additional information described in the Attachment to complete its review.

The Draft RAI request was issued on November 23, 2016. A clarification was held on December 8, 2016. You are requested to respond to these RAIs no later than January 31, 2017. Please treat these RAIs as formal transmittal of RAIs.

Thanks.

Hearing Identifier: NRR_PMDA
Email Number: 3206

Mail Envelope Properties (f5e16280af424530b02ce0f151015758)

Subject: Request for Additional Information (RAI) - Diablo Canyon Power Plant License Amendment Request for Adoption of NEI 94-01 (CAC Nos. MF7731 and MF7732)
Sent Date: 12/9/2016 1:23:11 PM
Received Date: 12/9/2016 1:23:00 PM
From: Singal, Balwant

Created By: Balwant.Singal@nrc.gov

Recipients:
"DeVaughn, Elijah" <E1D7@pge.com>
Tracking Status: None
"'Richardson, Michael'" <MJRm@pge.com>
Tracking Status: None

Post Office: HQPWMSMRS05.nrc.gov

Files	**Size**		**Date & Time**
MESSAGE	752		12/9/2016 1:23:00 PM
MF7731-RAI-APLA.docx		38950	

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received: ZZZ

<u>Request for Additional Information</u>

<u>Diablo Canyon Power Plant, Units 1 and 2 (DCPP)</u>

<u>Application to Adopt NEI 94-01, Revision 2A, "Industry Guidance for Implementing</u>

<u>Performance-Based Option of Title 10 of the *Code of Federal Regulations*</u>

<u>(10 CFR) Part 50, Appendix J</u>

By letter dated May 12, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16146A100), Pacific Gas and Electric Company (PG&E, the licensee) submitted an application to adopt NEI 94-01, Revision 2A, "Industry Guidance for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008 (ADAMS Accession No. ML100620847). The U.S. Nuclear Regulatory Commission (NRC) staff needs the following additional information to complete its review.

Request for Additional Information (RAI)

APLA RAI 01

Regulatory Guide (RG) 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," May 2011 (ADAMS Accession No. ML100910006) states, "When the calculated increase in LERF [large early release frequency] is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year (Region II)."

Section 5.3.1 in Attachment 3 of letter dated May 12, 2016 (the LAR) provides a total change in LERF of 7.87E-7/year (yr) for DCPP, Unit 1 and 8.09E-7/yr for DCPP, Unit 2, which is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year as stated in RG 1.174. Therefore, the total LERF for each unit must be reasonably shown to be less than 10^{-5} per reactor year. The total LERF for Unit 1 and Unit 2 is calculated from Table 5-2, page 28, and page 29 of Attachment 3 of the LAR as follows:

- Unit 1 total LERF = $LERF_{internal}$ + $LERF_{fire}$ + $LERF_{seismic}$ + ΔLERF
  Unit 1 total LERF = 2.26E-6/yr + 2.45E-6/yr + 3.29E-6/yr + 7.87E-7/yr = 8.78E-6/yr

- Unit 2 total LERF = $LERF_{internal}$ + $LERF_{fire}$ + $LERF_{seismic}$ + ΔLERF
  Unit 2 total LERF = 2.18E-6/yr + 2.17E-6/yr + 3.29E-6/yr + 8.09E-7/yr = 8.45E-6/yr

Section 4.2.7 of Electric Power Research Institute (EPRI) Technical Report (TR) 1009325, Revision 2-A, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," (ADAMS Accession No. ML14024A045) states that "[w]here possible, the analysis should include a quantitative assessment of the contribution of external events (for example, fire and seismic) in the risk impact assessment for extended ILRT [integrated leak rate testing] intervals." Section 5.3.1 of Attachment 3 of the LAR states, "The DCPP IPEEE [Individual Plant Examination of External Events] determined that each of the 'other' external events evaluated

contributed less than 1.0E-06 per year to core damage and was screened out as a result [Reference 32[1]]. Therefore, the 'other' external events are also screened for this application."

If "other" external events are added to the equations to the LERF calculations above, contributions of less than 1.0E-6, depending upon how much less, may cause the total LERF to exceed the 10^{-5} threshold:

a. Due to the total LERF of DCPP, Unit 1 and DCPP, Unit 2 being close to the 10^{-5} threshold, please justify why "other" external events were screened for this application.

b. Please demonstrate that excluding "other" external events from the DCPP IPEEE will not cause total LERF for DCPP, Unit 1 and DCPP, Unit 2 to exceed the 10^{-5} threshold.

APLA RAI 02

Section 5.3.1 on page 30 of Attachment 3 of the LAR states, "Although the total change in LERF is somewhat close to the RG 1.174 limit [Reference 4[2]] when external event risk is included, several conservative assumptions were made in this ILRT analysis, as discussed in Sections 4.0, 5.1.3, 5.2.1, and 5.2.4;; therefore the total change in LERF is considered conservative for this application."

With the total LERF being so close to the 10^{-5} threshold, please confirm that the "conservative assumptions" made in this ILRT analysis maintain delta LERF (ΔLERF) within Region II of RG 1.174. When citing conservatism in the base PRA model, please confirm that calculation of the differential risk for the application is also conservative (i.e., the risk estimated for the before versus after condition uses the same assumptions, etc., except for the change to any basic event values affected by the application, ensuring that the before value is not overestimated such that subtracting it from the after value could underestimate the risk increase).

APLA RAI 03

In Tables 6-1 and 6-2 of Attachment 3 of the LAR, the calculated change in dose rates, when summed for Classes 3a and 3b, due to the ILRT extension are higher than what is recorded as changes in the total dose rate in the tables (i.e., the rows for "delta total dose rate") and reported in Section 3.3.3 on page 18 of the enclosure to the LAR, and in Section 7.0 of Attachment 3 of the LAR.

<u>Table 6-1</u>

The recorded value for "Base Case" to "Extend to 1 in 10 Years" is 4.44E-2 person-rem/yr. The total change in dose rate from the "Base Case" to "Extend to 1 in 10 Years" is calculated as follows using the Class-specific values for 3a and 3b:

$$\Delta \text{Dose Rate} = (\text{DoseRate}_{1in10yrs3a} + \text{DoseRate}_{1in10yrs3b}) - (\text{DoseRate}_{BaseCase3a} + \text{DoseRate}_{BaseCase3b})$$
$$\Delta \text{Dose Rate} = (4.69E\text{-}2 + 1.88E\text{-}2) - (1.41E\text{-}2 + 5.65E\text{-}3) = 4.59E\text{-}2 \text{ person-rem/yr}$$

The recorded value for "Base Case" to "Extend to 1 in 15 Years" is 7.60E-2 person-rem/yr. The total change in dose rate from the "Base Case" to "Extend to 1 in 15 Years" is calculated as follows:

---

[1] Calculation C.10, Revision 5, Diablo Canyon Power Plant, "DCPP PRA [Probabilistic Risk Assessment] Model Technical Adequacy.

[2] RG 1.174, Revision 2

ΔDose Rate = (DoseRate$_{1in15yrs3a}$ + DoseRate$_{1in15yrs3b}$) – (DoseRate$_{BaseCase3a}$ + DoseRate$_{BaseCase3b}$)
ΔDose Rate = (7.03E-2 + 2.83E-2) – (1.41E-2 + 5.65E-3) = 7.88E-2 person-rem/yr

The recorded value for “Extend to 1 in 10 years” to “Extend to 1 in 15 Years” is 3.17E-2 person-rem/yr. The total change in dose rate from the “Extend to 1 in 10 Years” to “Extend to 1 in 15 Years” is calculated as follows:

ΔDose Rate = (DoseRate$_{1in15yrs3a}$ + DoseRate$_{1in15yrs3b}$) – (DoseRate$_{1in10yrs3a}$ + DoseRate$_{1in10yrs3b}$)
ΔDose Rate = (7.03E-2 + 2.83E-2) – (4.69E-2 + 1.88E-2) = 3.29E-2 person-rem/yr

Table 6-2
The recorded value for “Base Case” to “Extend to 1 in 10 Years” is 3.90E-2 person-rem/yr. The total change in dose rate from the “Base Case” to “Extend to 1 in 10 Years” is calculated using the same method as Table 6-1, and is 4.05E-2 person-rem/yr.

The recorded value for “Base Case” to “Extend to 1 in 15 Years” is 6.69E-2 person-rem/yr. The total change in dose rate from the “Base Case” to “Extend to 1 in 15 Years” is calculated using the same method as Table 6-1, and is 6.94E-2 person-rem/yr.

The recorded value for “Extend to 1 in 10 years” to “Extend to 1 in 15 Years” is 2.79E-2 person-rem/yr. The total change in dose rate from the “Extend to 1 in 10 Years” to “Extend to 1 in 15 Years” is calculated using the same method as Table 6-1, and is 2.89E-2 person-rem/yr.

Based on the discussion above:

a. Please explain the discrepancy from the calculated values of change in dose rate from the recorded values of change in dose rate in Table 6-1 and Table 6-2, and the reported value of change in dose rate mentioned in Section 3.3.3 on page 18 of the enclosure to the LAR, and in Section 7.0 of Attachment 3 of the LAR.

b. If the discrepancy is a calculational error, please correct the error.

APLA RAI 04

According to Regulatory Issue Summary (RIS) 2007-06, “Regulatory Guide 1.200 Implementation,” (ADAMS Accession No. ML070650428), the NRC staff expects that licensees fully address all scope elements with Revision 2 of RG 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” March 2009 (ADAMS Accession No. ML090410014) by the end of its implementation period (i.e., one year after the issuance of Revision 2 of RG 1.200). Revision 2 of RG 1.200 endorses, with exceptions and clarifications, the combined American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard (ASME/ANS RA-Sa-2009)[3].

Given the requirement listed above:

[3] ASME/ANS RA-Sa-2009,” Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power plant Applications.”

a. Please confirm that the peer review of the Fire PRA, cited on page 38 of Attachment 3 of the LAR meets the requirements of RG 1.200, Revision 2. If the requirements are not met, please identify any gaps between the peer review and the requirements in RG 1.200, Revision 2 and assess any effect upon the conclusions for this application. Also, confirm for the disposition of SR IE-A5 in Table A-1 of Attachment 3 of the LAR, that the system review includes, as needed, the possibility of an initiating event caused by system failure down to the subsystem or train level as required by RG 1.200, Revision 2.

b. Please identify when the peer review of the Seismic PRA was performed, as cited on page 39 of Attachment 3 of the LAR. Please confirm that the peer review meets the requirements of RG 1.200 Revision 2. If the requirements are not met, please identify any gaps between the peer review and the requirements in RG. 1.200, Revision 2 and assess any effect upon the conclusions for this application.

c. RG 1.200, Revision 2 contains several enhancements that are related to Supporting Requirement (SR) HR-D3. In Table A-1 of Attachment 3 of the LAR, for SR HR-D3, please confirm that the quality of procedures, administrative controls, and human-machine interface reviews have been addressed in accordance with the guidance in RG 1.200, Revision 2.

d. Rev. 2 of RG 1.200 adds the following SR DA-D9 Capability Category (CAT) I, II, and III: "For each SSC [Structure, System, or Component] for which repair is to be modeled, ESTIMATE, based on the data collected in DA-C15, the probability of failure to repair the SSC in time to prevent core damage as a function of the accident sequence in which the SSC failure appears." Please confirm this SR was included in the peer review. If not, please address it in Table A-1 of Attachment 3 of the LAR.

APLA RAI 05

In Table A-1 of Attachment 3 of the LAR, for the disposition of SR HR-G7, identify the joint Human Error Probability (HEP) floors that were used. Please confirm that none were < 1E-6 for internal events. If any were <1E-6 for internal events, provide the basis as well as the results of a sensitivity evaluation using 1E-6, and include any effects on the "no impact" statement in the “Impact on ILRT Extension” column. For external events, please confirm that no floor < 1E-5 was used, or else perform a similar sensitivity evaluation.

Note that this RAI is applicable to SR QU-C2, SR HRA-C1, SR SPR-B2, and SR SPR-B9; the response for SR HR-G7 should be such as to cover these as well.

APLA RAI 06

In Table A-1 of Attachment 3 of the LAR, for the “Impact on ILRT Extension” column of SR LE-C2, the licensee states, “Treatment is conservative for overall LERF. While reduction in LERF could lead to an overall increase in Class 3b, the change is expected to be small, and there is sufficient margin for ∆LERF to the upper level of RG 1.174 Region II (see Section 5.2.4). Therefore the ILRT analysis is not adversely impacted.”

For the “Impact on ILRT Extension” column of SR LE-E2, the licensee also states, “Using a small containment isolation size is not conservative for ∆LERF, but is conservative for total LERF. While reduction in LERF could lead to an overall increase in Class 3b, the change is

expected to be small, and there is significant margin for ΔLERF to the upper level of RG 1.174 Region II (see Section 5.2.4.). There is no impact on the ILRT Extension Risk Analysis."

From the discussion in RAI 01, and the information provided in Attachment 3 of the LAR, the ΔLERF is 7.87E-7/yr for DCPP, Unit 1 and 8.09E-7/yr for DCPP, Unit 2 with external events (i.e., fire and seismic, and "other"). Since these values are between 10^{-7} and 10^{-6}, the total LERF must be less than 10^{-5}/yr to remain in Region II of RG 1.174. The total LERF is 8.78E-6/yr for Unit 1 and 8.45E-6/yr for Unit 2, with the "other" external events being excluded. There is a possibility the total LERF for DCPP, Unit 1 and DCPP, Unit 2 could be closer to 10^{-5}/yr when added to the calculation. Therefore, there may not be sufficient margin for ΔLERF to the upper level of RG 1.174 Region II when external events are added to the ΔLERF calculation. Based on this position:

a. Please address the potential for exceeding the Region I-Region II threshold in RG 1.174 for SR LE-E2 and SR LE-C2.

b. Please provide additional quantitative justification to conclude that there is no adverse impact from SR LE-E2 and SR LE-C2 on the ILRT analysis.

APLA RAI 07

The following RAIs involve taking credit for future occurrences. The implication is taking credit for future occurrences may result in fire risks that are under-estimated. Please provide additional information for the following:

a. In Table A-3 of Attachment 3 of the LAR, for the disposition of SR PRM-C1, the licensee states, "This F&O [Finding and Observation] has been resolved by a model update. The RCP [reactor coolant pump] seal model was modified based on the vendor guidance documents to include both human error and random failure modes." The ILRT results are therefore conditional upon the installation of the new RCP seals and the validity of the assumptions that credit the seals for the reduction in likelihood of an RCP Seal Loss-of-Coolant Accident (LOCA). Since this will be a future occurrence, please provide a sensitivity analysis that considers reduced credit for the RCP shutdown seals (i.e., reduce credit by a factor of 2). Please confirm that the results of this sensitivity analysis meet the risk acceptance criteria for this application.

b. In relation to SR FSS-D11, the installation of the incipient detection system is also a future action. New guidance on the credit taken for very early warning fire detection system (VEWFDS) is available in NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities, (DDELORES-VEWFIRE)" of which the pre-publication final version is available at ADAMS) Accession Nos. ML16286A000 and ML16286A002 (note that the accession numbers may change when the final version is published). The methodology in NUREG-2180 is acceptable to the NRC because it is currently the best available guidance. The guidance provided in frequently asked question (FAQ) 08-0046, Memo dated November 23, 2009, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0046 Incipient Fire Detection Systems" (ADAMS Accession No. ML093220426), has been retired and alternative approaches for staff evaluation are necessary to complete the safety evaluation.

If the Fire PRA (FPRA) credits the future installation of VEWFDS, please explain how credit (e.g., approach, methods, data, and assumptions) taken for the proposed VEWFDS is consistent with the guidance in NUREG-2180 or bounds the risk metrics in this that would be obtained had the guidance in NUREG-2180 been applied. If credit taken for VEWFDS in the FPRA is not consistent with or bounded by NUREG-2180, please provide:

1) The risk metrics that would be obtained had the guidance in NUREG-2180 been applied, or that would be obtained had an alternative method been used, along with a description and justification for the alternative method. Development and use of an alternative proposal may extend the time required to complete the review. The new risk results can be generated from a sensitivity study type evaluation insofar as formal incorporation of the new method into the PRA model of record is not required.

2) Please explain how the any increases in the risk metrics are consistent with the acceptance criteria for this application.

APLA RAI 08

In Table A-3 of Attachment 3 of the LAR, for the disposition SR IGN-A4, the licensee states, "This F&O has been resolved by a documentation update. The two DG [diesel generator] fires have been further reviewed and additional justification for not using the plant-specific experience has been documented." It appears this F&O was resolved prior to the 2010 peer review, and therefore was not confirmed as closed by the 2010 peer review. Based on this observation, please explain the justification for excluding the plant-specific experience.

APLA RAI 09

In Table A-4 of Attachment 3 of the LAR, for the "ILRT Disposition" columns of SR SPR-B1-01 and SR SPR-B9-01, the licensee states, "The seismic HRA [human reliability analysis] is relatively unimportant, and any increase in HEP values would not have a significant impact on seismic risk." From Table 5-1, Table 5-2, and Sections 5.1.2 and 5.3.1 of Attachment 3 of the LAR, the seismic contribution to CDF and LERF is calculated by the Fussell-Vesely Importance Measure:

$FV_{CDFseismic} = CDF_{seismic} / (CDF_{internal} + CDF_{fire} + CDF_{seismic})$

$FV_{CDFseismic} = 2.7E\text{-}5 / (1.8E\text{-}5 + 5.0E\text{-}5 + 2.7E\text{-}5) = 0.28$

$FV_{LERFseismic} = LERF_{seismic} / (LERF_{internal} + LERF_{fire} + LERF_{seismic})$

$FV_{LERFseismic} = 3.3E\text{-}6 / (2.2E\text{-}6 + 2.3E\text{-}6 + 3.3E\text{-}6) = 0.42$

Therefore, the NRC staff believes that the seismic contribution to LERF is significant. Based on the position that seismic LERF is a significant contributor to LERF risk, please provide quantitative justification that the seismic HRA is relatively unimportant by assessing its contribution to total LERF as well as ΔLERF.

APLA RAI 10

In Table A-4 of Attachment 3 of the LAR, for SR SPR-B1-03, the Finding/Observation for offsite power fragility states, "the value appears to underestimate the probability," and "the range appears to be under predicting the likelihood of a loss of offsite power." In response, the

licensee states, “the current SPRA 9seismic PRA] model provides a reasonable estimate of the seismic CDF [core damage frequency] and LERF for the purposes of the ILRT extension risk analysis.” Please provide justification, preferably quantitatively, that the current SPRA model provides a reasonable estimate of the seismic CDF and LERF for the ILRT extension analysis.

APLA RAI 11

For the following items, please provide a confirmation or explanation of the requested information:

a. In Table A-1 of Attachment 3 of the LAR, for SR IE-A7, the Finding and Observation states, “no interviews were conducted with plant personnel to determine if potential initiating events have been overlooked.” Please confirm that these interviews have been completed under the disposition of SR IE-A7.

b. In Table A-1 of Attachment 3 of the LAR, for the disposition of SR IE-C15, please confirm that mean values were also provided. If not, please provide justification as to why they were excluded. In addition, please confirm if uncertainties were characterized for initiating events other than those for LOCAs.

c. In Table A-1 of Attachment 3 of the LAR, for the disposition of SR SC-A1, the licensee states, “The use of core uncovery vs. peak cladding temperature of 1800^{o} Fahrenheit results in a slightly conservative time available for HFE [human factors engineering].” Please confirm that the slightly conservative time means there is less time available for actions to mitigate core uncovery than actions to mitigate peak cladding temperature, which would lead to higher human error probabilities to mitigate core uncovery.

d. In Table A-1 of Attachment 3 of the LAR, for the disposition of SR SY-A4, the licensee states, “It is not likely the current models including Internal Events and Fire still contain *gross* model errors or assumptions which result in significant deviation from the as- built as-operated plant condition or configuration.” Please confirm that “gross” model errors or assumptions do not exist in the current Internal Events and Fire models.

e. In Table A-1 of Attachment 3 of the LAR, for the disposition of SR HR-H2, the licensee states, “the minor change to non-Operations staffing levels does not impact existing HFEs.” Please confirm if there was any effect when minimal vs. normal staffing levels were assumed. Also please confirm if any staff that is credited for an internal event response could be unavailable for an external event response due to collateral duty (e.g., fire brigade).

f. In Table A-1 of Attachment 3 of the LAR, for the disposition of SR QU-F6, please confirm if the Fussell-Vesely importance was also calculated. If so, please provide the defined level of significance. If not, please address this and its potential effect on the application.

g. In Table A-3 of Attachment 3 of the LAR, for the disposition of SR CF-A1 and SR CF-A2, please clarify if these failure probabilities are based on NUREG/CR-6850[4] instead of

[4] NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities,” Volumes 1 and 2 (ADAMS Accession Nos. ML15167A401 and ML15167S411).

NUREG/CR-7150, Volume 2[5], assess whether they remain conservative. If not, provide a sensitivity evaluation using the NUREG/CR-7150 updated values.

[5] NUREG/CR-7150, Volume 2, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE) (ADAMS Accession No. ML14141A129).