NRR-PMDAPEm Resource

From: Lingam, Siva
Sent: Wednesday, December 02, 2015 8:19 AM
To: Schrader, Kenneth (KJSe@pge.com)
Cc: Pascarelli, Robert; Shoop, Undine; Bucholtz, Kristy; Mazaika, Michael; 'mjrm@pge.com'; rntt@pge.com
Subject: Diablo Canyon 1 and 2 - Requests for Additional Information for License Amendment Request 15-03 to Adopt the Alternative Source Term per 10 CFR 50.67 (TAC Nos. MF6399 and MF6400)

By a letter dated June 17, 2015 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15176A539), as supplemented by letters dated August 31, October 22, November 2, and November 6, 2015 (ADAMS Accession Nos. ML15243A363, ML15295A470, ML15321A235, and ML15310A522, respectively), Pacific Gas and Electric (PG&E, the licensee), submitted a license amendment request (LAR) to revise the licensing bases to adopt the alternative source term (AST) as allowed by Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.67, "Accident source term," for Diablo Canyon Power Plant (DCPP), Units 1 and 2. Please note the following **official** requests for additional information (RAIs) from our Radiation Protection and Consequence Branch (ARCB) for the AST LAR. Please provide your responses within 60 days from the date of this e-mail. We transmitted the draft RAIs to you on November 9, 2015, and we had a clarification call on December 1, 2015. Your timely responses will allow the U.S. Nuclear Regulatory Commission staff to complete its review on schedule.

The Nuclear Regulatory Commission (NRC) staff reviewed the impact of implementing an alternative radiological source term for evaluating design basis accidents (DBAs) on all DBAs currently analyzed in the DCPP updated final safety analysis report (UFSAR) that could have the potential for significant dose consequences. The NRC staff used Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792) to perform the NRC staff's review. The NRC staff determined that more information is needed to complete the review.

Please note that many of the following RAIs might be answered by information already contained in the radiological accident analysis calculations performed by the licensee. Where the information is already provided in the calculations, it is acceptable to provide the calculations and state where the information is located. It is helpful to provide these calculations (including any electronic files for the RADTRAD computer code used to model the DBAs), because it usually increases the efficiency of the review.

ARCB-RAI-1

In the Enclosure to the application dated June 17, 2015 it states:

> Full implementation of [alternate source term] AST for DCPP Units 1 and 2 does not include revising … NUREG-0737 responses associated with shielding and vital area access.

However, RG 1.183, Regulatory Position 4.3, "Other Dose Consequences," states that:

> The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of [total effective dose equivalent] effective TEDE.

In evaluating the submittal, the NRC staff could not determine how RG 1.183, Regulatory Position 4.3 has been assessed for DCPP. Please provide additional information describing how Regulatory Position 4.3 has been assessed for DCPP.

ARCB-RAI-2

In Enclosure Attachment 4, Section 7.2.6 it states:

> To address the existing licensing basis, a TEDE dose is estimated for operator access to the control room. Because RG 1.183 does not provide guidance on determining the egress and ingress to the control room following an accident, the same inputs used to estimate the current licensing basis values for access to the control room, along with the associated dose estimate presented in the UFSAR, are used to determine the TEDE dose estimate for ingress/egress.

In addition, Enclosure Table 1 and Enclosure Attachment 4 Table 8.1-1 state that the control room total effective dose equivalent presented for the loss of coolant accident represents the operator dose due to occupancy which is 3.7 rem and that the value shown in parenthesis represents that portion of the total dose reported that is the contribution of direct shine from contained sources/external cloud which is 0.7 rem. Furthermore, they stated that the dose to the control room operator during routine access for the 30 day duration of the accident is discussed in Enclosure Attachment 4, Section 7.2.6 and summarized in the text of Enclosure Attachment 4, section 8.0 which is 0.037 rem.

10 CFR 50.67, “Accident source term,” (b)(2) states:

The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(i)…

(ii)…

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

In order to meet 10 CFR 50.67, the radiation dose for accessing the control room must be evaluated. PG&E proposes to use the same inputs used to estimate the current licensing basis values for accessing the control room. However, the AST uses a different source term than that in the current licensing basis. Therefore, please provide an analysis of the radiation dose received from accessing the control room that reflects the new source term proposed with implementing the AST.

ARCB-RAI-3

Please provide the RADTRAD input and output files, in electronic format, for each of the AST DBAs described in the LAR.

ARCB-RAI-4

Please confirm that the failed fuel percentage (10%) stated in the Enclosure Attachment 4 Table 4.3-1 note, which is applied to the control rod ejection accident and the locked rotor accident, is equivalent to 10% of the rods in the core as opposed to 10% of the rods in an assembly.

ARCB-RAI-5

DELETED.

ARCB-RAI-6

Enclosure Attachment 4, Section 6.2, “Direct Shine Dose from External and Contained Sources,” states:

> CB&I S&W Inc. [Chicago Bridge and Iron Stone and Webster Incorporated] [A CB&I Company] point kernel shielding computer program SW-QADCGGP is used to calculate the deep dose equivalent (DDE) in the control room, [technical support center] TSC and at the [exclusion area boundary] EAB due to external and contained sources. The calculated DDE is added to the inhalation (CEDE) and the submersion factors are used and the geometry models are prepared to ensure that un-accounted streaming/scattering paths were eliminated. The dose albedo method with conservative albedo values is used to estimate the scatter dose in situations where the scattering contributions are potentially significant. [American National Standards Institute/American Nuclear Society] ANSI/ANS 6.1.1-1977 "neutron and gamma-ray flux-to-dose-rate factors" (Reference 31) is used to convert the gamma flux to the dose equivalent rate.

Enclosure Attachment 4, Section 7.2.7, "Technical Support Center Dose," provides further detail on the direct shine dose to the technical support center from external and contained sources. Enclosure Attachment 4, Section 7.2.7, states:

> CB&I S&W Inc. computer code PERC2 is used to calculate the dose to TSC personnel due to airborne radioactivity releases following a [loss of coolant accident] LOCA. The direct shine dose to an operator in the TSC due to contained or external sources resulting from a postulated LOCA is calculated using CB&I S&W Inc. point kernel shielding computer program SW-QADCGGP. The post-LOCA gamma energy release rated [mega electron volt per second] (MeV/sec) and integrated gamma energy release [mega electron volt hour per second] (MeV-hr/sec) in the various external sources are developed with computer program PERC2.

In evaluating the LAR, the NRC staff could not thoroughly review and perform confirmatory calculations of the direct shine dose due to external and contained sources at the control room, TSC, low population zone (LPZ), and at the EAB from the discussions described in the LAR. Therefore, please provide additional information in enough detail that will enable the NRC staff to be able to perform an independent calculation of the direct shine dose to the control room and the TSC from external and contained sources.

ARCB-RAI-7

RG 1.183 Appendix A, Section 3.3 states:

> The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified.

In Enclosure Attachment 4 it states:

> In accordance with RG 1.183, Appendix A, Section 3.3, prior to [containment fan cooler unit] CFCU initiation, the dose consequence model assumed a mixing rate attributable to natural convection between the sprayed and unsprayed regions of 2 turnovers of the unsprayed region per hour.

However, there was no discussion presented that demonstrates that adequate flow exists between the sprayed and unsprayed regions when the CFCUs are not in operation. Therefore, please provide a discussion that demonstrates that adequate flow exists between the sprayed and unsprayed regions when the CFCUs are not in operation.

ARCB-RAI-8

In Enclosure Attachment 4, Section 7.2.3.4, it states that the residual heat removal (RHR) pump seal failure resulting in a filtered release via the plant vent is DCPP's licensing basis with respect to the worst case passive single failure in the RHR system and is being retained as a release pathway for the AST LOCA dose consequence analysis. The analysis provided appears to be consistent with RG 1.183 Appendix A, Regulatory Positions section 5 with the exception of Regulatory Position 5.2.

RG 1.183 Appendix A, Regulatory Position 5.2 states that the engineered safety features (ESF) leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specification, or licensee commitments to item III.D.1.1 of NUREG-0737 would require declaring such systems inoperable.

The RHR pump seal failure is considered to be ESF leakage. However, the analysis does not take into account two times the leakage in accordance with RG 1.183 Appendix A, Regulatory Position 5.2.

Please provide an analysis that is consistent with RG 1.183 Appendix A, Regulatory Position 5.2 or provide a technical evaluation of the deviation from RG 1.183.

ARCB-RAI-9

In Enclosure Attachment 4, Section 7.2.3.5, it states that as part of this application, DCPP is proposing to establish administrative acceptance criteria to ensure the total as-tested back leakage into the refueling water storage tank (RWST) from the containment recirculation sump is less than or equal to 1 gallon per minute (gpm). However, there is no further discussion about the administrative acceptance criteria. Furthermore, 10 CFR 50.36(c)(2)(ii) states:

A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

> *(A) Criterion 1…*
> *(B) Criterion 2*. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
> *(C) Criterion 3…*
> *(D) Criterion 4…*

The 1 gpm back leakage into the RWST is an initial condition of the design basis loss of coolant accident radiological consequence that assumes the failure of a fission product barrier. Please explain how the DCPP Technical Specifications meet 10 CFR 50.36(c)(2)(ii)(B) for the RWST back leakage.

ARCB-RAI-10

In Enclosure Attachment 4, Section 7.2.3.6, it states that as part of this application, DCPP is proposing to establish administrative acceptance criteria to ensure the total as-tested flow hard piped to the miscellaneous equipment drain tank (MEDT) is less than 950 cubic centimeters per minute (cc/min) of ESF system leakage and 484 cc/min of non-radioactive fluid leakage. However, there is no further discussion about the administrative acceptance criteria. Furthermore, 10 CFR 50.36(c)(2)(ii) states:

A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

> *(A) Criterion 1…*
> *(B) Criterion 2*. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
> *(C) Criterion 3…*
> *(D) Criterion 4…*

The 950 cc/min leakage into the MEDT is an initial condition of the design basis loss of coolant accident radiological consequence that assumes the failure of a fission product barrier. Please explain how DCPP Technical Specifications meet 10 CFR 50.36(c)(2)(ii)(B) for this parameter.

ARCB-RAI-11

In Enclosure Attachment 4, Section 7.2.3.1, “Containment Pressure/Vacuum Relief Line Release,” it states:

> It is conservatively assumed that 40% of release flashes and is instantaneously and homogeneously mixed in the containment atmosphere, and that the activity associated with the volatiles, i.e., 100 % of the noble gases and 40% of the iodine in the reactor coolant, is available for release to the environment via this pathway.

RG 1.183 Appendix A, Regulatory Position 3.8 states that the purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA and that this inventory should be based on the TS reactor coolant system equilibrium activity. RG 1.183 Appendix A, Regulatory Position 3.8 does not make any statements about a reactor coolant system liquid flashing fraction. However, it is the NRC staff’s position that <u>100% of the radionuclide inventory</u> in the reactor coolant system liquid is released to the containment and therefore, the flashing fraction would be 100% which is conservative and meets the intent of RG 1.183 Appendix A, Regulatory Position 3.8.

Please provide a containment pressure / vacuum relief line release analysis that is consistent with RG 1.183 Appendix A, Regulatory Position 3.8, as described above or explain how the flashing fraction of 40% is more conservative than a flashing fraction of 100%.

ARCB-RAI-12

In Enclosure Attachment 4, Section 7.1, “Control Room Design / Operation / Transport Model,” it describes the basic operation of the control room ventilation system (CRVS), provides the Mode 4 parameter values assumed in the dose consequence analysis and references a December 2012 control room tracer gas test for the maximum unfiltered in leakage to the control room envelope (CRE). However, the Mode 4 parameter values do not seem to match the description of the CRVS operation provided in Section 7.1. One example is that the discussion states that in Mode 4 the pressurization flow at either intake is between 650 – 900 cubic feet per minute (cfm) and the air is taken from the less contaminated intake, however, in the Mode 4 parameter table it shows the pressurization flow of 650 – 900 cfm and also includes a filtered intake of 550 – 800 cfm. It is not clear to the NRC staff if this is another outside air intake or if this is part of the recirculation flow path, therefore, please provide a simplified diagram of the CRVS and explain in further detail the operation CRVS including the specific flow rates through the components as compared to the values assumed in the dose consequence analyses.

In addition, please provide a summary describing the December 2012 CRE test results and the test configurations.

ARCB-RAI-13

In Enclosure Attachment 4, Section 7.2.7, “Technical Support Center Dose,” it describes the basic operation of the TSC ventilation system and then states that it utilizes the CRVS. However, it is not clear to the NRC staff how the TSC ventilation system and the CRVS are combined in operation. Therefore, please provide a simplified diagram of the TSC ventilation system/CRVS and explain in further detail the operation TSC/CRVS including the specific flow rates through the components as compared to the values assumed in the dose consequence analyses.

ARCB-RAI-14

In the Enclosure Section 2.1, “Proposed Changes to Current Licensing Basis,” item 22 PG&E requests the following:

> Credit the following existing administrative controls reflected in plant procedures. These administrative controls ensure the [fuel handling building] FHB is maintained at a negative pressure relative to

atmosphere during movement of irradiated fuel in the spent fuel pool, thus ensuring that the environmental releases occur via the Unit vent.

- The movable wall is in place and secured.
- No exit door from the FHB is propped open.
- At least one FHB ventilation system exhaust fan is running.

Attachment 4, Section 7.3 presents the [fuel handling accident] FHA. Credit for the above administrative controls is taken to facilitate that the post-accident environmental release of radioactivity occurs via the plant vent.

In Enclosure Attachment 4, Section 7.3, it states that the radioactivity release pathways following a FHA in the FHB are established taking into consideration the following administrative controls:

- The movable wall is put in place and secured.
- No exit door is propped open.
- One fuel handling building ventilation system (FHBVS) exhaust fan is operating.

It further states that operation of the FHBVS with a minimum of one exhaust fan operating and all significant openings administratively closed will ensure negative pressure in the FHB which will result in post-accident environmental release of radioactivity occurring via the plant vent. However, TS 3.7.13, "FHBVS," does not require one exhaust fan to be operating, the movable wall to be in place and secured, or require the exit to be closed.

10 CFR 50.36(c)(2)(ii) states:

A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

(A) Criterion 1…
(B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
(C) Criterion 3…
(D) Criterion 4…

In addition RG 1.183, Regulatory Position 5.1.2 states

That credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications …"

The administrative controls discussed above are initial conditions of the design basis FHA in the FHB radiological consequence that assumes the failure of a fission product barrier. Please explain how DCPP Technical Specification 3.7.13 meets 10 CFR 50.36(c)(2)(ii)(B) for these administrative control conditions and meets RG 1.183, Regulatory Position 5.1.2.

ARCB-RAI-15

In Enclosure Attachment 4, Section 7.3 it has been determined that for the FHA in the FHB, the actual release rate lambda based on the FHBVS exhaust (i.e., 8.7 hr^{-1}) is larger than the release rate applicable to "a 2-hr release" per regulatory guidance (i.e., 3.45 hr^{-1}). Thus the larger exhaust rate lambda associated with FHBVS operation plus the exhaust rate lambda for the 500 cfm outleakage is utilized in the analysis.

Please provide the technical basis for why this approach is conservative for control room doses.

ARCB-RAI-16

Appendix B of Regulatory Guide (RG) 1.183, Regulatory Position 1.1 states:

> The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.

In Enclosure Attachment 4 Section 7.3, it postulates that a spent fuel assembly is dropped during refueling in the spent fuel pool (SFP) located in the FHB, or in the reactor cavity located in containment and that all the rods in the dropped assembly are damaged.

The NRC staff reviewed the licensee's FHA analysis. The FHA activity is assumed to be released from (1) the damaged fuel via the SFP to the FHB, or (2) from the damaged fuel via the reactor cavity to the containment, from which it is assumed to be released to the environment within a two hour period as an unfiltered release via the plant vent or the containment equipment hatch, respectively. Credit was taken for the operation of one fuel handling building ventilation system exhaust fan, and the closure of the FHB exit doors. Credit was not taken for the fuel handling building ventilation system filtration. When evaluating the dose to operators in the control room, it was assumed that the control room normal intake radiation monitors would initiate CRVS pressurization mode 4 at 22 seconds following the start of the event.

DCPP technical specification (TS) 3.3.7, "Control Room Ventilation System (CRVS) Actuation Instrumentation," states that the CRVS actuation instrumentation for each function in Table 3.3.7-1 shall be operable, and is applicable according to Table 3.3.7-1, which is during MODES 1, 2, 3, 4, 5, and 6, and during movement of recently irradiated fuel assemblies.

DCPP TS 3.3.8, "Fuel Handling Building Ventilation System (FHBVS) Actuation Instrumentation," states that the FHBVS actuation instrumentation for each function in Table 3.3.8-1 shall be operable, and is applicable according to Table 3.3.8-1, which is during movement of recently irradiated fuel assemblies in the fuel handling building.

DCPP TS 3.7.10, "Control Room Ventilation System (CRVS)," states that two CRVS trains shall be operable, and is applicable during MODES 1, 2, 3, 4, 5, and 6, and during movement of recently irradiated fuel assemblies.

DCPP TS 3.7.13, "Fuel Handling Building Ventilation System (FHBVS)," states that two FHBVS trains shall be operable and is applicable during movement of recently irradiated fuel assemblies in the fuel handling building.

DCPP TS 3.7.15, "Spent Fuel Pool Water Level," states that the spent fuel pool water level shall be ≥ 23 feet over the top of irradiated fuel assemblies seated in the storage racks and is applicable during movement of irradiated fuel assemblies in the spent fuel pool.

DCPP TS 3.9.7, "Refueling Cavity Water Level," states that the refueling cavity water level shall be maintained ≥ 23 feet above the top of reactor vessel flange and is applicable during movement of irradiate fuel assemblies within containment.

Given that the licensee voluntarily has requested a change to its licensing basis, please explain how the proposed revised fuel handling accident analysis meets or bounds Regulatory Guide 1.183, Appendix B Regulatory Position 1.1. Specifically, taking into account that the technical specification applicability does not:

1. Require CRVS actuation instrumentation or CRVS to be operable when in no mode during the movement of other radioactive loads (those other than recently irradiated fuel such as sources) in the SFP and other loads (such as fresh fuel) over irradiated fuel in the FHB or containment.

2. Require the FHBVS actuation instrumentation to be operable, a FHBVS exhaust fan to be running, or exit doors to be closed, or any of the equipment to be operable during movement of other radioactive loads (not defined as “recently” irradiated fuel) or loads (such as fresh fuel assemblies or sources, etc.) over irradiated fuel in the FHB.

3. Require the SFP or reactor cavity water level to be at least 23 feet during the movement of other radioactive loads (those other than irradiated fuel such as sources) and other loads (such as a fresh fuel) over irradiated fuel in the FHB or containment.

In addition, clarify how the revised analysis determines the most limiting case and how the fuel handling analysis shows that the limiting case is not the drop of a fuel assembly or object other than a recently irradiated fuel assembly.

Siva P. Lingam
U.S. Nuclear Regulatory Commission
Project Manager (NRR/DORL/LPL4-1)
Diablo Canyon Nuclear Power Plant
Location: O8-D5; Mail Stop: O8-B3
Telephone: 301-415-1564; Fax: 301-415-1222
E-mail address: siva.lingam@nrc.gov

Hearing Identifier: NRR_PMDA
Email Number: 2521

Mail Envelope Properties (Siva.Lingam@nrc.gov20151202081900)

Subject: Diablo Canyon 1 and 2 - Requests for Additional Information for License Amendment Request 15-03 to Adopt the Alternative Source Term per 10 CFR 50.67 (TAC Nos. MF6399 and MF6400)
Sent Date: 12/2/2015 8:19:06 AM
Received Date: 12/2/2015 8:19:00 AM
From: Lingam, Siva

Created By: Siva.Lingam@nrc.gov

Recipients:
"Pascarelli, Robert" <Robert.Pascarelli@nrc.gov>
Tracking Status: None
"Shoop, Undine" <Undine.Shoop@nrc.gov>
Tracking Status: None
"Bucholtz, Kristy" <Kristy.Bucholtz@nrc.gov>
Tracking Status: None
"Mazaika, Michael" <Michael.Mazaika@nrc.gov>
Tracking Status: None
"'mjrm@pge.com'" <mjrm@pge.com>
Tracking Status: None
"rntt@pge.com" <rntt@pge.com>
Tracking Status: None
"Schrader, Kenneth (KJSe@pge.com)" <KJSe@pge.com>
Tracking Status: None

Post Office:

Files	**Size**	**Date & Time**
MESSAGE	25549	12/2/2015 8:19:00 AM

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