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August 31, 2015

PG&E Letter DCL-15-105

10 CFR 50.90

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

Diablo Canyon Units 1 and 2
Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Supplement to License Amendment Request 15-03, "Application of Alternative Source Term"

References: 1. PG&E Letter DCL-15-069, "License Amendment Request 15-03, 'Application of Alternative Source Term,'" dated June 17, 2015 (ADAMS Accession No. ML 15176A539)

2. NRC Letter, "Diablo Canyon Power Plant, Unit Nos. 1 AND 2 - Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Review of Alternative Source Term License Amendment Request (TAC Nos. MF6399 AND MF6400)," dated August 13, 2015 (ADAMS Accession No. ML15219A016)

Dear Commissioners and Staff:

In Reference 1, Pacific Gas and Electric Company (PG&E) submitted License Amendment Request (LAR) 15-03 that proposes to revise the Diablo Canyon Power Plant Units 1 and 2 licensing bases to adopt the alternative source term (AST) as allowed by 10 CFR 50.67.

In Reference 2, the NRC provided the results of the NRC staff's acceptance review of this LAR. The NRC staff concluded that in order to make the application complete PG&E needs to supplement the application to address the following information:

1. In Appendix B of Attachment 4 of the LAR, the thermal-hydraulic inputs to the accident dose calculations have been changed from the current licensing basis (CLB) without providing justification. For NRC to accept the LAR, the NRC staff requires justification for these changes made to the thermal-hydraulic analysis from the CLB to the AST contained in Appendix B of Attachment 4 of the LAR.

2. Please provide a license condition indicating that the modifications will be completed in support of the commitments made in Attachment 7 of the LAR. During the clarification call on July 28, 2015, the licensee stated that these modifications will be implemented prior to application of the amendment.

The Enclosure to this letter contains summary information in response to the NRC's first request listed above. Additional detailed information on changes made to the thermal-hydraulic analysis from the CLB to the AST analysis, in addition to that previously contained in Appendix B of Attachment 4 of the LAR 15-03, will be submitted by September 30, 2015.

In response to the NRC's second request listed above, PG&E proposes the following license condition to support application of AST:

Upon implementation of the amendment adopting the alternative source term, the following plant modifications will be complete:

- Install shielding material, equivalent to that provided by the Control Room outer walls, at the external concrete west wall of the Control Room briefing room prior to implementation of Alternate Source Term.
- Install a high efficiency particulate air filter in the Technical Support Center normal ventilation system.
- Re-classify a portion of the 40-inch Containment Penetration Area (GE/GW) Ventilation line from PG&E Design Class II to PG&E Design Class I and upgrade the damper actuators, pressure switches, and the damper solenoid valves to PG&E Design Class I prior to implementation of Alternate Source Term.
- Re-classify a portion of the 2-inch gaseous radwaste system line which connects to the Plant Vent as PG&E Design Class I prior to implementation of Alternate Source Term.
- Update setpoints for the redundant safety related gamma sensitive area radiation monitors (1-RE 25/26, 2-RE 25/26) prior to implementation of Alternate Source Term.

Attachment 1 to the Enclosure of this letter contains the proposed markup for the license conditions in Appendix D of the Unit 1 and 2 operating licenses. The plant modifications contained in Attachment 1 to the Enclosure were previously proposed as license commitments one through five in Attachment 7 to the Enclosure of Reference 1. Since the modifications are being proposed as license conditions, the commitments one through five previously provided in Attachment 7 to the Enclosure of Reference 1 are deleted.

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

This communication contains new and revised regulatory commitments (as defined by NEI 99-04). The new commitment is contained in Attachment 2 to the Enclosure.

In accordance with site administrative procedures and the Quality Assurance Program, the proposed amendment has been reviewed by the Plant Staff Review Committee.

Pursuant to 10 CFR 50.91, PG&E is sending a copy of this proposed amendment to the California Department of Public Health.

If you have any questions or require additional information, please contact Hossein Hamzehee at 805-545-4720.

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

Executed on August 31, 2015.

Sincerely,

Barry S. Allen

Barry S. Allen
Vice President, Nuclear Services

kjse/4328/50705089
Enclosure
cc: Diablo Distribution
cc/enc: Marc L. Dapas, NRC Region IV
Thomas R. Hipschman, NRC Senior Resident Inspector
Siva P. Lingam, NRR Project Manager
Gonzalo L. Perez, Branch Chief, California Dept of Public Health

Justification for Changes Made to the Thermal-Hydraulic Analysis from the Current Licensing Basis to the Alternative Source Term Contained in License Amendment Request 15-03

NRC Request:

In Appendix B of Attachment 4 of the LAR, the thermal-hydraulic inputs to the accident dose calculations have been changed from the current licensing basis (CLB) without providing justification. For NRC to accept the LAR, the NRC staff requires justification for these changes made to the thermal-hydraulic analysis from the CLB to the AST contained in Appendix B of Attachment 4 of the LAR.

PG&E Response:

This response provides the requested clarification with respect to License Amendment Request (LAR) 15-03 Attachment 4, Appendix B (hereafter referred to as "LAR Appendix B"), regarding the technical basis for the revised thermal hydraulic (T/H) inputs being used in the Alternate Source Term (AST) dose analyses compared to the T/H inputs used in the current licensing basis (CLB) dose analyses.

The T/H inputs to dose analysis are characterized as initial plant conditions [such as reactor coolant system (RCS) liquid mass, minimum useable water volume in the refueling water storage tank, steam generator (SG) liquid mass etc.], and plant transient response values (such as actuation times of plant safeguards features, time-dependent break flowrates and associated flash fractions, steam releases to the atmosphere, etc.). Heating ventilation and air conditioning system input parameters (such as air flow rates and filter efficiencies) are not considered T/H parameters since these are independent of the plant accident response.

As discussed below, the T/H inputs to the AST dose analyses for the loss of coolant accident (LOCA), main steam line break (MSLB) and steam generator tube rupture (STGR) are directly obtained from the T/H safety analyses results documented in the Updated Final Safety Analysis Report (UFSAR). The AST dose analyses for the loss of load (LOL), locked rotor accident (LRA), and control rod ejection accident (CREA) are based on conservatively calculated LOL secondary steam releases to the atmosphere that bound the T/H safety analyses results documented in the UFSAR. The fuel handling accident does not involve any T/H inputs as defined above and therefore, is not included within this response.

Table 1 lists each of the T/H safety analyses that support LAR 15-03 and the corresponding source documents that summarize the associated analyses of record. The computer code methodology used for the listed analyses are also identified. These analyses and codes are considered the CLB based on receipt of NRC approval of the LAR or via the 10 CFR 50.59 process and are documented in the UFSAR. This information is further discussed in item 6 below.

The following information is provided to address the six points of clarification discussed in the follow up conference call between PG&E and NRC held on August 24, 2015.

1. Provide a description for each analysis at a level of detail that is consistent with the UFSAR

Table 2 lists the specific UFSAR section that contains a T/H description of each of the six T/H safety analyses that support the LAR 15-03.

2. Provide the initial conditions used for each analysis

Table 2 lists the specific UFSAR section that contains a description of the key initial conditions used for each of the six T/H safety analyses that support the LAR 15-03.

3. Provide a summary level explanation for how the inputs were determined or maximized for each analysis

In 2006 – 2007, PG&E implemented the Replacement Steam Generator (RSG) Program in which the majority of the UFSAR accident analyses were updated per 10 CFR 50.59 to incorporate the new Westinghouse Model Delta 54 SGs (Reference 1). Table 1 shows that the LRA and the CREA T/H analyses have not been revised since 1989 and did not require updating for the RSGs since these are faults that occur in the primary side and are not sensitive to the specific SG characteristics. As part of the RSG Program, Westinghouse revised the RCS liquid volume/mass, SG liquid volume/mass, and SG pressure values used for initial condition input assumptions. These RSG analyses were based on the Unit 1 full power design RCS T_{avg} of 577.3°F, the Unit 2 design RCS T_{avg} of 577.6°F, and the full power T_{feed} of 435°F for both units. Soon after the RSG Program, PG&E implemented a T_{avg}/T_{feed} Ranges Program (Reference 2) to expand the allowable range of these full power operating parameters. The T_{avg}/T_{feed} Ranges Program performed the necessary nuclear steam supply system (NSSS), component, and safety analyses to support operation of Unit 1 for the full power RCS T_{avg} range of 565°F to 577.3°F, Unit 2 for the RCS T_{avg} range of 565°F to 577.6°F, and both units for the T_{feed} range of 425°F to 435°F. The majority of the T/H accident analyses were bounding based on the already assumed maximum RCS T_{avg}/T_{feed} values for the RSG analyses, but as listed in Table 1, the LOL steam releases and the MSLB steam releases required revision to bound the increased T_{avg}/T_{feed} range.

As described in the LAR 15-03, the Licensing Basis Verification Program identified several non-conformances in the CLB dose calculation inputs/methodology with respect to Diablo Canyon Power Plant (DCPP) licensing and design basis that are currently evaluated in an operability determination. In the process of revising these dose analyses to implement the AST methodology per LAR 15-03, the AST analysis T/H inputs were updated to be consistent with (or conservative with respect to) these latest UFSAR safety analyses that are summarized in Table 1. For each of the six AST accidents listed in Table 1, the RCS liquid mass, SG liquid mass, and SG secondary steam releases are based on the RSG/RCS T_{avg}/T_{feed} Ranges Program, and the

minimum or maximum value selected in order to maximize the calculated AST dose. In certain instances, additional conservatism was assumed for the AST inputs independent of that assumed in the UFSAR T/H analysis. For example, for an AST dose analysis that assumes a minimum RCS mass, the volume associated with the pressurizer and surge line was conservatively ignored. Similarly, SG level uncertainties were applied to the SG mass input value to maximize the calculated AST dose independent of how the uncertainty was applied for the T/H UFSAR analysis. Each AST dose analysis is documented in a formal design calculation that includes a detailed description of the technical basis for each AST T/H input value and assumption. Specifically, the AST analyses identify the T/H analysis of record that are considered the source documents, and address the considerations that modify the specific design input for use in the dose analyses in a conservative manner.

4. Describe the major assumptions made for each analysis

Table 2 lists the specific UFSAR section that contains a description of the major assumptions used for each of the six T/H safety analyses that support the LAR 15-03.

5. Provide the peak cladding temperatures (PCT) for the LOCA PCT analysis

The current small break LOCA PCT for Unit 1 is 1391°F and for Unit 2 is 1288°F. The current large break best-estimate LOCA PCT for Unit 1 is 2124°F and for Unit 2 is 2125°F. These were submitted in PG&E Letter DCL-15-093, "10 CFR 50.46 Annual Report of Emergency Core Cooling System Evaluation Model Changes for Peak Cladding Temperature for 2014," dated August 6, 2015.

6. Provide a summary level description of the RSG and RCS T_{avg}/T_{feed} Ranges Program analyses that were performed in accordance with generic NRC approved methodology and/or NRC approved DCPP licensing basis analytical models

The RSG and RCS T_{avg}/T_{feed} Ranges Program analyses were all performed in accordance with NRC approved methodology that has been a) specifically approved by the NRC for DCPP or b) NRC approval was provided to another utility that, in accordance with 10 CFR 50.59, has been evaluated and found acceptable for use by DCPP. Table 1 summarizes which T/H safety analyses have been revised by License Amendment (LA) or 10 CFR 50.59 evaluation since each original CLB dose analysis was completed. The summary below provides additional detail on the licensing and technical basis for each T/H accident analysis that provides input to the associated AST dose analysis listed in LAR Appendix B.

As described previously in item 3, Table 1 shows that the LRA and the CREA T/H analyses have not been revised since 1989 and were not updated as part of the RSG and RCS T_{avg}/T_{feed} Ranges Program analyses.

Loss of Coolant Accident (LOCA)
The LOCA T/H analysis includes both the short term PCT analysis and the long term containment integrity response analysis.

As summarized in Table 1, both Unit 1 and Unit 2 have revised the LOCA PCT analyses in UFSAR 15.4.1 using the NRC approved codes as documented in LAs 191 (Reference 3) and 192 (Reference 4), respectively. These LOCA analyses were coordinated with the RSG Program and remained bounding for the T_{avg}/T_{feed} Ranges Program.

The LOCA containment integrity response analysis in UFSAR Appendix 6.2D was revised as part of the RSG Program and incorporated under 10 CFR 50.59 and per the guidance of NEI 96-07, Section 4.3.8.2 based on referencing the NRC Safety Evaluation Report of GOTHIC 7.2 for the Ginna plant (Reference 5).

Subsequent to the above, the Unit 1 and Unit 2 LOCA containment integrity response analyses have recently been updated per 10 CFR 50.59 in support of the DCPP design change document packages (DDP) listed in Table 1 to account for a reduction in the minimum containment fan cooler unit (CFCU) flow rate and a change to the engineered safety feature (ESF) timer sequence for loading the CFCUs onto the respective Class 1E 480V buses.

As described in LAR 15-03, PG&E is crediting recirculation spray in support of the AST dose analyses. While recirculation spray is not being credited in the UFSAR LOCA containment integrity analysis, Westinghouse used the same UFSAR GOTHIC 7.2 model to evaluate the effect on the long term post-LOCA containment response with the proposed recirculation spray duration following AST implementation. This containment response evaluation established the bounding conditions for the LOCA AST dose analysis and forms the basis for the associated T/H input values listed in LAR Appendix B for the LOCA.

The other major T/H input change from the CLB dose analysis is incorporating the increased technical specification minimum refueling water storage tank (RWST) useable volume which was implemented in LAs 199/200 (Reference 6) to demonstrate acceptable submergence of the new recirculation sump strainer in accordance with GL 2004-02.

Main Steam Line Break
The MSLB CLB T/H analysis is based on the reanalysis performed for the RSG Program per 10 CFR 50.59 using the NRC approved code RETRAN-02W documented in WCAP-14882-P-A (Reference 7). The MSLB AST dose analysis incorporates the revised RCS and SG initial condition T/H inputs and secondary steam releases that are consistent with the RSG Program and bound the lower RCS temperature range associated with the T_{avg}/T_{feed} Ranges Program.

Steam Generator Tube Rupture
The SGTR CLB T/H analysis was reanalyzed as part of the RSG Program per 10 CFR 50.59 based on using the NRC approved code RETRAN-02W documented in WCAP-14882-P-A (Reference 7). The SGTR AST dose analysis incorporates revised RCS and SG T/H inputs consistent with the RSG Program.

Loss of Load
The LOL CLB steam releases are based on the reanalysis performed for the RSG Program per 10 CFR 50.59 using a simplified and conservatively bounding heat balance calculation methodology that determines the amount of heat that would need to be dissipated via steam releases until residual heat removal operating conditions are reached. The LOL AST dose analysis incorporates the revised RCS and SG initial condition T/H inputs and secondary steam releases that are consistent with the RSG Program and bound the lower RCS temperature range associated with the T_{avg}/T_{feed} Ranges Program.

Locked Rotor Accident
The LRA CLB T/H analysis has not been revised and is based on the same methodology approved in LA 37/36 (Reference 8) in 1989. The LRA AST dose analysis incorporates revised RCS and SG initial condition T/H inputs, and uses the secondary steam releases estimated for the LOL event since that will bound the LRA (Reference 2).

Control Rod Ejection Accident
The CREA CLB T/H analysis has not been revised and is based on the same methodology approved in LA 37/36 (Reference 8) in 1989. The CREA AST dose analysis incorporates revised RCS and SG initial condition T/H inputs, and uses secondary steam releases estimated for the LOL event since that will bound the CREA (Reference 2).

References:

1. Westinghouse WCAP-16638-P, "Diablo Canyon Units 1 and 2 Replacement Steam Generator Program NSSS Licensing Report," Revision 1, dated January 2008

2. Westinghouse WCAP-16985-P, "Diablo Canyon Units 1 and 2 T_{avg} and T_{feed} Ranges Program NSSS Engineering Report," Revision 2, dated April 2009

3. NRC Letter, "Diablo Canyon Power Plant, Unit No. 1 – Issuance of Amendment Re: Technical Specification 5.6.5, 'Core Operating Limits Report (COLR),' (TAC NO. MC9299)," Unit 1 License Amendment 191, dated November 21, 2006

4. NRC Letter, "Diablo Canyon Power Plant, Unit No. 2 - Issuance of Amendment Re: TS 5.6.5, 'Core Operating Limits Report (COLR),' (TAC NO. MC9567)," Unit 2 License Amendment 192, dated December 20, 2006

5. R. E. Ginna Nuclear Power Plant, Docket No. 50-244, letter from Patrick D. Milano (NRC) to Mrs. Mary G. Korsnick, "R. E. Ginna Nuclear Power Plant - Amendment Re: 16.8 Percent Power Uprate (TAC No. MC7382)," dated July 11, 2006

6. NRC Letter, "Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Issuance of Amendments Re: Technical Specification 3.5.4, 'Refueling Water Storage Tank (RWST)' (TAC NOS. MD6895 AND MD6896)," Units 1 and 2 License Amendments 199/200, dated March 26, 2008

7. WCAP-14882-P-A, "RETRAN-02W Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," dated April 1999

8. NRC Letter, "Issuance of Amendments, (TAC NOS. 71387 and 71388)," Units 1 and 2 License Amendments 37/36, dated May 10, 1989

Table 1: Current License Basis Summary for AST Thermal Hydraulic Safety Analyses

UFSAR Accident Description	Source Document	Implemented by LA/50.59	Analysis Date	Code Methodology
LOCA PCT	WCAP-16638-P RSG Unit 1 LAR-05-07	LA 191	November 21, 2006	BELOCA WCAP-12945-P-A Addendum 1A
	WCAP-16638-P RSG Unit 2 LAR-06-02	LA 192	December 29, 2006	ASTRUM WCAP-16009-P-A
LOCA Containment Response	WCAP-16638-P RSG Unit 1 DDP 1000025064	50.59	December 2014	GOTHIC 7.2 (Kewaunee SER)
	WCAP-16638-P RSG Unit 2 DDP 1000024983	50.59	July 2014	GOTHIC 7.2 (Kewaunee SER)
LRA	LAR 88-08	LA 37/36	October 5, 1989	LOFTRAN/FACTRAN /THINC
CREA	LAR 88-08	LA 37/36	October 5, 1989	TWINKLE/FACTRAN /THINC
MSLB Steam Releases	WCAP-16638-P RSG WCAP-16985-P T_{avg}/T_{feed}	50.59	January 2008	RETRAN-02W WCAP-14882-P-A
SGTR	WCAP-16638-P RSG	50.59	January 2008	RETRAN-02W WCAP-14882-P-A
LOL Steam Releases	WCAP-16638-P RSG WCAP-16985-P T_{avg}/T_{feed}	50.59	April 2009	N/A - Conservative Heat Balance Calculation

Table 2: Applicable Updated Final Safety Analysis Report Sections for Alternate Source Term Thermal Hydraulic Safety Analyses

FSAR Accident Description	UFSAR T/H Description	UFSAR Initial Conditions	UFSAR Assumptions
LOCA - Unit 1 LOCA - Unit 2	15.4.1.4A 15.4.1.4B	Table 15.4.1-7A Table 15.4.1-7B	Table 15.4.1-7A Table 15.4.1-7B
LOCA Containment Response	App 6.2D.3	Table 6.2D-1	Table 6.2D-17 Table 6.2D-18 Table 6.2D-19 Table 6.2D-20
LRA	15.4.4	15.4.4.2	15.4.4.2
CREA	15.4.6	15.4.6.2	15.4.6.2
MSLB	15.4.2.1.2	15.4.2.1.3	15.4.2.1.3
SGTR	15.4.3.2	15.4.3.3.2	15.4.3.3.2
LOL	15.2.7.2	15.2.7.3	15.2.7.3

Proposed Markup of Appendix D of the License Conditions in the Unit 1 and 2 Operating Licenses

Facility Operating License No. DPR-80

REMOVE	INSERT
Appendix D page 3	Appendix D page 3
	Appendix D page 4

Facility Operating License No. DPR-82

REMOVE	INSERT
Appendix D page 3	Appendix D page 3
	Appendix D page 4

Amendment Number	Additional Conditions	Implementation Date
	Following implementation, this condition will be performed as stated in the condition:	

> The first performance of SR 3.7.10.5, in accordance with Specification 5.5.19.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful tracer gas test, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.
>
> The first performance of the periodic assessment of CRE habitability, Specification 5.5.19.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful tracer gas test, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.
>
> The first performance of the periodic measurement of CRE pressure, Specification 5.5.19.d, shall be within 24 months, plus the 182 days allowed by SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful pressure measurement test, or within 182 days if not performed previously.

INSERT #1

Amendment Number	Additional Conditions	Implementation Date
	Following implementation, this condition will be performed as stated in the condition: The first performance of SR 3.7.10.5, in accordance with Specification 5.5.19.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful tracer gas test, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years. The first performance of the periodic assessment of CRE habitability, Specification 5.5.19.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful tracer gas test, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years. The first performance of the periodic measurement of CRE pressure, Specification 5.5.19.d, shall be within 24 months, plus the 182 days allowed by SR 3.0.2, as measured from February 3, 2005, the date of the most recent successful pressure measurement test, or within 182 days if not performed previously.	

INSERT #1

Insert #1

XXX	Implementation of the amendment adopting the alternative source term shall include the following plant modifications: Install shielding material, equivalent to that provided by the Control Room outer walls, at the external concrete west wall of the Control Room briefing room. Install a high efficiency particulate air filter in the Technical Support Center normal ventilation system. Re-classify a portion of the 40-inch Containment Penetration Area (GE/GW) Ventilation line from PG&E Design Class II to PG&E Design Class I and upgrade the damper actuators, pressure switches, and the damper solenoid valves to PG&E Design Class I. Re-classify a portion of the 2-inch gaseous radwaste system line which connects to the Plant Vent as PG&E Design Class I. Update setpoints for the redundant safety related gamma sensitive area radiation monitors (1-RE 25/26, 2-RE 25/26).	The amendment is effective as of the date of its issuance and the condition shall be implemented within 365 days of its issuance

Regulatory Commitments

Commitment 1

Additional detailed information on changes made to the thermal-hydraulic analysis from the current licensing basis to the alternate source term analysis, in addition to that previously contained in Appendix B of Attachment 4 of the License Amendment Request 15-03, will be submitted by September 30, 2015.