UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 14, 2016

Mr. Edward D. Halpin
Senior Vice President and
Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 56, Mail Code 104/6
Avila Beach, CA 93424

SUBJECT: DIABLO CANYON POWER PLANT, UNITS 1 AND 2 - REGULATORY AUDIT REPORT FOR THE JANUARY 12-14, 2016, AUDIT AT THE WESTINGHOUSE FACILITY IN NORTH BETHESDA, MARYLAND, FOR THE LICENSE AMENDMENT REQUEST FOR ALTERNATE SOURCE TERM (CAC NOS. MF6399 AND MF6400)

Dear Mr. Halpin:

By letter dated June 17, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15176A539), as supplemented by letters dated August 31, October 22, November 2 and 6, and December 17, 2015 (ADAMS Accession Nos. ML15243A363, ML15295A470, ML15321A235, ML15310A522, and ML16004A363, respectively), Pacific Gas and Electric Company (PG&E, the licensee) submitted a license amendment request to revise the current licensing bases to adopt the alternative source term (AST) in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident source term," for Diablo Canyon Power Plant (DCPP), Units 1 and 2.

The U.S. Nuclear Regulatory Commission (NRC) staff conducted a regulatory audit at the Westinghouse facility in North Bethesda, Maryland, from January 12-14, 2016, to review the DCPP, Unit 1 and 2, calculations, which form the bases of the proposed AST. The enclosure to this letter describes the results of the NRC staff's audit.

If you have any questions, please contact me at 301-415-3016 or via e-mail at Balwant.Singal@nrc.gov.

Sincerely,

Balwant K. Singal

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosure:
Audit Report

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

REGULATORY AUDIT REPORT PERFORMED AT

WESTINGHOUSE FACILITY FROM JANUARY 12-14, 2016

IN SUPPORT OF THE LICENSE AMENDMENT REQUEST TO REVISE THE CURRENT

LICENSING BASES TO ADOPT THE ALTERNATIVE SOURCE TERM

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 BACKGROUND

By letter dated June 17, 2015 (Reference 1), as supplemented by letters dated August 31, October 22, November 2 and 6, and December 17, 2015 (References 2, 3, 4, 5, and 6, respectively), Pacific Gas and Electric Company (PG&E), submitted a license amendment request to revise the current licensing bases to adopt the alternative source term (AST) in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident source term," for Diablo Canyon Power Plant (DCPP), Units 1 and 2. In Attachment 4 to the letter dated June 17, 2015, the licensee summarized the analyses and results of the transient events that are expected to produce the most limiting dose consequences. To assist in its review, the U.S. Nuclear Regulatory Commission (NRC) staff conducted an audit at the Westinghouse Electric Company LLC (WEC) offices in Rockville, Maryland, from January 12-14, 2016. The purpose of this audit was to review the DCPP, Units 1 and 2, engineering calculations which form the bases of the proposed AST. More detail is provided in the audit plan dated December 24, 2015 (Reference 7). PG&E, WEC and NRC staff which participated in the audit are listed in the attached Table 1. Accession numbers in the Agencywide Documents Access and Management System (ADAMS) for WEC WCAPs, Calculations Notes (CNs), and Engineering Calculations (References 10 through 43) reviewed during the audit are not listed since these documents have not been submitted on the docket.

Team Assignments:

Area of Review	Assigned Auditor
Audit Team Lead	Matthew Hardgrove (NRC)
Technical Reviewer	William MacFee (NRC)
Technical Reviewer	Kristy Bucholtz (NRC)

Enclosure

2.0 AUDIT REPORT

Section 1 of the Enclosure to the letter dated June 17, 2015, provides the following description of the revised source terms.

> The AST methodology as established in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 [Reference 8] is used to calculate the offsite and Control Room radiological consequences for DCPP Units 1 and 2. Attachment 4 [of the licensee's letter dated June 17, 2015] contains a summary of the analyses and results for the following events that are expected to produce the most limiting dose consequences. Conformance to RG 1.183 is provided in Attachment 5 [of the licensee's letter dated June 17, 2015]. Those expected events are as follows:
>
> - Loss of Coolant Accident (LOCA)
> - Locked Rotor Accident (LRA)
> - Control Rod Ejection Accident (CREA)
> - Main Steam Line Break (MSLB)
> - Steam Generator Tube Rupture (SGTR)
> - Loss-of-Load (LOL) Event

During the audit, the NRC staff reviewed WEC topical report WCAP-16638-P, Revision 1, "Diablo Canyon Units 1 and 2 Replacement Steam Generator Program NSSS [Nuclear Steam Supply System] Licensing Report," January 2008 (Reference 10), and WCAP-16985-P, Revision 2, "Diablo Canyon Units 1 and 2 T_{avg} and T_{feed} Ranges Program NSSS Engineering Report," April 2009 (Reference 11). WCAP-16638-P, Revision 1 documents the technical basis for the steam generator (SG) replacements for Unit 1 prior to Cycle 16 and Unit 2 prior to Cycle 15. WCAP-16638-P, Revision 1, for the SG replacement, was implemented under the 10 CFR 50.59 process. WCAP-16638-P, Revision 1, and WCAP-16985-P, Revision 2, for the SG replacement provided a frame of reference for the changes of the assumed parameters for the thermal hydraulics of the AST. The following assumptions and bases were employed for the SG replacement analysis:

1. Current NSSS power level of 3425 MegaWatt ($MW_{t)}$ (3411 MW_t core power),

2. Nominal full-power main feedwater temperature of 435 degrees Fahrenheit (°F),

3. Current fuel type of 17x17 V5,

4. Unit 1 thermal design flow (TDF) of 87,700 gallons per minute (gpm)/loop and 2.3 percent flow measurement uncertainty (MU),

5. Unit 2 TDF of 88,500 gpm/loop and 2.3 percent flow MU,

6. Unit 1 core bypass flow of 7.5 percent, which includes intermediate flow mixes (IFMs), thimble plugs removed (TPR), and allowance for a future upflow conversion,

7. Unit 2 core bypass flow of 9.0 percent, which includes IFMs, TPRs, upflow conversion, and upper-head temperature reduction (UHTR),

8. SG tube plugging (SGTP) of 0 to 10 percent.

During the audit, the NRC staff reviewed the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses contained in WCAP-16638-P, Revision 1, which includes, Large Break Loss of Coolant Accident (LBLOCA), Small Break LOCA (SBLOCA), LOL, MSLB, LRA, and SGTR. The computer codes used to analyze the accidents were as follows:

- LBLOCA – WCOBRA/TRAC
- SBLOCA – NOTRUMP, SBLOCTA
- NON-LOCA – RETRAN-02W
- SGTR – RETRAN-02W
- MSLB – RETRAN-02W, LOFTRAN

During the audit, the licensee provided additional explanation regarding its response to request for additional information (RAI), RAI SRXB-RAI-2 (Reference 6). In its response, the licensee explained how DCPP, Units 1 and 2, conformed with the bounding power history outlined in Figure 1 of Draft Guide 1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," October 2009 (Reference 9). From the initial response, it was not clear how the peak linear heat generation rate (LHGR) and F_Q surveillance related to a bounding power envelope.

The licensee provided an example of peak LHGRs surveyed over the course of various previous cycles outlined in Figure 10 of Attachment P-1 of the calculation file 14078104-C-M-00011, Revision 1 (Reference 28). The previous cycle data was within the bounding power history. A discussion followed on how to confirm in the future that peak LHGRs were within the bounding power history and whether surveillances would be required or calculations would suffice. The issue was resolved using the language in RG 1.183, Revision 0, Footnote 11, which stated "calculations that use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load." It was concluded that using calculated peak LHGRs at the reload safety analysis point would suffice because they are bounding and is consistent with the language in RG 1.183, Revision 0.

To demonstrate compliance with the bounding power envelope in future cycles, the licensee stated in its RAI response that an update to the Administrative Procedure TS6.DC3, "Reload Core Design Process," will include an additional verification of core power peaking to confirm the peak rod LHGRs remain within the power envelope. The licensee showed an example during the audit of where this confirmation would take place. The licensee provided a document titled NF-PGE-15-70, "Summary of Reload Safety Analysis Checklist (RSAC) Values for Unit 2 Cycle 20," dated December 16, 2015, that demonstrated where the burnup dependent power peaking factor verification line items will reside.

In response to a clarification question by the NRC staff, the licensee confirmed that for the dose analysis the gap fractions would be applied to each failed fuel rod.

The NRC staff reviewed various calculation notes for different Chapter 15 events such as MSLB, SGTR, LOL, LRA, and CREA. Various thermal hydraulic parameters were of interest for each event, and are listed below.

<u>Main Steam Line Break</u>

(Assumptions used in analysis reside in the notes in Calculation File 14078104-C-M-00015 Revision 0, Reference 32)

The thermal hydraulic parameters of interest for MSLB are Minimum Reactor Coolant System (RCS) mass following accident, leak rate to faulted SG, liquid mass in each SG, release rate of SG liquid activity from faulted SG, and steam releases from intact SG. Each parameter was compared from the current licensing basis value to the value used in the dose analysis, was documented using the references that capture where and why the changes occurred were logged, and annotated appropriately.

Minimum RCS mass following the accident changed from 566,000 pounds-mass (lbm) to 446,486 lbm. The appropriate reference documents were ASTAP-14-13, dated June 3, 2014, Table I-1, Item 15 (Reference 44), and CN-CRA-08-16, Revision 0 (Reference 22). The RCS mass is a calculated value and a lower value is considered conservative with respect to iodine dose consequences.

The leak rate to the faulted SG changed from 10.5 gpm to 0.75 gpm. The original submittal covers the reason for the change, notably the new Technical Specification (TS) surveillance under TS 5.5.9, "Steam Generator (SG) Tube Inspection Program." The leakage rate will be confirmed by measurement and is considered conservative for the event. The leak rate also uses the RG 1.183, Revision 0, leakage density assumption.

The liquid mass in both the faulted and intact SGs changed in the AST analysis. The faulted mass changed from 162,784 lbm to 182,544 lbm. The intact SG changes from 81,500 lbm/SG to 92,301 lbm/SG. The faulted SGs were captured in ASTAP-14-013 Table I-1, Item 14 and CN-CRA-08-16, Revision 0, Appendix A, Tables A-2 and A-3 (Reference 22). It may be noted that the mass is rounded up and a 10 percent uncertainty is added based on WEC assumptions to add conservatism. For the intact SGs, the masses were captured in CN-CRA-05-32, Revision 1 Tables 2-1 and 2-2, Case 1-7 and 2-7 (Reference 18). The calculated value was decreased by 10 percent per WEC suggestion to add conservatism.

The release rate of the SG liquid activity from the faulted SG changed from instantaneous to the dryout of SG liquid in 10 seconds. The change is captured in ASTAP-14-013 Item 11 and WCAP-16638-P, Figure 6.3.3.14-11 (Reference 10). WCAP-16638-P captures a new analysis of record completed in CN-TA-05-137, and WCAP-16638-P covers the analysis supporting the change in the assumption.

The steam releases from the intact SG changed for the different time periods. For 0 to 2 hours, the value changed 393,464 lbm to 684,000 lbm. For 2 to 8 hours, the value changed from 915,000 lbm to 893,000 lbm. Also, the timeframe was extended to 10.73 hours. The key references are WCAP-16985-P, Revision 2, Table 6.4.2-1 and ASTAP-14-013, Items 7 and 18. It's important to note that blowdown is not a function of time and the steam release is assumed to be rapid at no-load conditions. CN-CRA-08-16, Table 2-1 (Reference 22), also shows that the change is due to a lower feedwater temperature in the 0 to 2-hour timeframe. The time was increased out to 10.73 hours to address cooldown limitations during the accident.

Steam Generator Tube Rupture

(Assumptions used in analysis reside in the notes in Calculation File 14078104-C-M-00015 Revision 0, Reference 32)

The thermal hydraulic parameters of interest for SGTR are the RCS mass, initial SG liquid mass, steam flow rate to the condenser from the ruptured SG before trip and from the intact SGs before trip, steam releases from the ruptured SG and intact SGs, the post-accident minimum SG liquid mass for the ruptured SG and intact SGs, tube leakage rate, and break flow from RCS into ruptured SG.

The RCS mass changed from 499,500 lbm to 446,486 lbm and the change was documented in ASTAP-14-013. The notes on why the change occurred are captured in the MSLB section.

The initial SG liquid mass changed for both the intact and faulted SG. The intact changed from 118,500 lbm to 89,707 lbm and the faulted changed from 106,000 lbm to 89,707 lbm. The changes are captured in CN-CRA-05-53, Revision 1 (Reference 13). For the ruptured opened Power Operated Relief Valves phase, the average was taken from the RETRAN outputs and rounded down to increase the Iodine inventory in the SG liquid.

The steam flow rate to condenser from the ruptured SG and intact SG before trip did not change as the value is from the nominal full power steam flow rate (63,000 lbm/min).

The steam releases from ruptured SG and intact SGs vary over break time. The values change over the break time however there was nothing provided to show the change from the current licensing basis to the AST values. However, the values were captured in CN-CRA-05-54, Revision 0, Appendix B, Table B-3 (Reference 21), and ASTAP-14-013, Items 7 and 8, and Table I-2.2 with the results of the analysis shown.

The post-accident minimum SG liquid mass changed for the ruptured SG and intact SG. For the ruptured SG, the liquid mass went from 106,000 lbm to 89,707 lbm. The current licensing basis for ruptured SGs is an average of initial mass and initial stuck open 10 percent atmospheric dump valve (ADV) phase. The smaller mass is conservative to increase the iodine activity for the dose consequences. The intact SGs changed from 118,500 lbm/SG to 89,707 lbm/SG. The key references are CN-CRA-05-53, Revision 1 (Reference 13), and ASTAP-14-013, Table I-2 and Item 16. The minimum mass is initial mass of the SG following the reactor trip.

The tube leakage rate is the same as the MSLB.

The break flow from the RCS into ruptured SG values are similar to the steam releases from the ruptured SG and intact SGs in references CN-CRA-05-54, Revision 0, Appendix B, Table B-3 (Reference 21), and ASTAP-14-013 Table I-2, Items 9 and 10, Table I-2.2. The results of the analysis were consistent between the calculation files and license amendment.

Loss of Load

(Assumptions used in analysis reside in the notes in calculation File 14078104-C-M-00015 Revision 0, Reference 32)

The thermal hydraulic parameters of interest for LOL are the RCS mass, primary to secondary SG tube leakage, initial and minimum SG liquid mass, and steam releases.

The RCS mass changed from 499,500 lbm to 446,486 lbm and the change was documented in ASTAP-14-013. The notes are captured in the MSLB notes.

The tube leakage rate is the same as the MSLB.

The intact SGs change from 81,500 lbm/SG to 92,301 lbm/SG. The faulted SG was captured in ASTAP-14-013 Table I-1, Item 14 and CN-CRA-08-16, Revision 0, Appendix A Table A-2, A-3 (Reference 22). The mass of the SGs are rounded up and a 10% uncertainty is added based on WEC assumptions to add conservatism. For the intact SGs, the masses were captured in CN-CRA-05-32, Revision 1 Tables 2-1 and 2-2, Case 1-7 and 2-7 (Reference 18). The calculated value was decreased by 10% per WEC suggestion to add conservatism.

The steam releases changed for the multiple time frames. From 0 to 2 hours, the steam releases changed from 656,000 lbm to 651,000 lbm. For 2 to 8 hours, the steam releases changed from 1,035,000 lbm to 1,023,000 lbm. Additionally, an 8 to 10.73 hour time frame was added with the same steam releases as the 2 to 8 hour time frame. WCAP-16985, Revision 2, Section 6.4.1.1 states that the mass of the environmental steam releases for the LOL event bound all Condition II events, as well as the steam releases following an LRA and CREA.

The LRA and CREA were based on the LOL references and thermal hydraulic parameters. See the discussion for the LOL event for the thermal hydraulic parameters for LRA and CREA.

Design Basis Accident Radiological Consequence Analyses

The NRC staff reviewed documentation related to the UFSAR Chapter 15 Radiological Consequence Analysis for the MSLB, SGTR, and LOCA for the proposed AST. Specifically, the NRC staff reviewed the RCS activity inputs for the MSLB and SGTR accidents, and reviewed the inputs and methodology for the LOCA back leakage into the refueling water storage tank (RWST) and miscellaneous equipment drain tank. In addition, the NRC staff reviewed the RADTRAD computer code used for the LOCA back leakage into the RWST and miscellaneous equipment drain tank pathways. The NRC staff reviewed the following documents to obtain the inputs and methodology discussed above:

Calculation Number / Revision	Title
14078104-C-M-00007, R0 (Reference 24)	Composite Equilibrium Reactor Core Isotopic Inventory Assuming an Initial U-235 Enrichment of 4.2% to 5%.
14078104-C-M-00008, R0 (Reference 25)	Primary and Secondary Coolant Design and Technical Specification Activity Concentrations Including Pre-Accident Iodine Spike Concentrations and Accident-Initiated Iodine Spike Appearance Rates.
14078104-C-M-00012, R0 (Reference 29)	Site Boundary and Control Room Doses following a Main Steam Line Break using Alternative Source Terms.
14078104-C-M-00014, R0 (Reference 31)	Site Boundary and Control Room Doses following a Control Rod Ejection Accident using Alternative Source Terms.
14078104-C-M-00015, R0 (Reference 32)	Site Boundary and Control Room Doses Following a SG Tube Rupture Using Alternative Source Terms.
14078104-C-M-00024, R0 (Reference 41)	Post-LOCA Iodine Releases and Gaseous Venting Rates from the Refueling Water Storage Tank.
14078104-C-M-00025, R0 (Reference 42)	Post-LOCA Iodine Releases and Gaseous Venting Rates from the Mechanical Equipment Drains Tank (MEDT).
14078104-C-M-00026, R1 (Reference 43)	Site Boundary and Control Room Doses following a Loss of Coolant Accident using Alternative Source Terms.

3.0 SUMMARY

During the audit, the NRC staff reviewed the underlying PG&E and WEC engineering calculations. The analytical technique, inputs, and assumptions used in the PG&E and WEC calculations were determined to be conservative and appropriate for UFSAR Chapter 15 accident thermal hydraulic analyses. Based upon this audit, the NRC staff determined that the proposed thermal hydraulic parameters for the UFSAR Chapter 15 Accident analyses appear to be acceptable.

In the submittal, PG&E proposed new gas gap fractions for the AST. The analytical technique, inputs, and assumptions used in the PG&E calculations were found to be conservative, appropriate, and consistent with RG 1.183, Revision 0 and Draft Guide 1199. Based upon this audit, the NRC staff finds the proposed multipliers on the RG 1.183 gap inventories to be reasonable.

4.0 REFERENCES

1. Welsch, J. M., Pacific Gas and Electric Company, letter to U.S. Nuclear Regulatory Commission, "License Amendment Request 15-03, 'Application of Alternative Source Term,'" dated June 17, 2015 (ADAMS Accession No. ML15176A539).

2. Allen, B. S., Pacific Gas and Electric Company, letter to U.S. Nuclear Regulatory Commission, "Supplement to License Amendment Request 15-03, 'Applicaiton of Alternative Source Term,'" dated August 31, 2015 (ADAMS Accession No. ML15243A363).

3. Allen, B. S., Pacific Gas and Electric Company, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding License Amendment Request 15-03, 'Applicaiton of Alternative Source Term,'" dated October 22, 2015 (ADAMS Accession No. ML15295A470).

4. Allen, B. S., Pacific Gas and Electric Company, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding License Amendment Request 15-03, 'Applicaiton of Alternative Source Term,'" dated November 2, 2015 (ADAMS Accession No. ML15321A235).

5. Allen, B. S., Pacific Gas and Electric Company, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding License Amendment Request 15-03, 'Applicaiton of Alternative Source Term,'" dated November 6, 2015 (ADAMS Accession No. ML15310A522).

6. Halpin, E. D., Pacific Gas and Electric Company, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding License Amendment Request 15-03, 'Applicaiton of Alternative Source Term,'" dated December 17, 2015 (ADAMS Accession No. ML16004A363).

7. Lingam, S. P., U.S. Nuclear Regulatory Commission, letter to Edward Halpin, Pacific Gas and Electric Company, "Diablo Canyon Power Plant, Units 1 and 2 – Regulatory Audit Plans for January 12-14. 2016, Audit at Westinghouse Facility in Rockville, MD, In Support of Alternative Source Term, and BEACON Power Distribution Monitoring System Methodology License Amendment Requests (CAC NOS MF6399 and MF6400, and MF6120 and MF6121)," (ADAMS Accession No. ML15355A157).

8. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

9. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," October 2009 (ADAMS Accession No. ML090960464).

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

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

12. CN-CRA-05-51, "Diablo Canyon Replacement Steam Generator Project: LOCA Containment Integrity Analysis," DC 6021773-8-1.

13. CN-CRA-05-53, "Steam Generator Tube Rupture Dose Input Analysis for the Diablo Canyon Units 1 and 2 Replacement Steam Generator Program," DC-6021773-10-1.

14. CN-LIS-06-15 Rev. 2, "Diablo Canyon Units 1 and 2 (PGE/PEG) Replacement Steam Generator (RSG) Project Post-LOCA Calculations," DC 6021773-22-1.

15. CN-CRA-12-5, "Diablo Canyon Containment Integrity Reanalysis to Address CFCU Fan Flow and NSAL-11-5 Issues."

16. CN-CRA-13-19 "Diablo Canyon Units 1 and 2 Containment Integrity Analysis in Support of the Alternate Source Term Project."

17. CN-TA-05-137, "Diablo Canyon Units 1 and 2 RSG – Hot Zero Power Steam Line Break Core Response Analysis."

18. CN-CRA-05-32, "GENF Runs for Diablo Canyon Units 1 and 2 to Support Replacement Steam Generator Project," DC-6021773-3-1.

19. CN-CRA-05-46," Diablo Canyon Steam Release for Dose for RSG Program."

20. CN-CRA-06-54 Rev. 1, "Diablo Canyon Steamline Break Mass/Energy Release Summary for RSG Project," DC 6021773-18-1.

21. CN-CRA-05-54, "Steam Generator Tube Rupture Radiological Consequences for the Diablo Canyon Units 1 & 2 Replacement Steam Generator Program," DC-6021773-11-1.

22. CN-CRA-08-16, "Diablo Canyon Steam Release for Dose to Support Increased Operating Range Program," DC 6021773-4-2.

23. CN-TA-01-27, "Diablo Canyon Power Plant Unit 2 Cycle 11, Reload Safety Evaluation."

24. 14078104-C-M-00007, Revision 0, "Composite Equilibrium Reactor Core Isotopic Inventory Assuming an Initial U-235 Enrichment of 4.2% to 5%."

25. 14078104-C-M-00008, Revision 0, "Primary and Secondary Coolant Design and Technical Specification Activity Concentrations Including Pre-Accident Iodine Spike Concentrations and Accident-Initiated Iodine Spike Appearance Rates."

26. 14078104-C-M-00009, Revision 0, "Determination of the RHR Initiation Time for Non-LOCA Events."

27. 14078104-C-M-00010, Revision 0, "Technical Support Center Doses following a Loss-of-Coolant Accident (LOCA) using Alternative Source Terms."

28. 14078104-C-M-00011, Revision 1, "Site Boundary and Control Room Doses following a Fuel Handling Accident in Containment or Fuel Handling Building using Alternative Source Terms."

29. 14078104-C-M-00012, Revision 0, "Site Boundary and Control Room Doses following a Main Steam Line Break using Alternative Source Terms."

30. 14078104-C-M-00013, Revision 1, "Site Boundary and Control Room Doses following a Locked Rotor Accident using Alternative Source Terms."

31. 14078104-C-M-00014, Revision 0, "Site Boundary and Control Room Doses following a Control Rod Ejection Accident using Alternative Source Terms."

32. 14078104-C-M-00015, Revision 0, "Site Boundary and Control Room Doses Following a Steam Generator Tube Rupture Using Alternative Source Terms."

33. 14078104-C-M-00016, Revision 0, "Site Boundary and Control Room Doses following a Worst Case Condition II Event (Loss of Load), using Alternative Source Terms."

34. 14078104-C-M-00017, Revision 0, "Development of Minimum ECCS and CSS Flow Rates during Post-LOCA Cold Leg Recirculation Using Computer Code FATHOM."

35. 14078104-C-M-00018, Revision 0, "Containment Spray Coverage during Post-LOCA Recirculation Mode."

36. 14078104-C-M-00019, Revision 0, "Post-LOCA Iodine & Aerosol Removal Rates during Injection and Recirculation Spray Modes."

37. 14078104-C-M-00020, Revision 0, "Impact of Containment Spray in the Recirculation Mode on Other Hydraulic Analyses."

38. 14078104-C-M-00021, Revision 0, "POST-LOCA Flow Rate via 12 inch Containment Pressure Relief Pathway Prior to Containment Isolation."

39. 14078104-C-M-00022, Revision 0, "Post-LOCA Containment Mixing Rate Between Sprayed and Unsprayed Regions."

40. 14078104-C-M-00023, Revision 0, "Development of the Ultimate Sump Water pH following a Loss-of-Coolant Accident."

41. 14078104-C-M-00024, Revision 0, "Post-LOCA Iodine Releases and Gaseous Venting Rates from the Refueling Water Storage Tank."

42. 14078104-C-M-00025, Revision 0, "Post-LOCA Iodine Releases and Gaseous Venting Rates from the Mechanical Equipment Drains Tank (MEDT)."

43. 14078104-C-M-00026, Revision 1, "Site Boundary and Control Room Doses following a Loss of Coolant Accident using Alternative Source Terms."

44. ASTAP-14-13, dated June 3, 2014 – Design Input Transmittal (DIT-50497328-5-0), Non-LOCA Steam Releases and Thermal Hydraulic Input Parameters (MSLB/SGTR), Tables I-1 and I-2.

Date of Issuance:

Attachment:
Table 1 – List of Attendees

Table 1: List of Attendees

Name	Affiliation	Contact Number	E-mail
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If you have any questions, please contact me at 301-415-3016 or via e-mail at Balwant.Singal@nrc.gov.

Sincerely,

/RA/

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosure:
Audit Report

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OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DSS/SRXB/BC (A)
NAME	BSingal	JBurkhardt	EOesterle*
DATE	3/7/16	3/4/16	3/14/16
OFFICE	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM	
NAME	RPascarelli	BSingal	
DATE	3/14/16	3/14/16	

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