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

January 9, 2020

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

SUBJECT: DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 234 AND 236 TO REVISE TECHNICAL SPECIFICATION 5.6.5b, "CORE OPERATING LIMITS REPORT (COLR)," FOR FULL SPECTRUM LOSS-OF-COOLANT ACCIDENT METHODOLOGY (EPID L-2018-LLA-0730)

Dear Mr. Welsch:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 234 to Facility Operating License No. DPR-80 and Amendment No. 236 to Facility Operating License No. DPR-82 for the Diablo Canyon Nuclear Power Plant (Diablo Canyon), Units 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated December 26, 2018, as supplemented by letters dated September 23, 2019, and October 24, 2019.

The amendments revise TS 5.6.5b, "Core Operating Limits Report (COLR)," to replace the existing loss-of-coolant accident (LOCA) methodologies with the NRC-approved LOCA methodology contained in WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," that was used for LOCA reanalysis at Diablo Canyon, Units 1 and 2.

Enclosure 3 to this letter contains proprietary information. When separated from Enclosure 3, this document is DECONTROLLED.

The NRC staff has determined that the related safety evaluation contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* Section 2.390, "Public inspections, exemptions, requests for withholding." The proprietary information is indicated by text enclosed within double brackets. The proprietary version of the safety evaluation is provided as Enclosure 3. Accordingly, the NRC staff has also prepared a non-proprietary version of the safety evaluation, which is provided as Enclosure 4.

A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

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

Docket Nos. 50-275 and 50-323

Enclosures:
1. Amendment No. 234 to DPR-80
2. Amendment No. 236 to DPR-82
3. Safety Evaluation (Proprietary)
4. Safety Evaluation (Non-Proprietary)

cc w/o Enclosure 3: Listserv

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

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 234
License No. DPR-80

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (the licensee), dated December 26, 2018, as supplemented by letters dated September 23, 2019, and October 24, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-80 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 234, are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. DPR-80
and Technical Specifications

Date of Issuance: January 9, 2020

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

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 236
License No. DPR-82

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (the licensee), dated December 26, 2018, as supplemented by letters dated September 23, 2019, and October 24, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-82 is hereby amended to read as follows:

   (2) Technical Specifications (SSER 32, Section 8)* and Environmental Protection Plan

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 236, are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. DPR-82
and Technical Specifications

Date of Issuance: January 9, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 234

TO FACILITY OPERATING LICENSE NO. DPR-80

AND LICENSE AMENDMENT NO. 236 TO FACILITY OPERATING LICENSE NO. DPR-82

DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

Replace the following pages of the Facility Operating License Nos. DPR-80 and DPR-82, and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License No. DPR-80

REMOVE	INSERT
3	3

Facility Operating License No. DPR-82

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
5.0-20	5.0-20

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 234 are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Initial Test Program

The Pacific Gas and Electric Company shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Pacific Gas and Electric Company's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of PG&E's Final Safety Analysis Report as amended as being essential;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.

(2) Technical Specifications (SSER 32, Section 8)* and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 236, are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Initial Test Program (SSER 31, Section 4.4.1)

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-10216-P-A, Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification, (Westinghouse Proprietary),
2. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, (Westinghouse Proprietary),
3. WCAP-8385, Power Distribution Control and Load Following Procedures, (Westinghouse Proprietary),
4. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology),"
5. Not used.
6. Not used.
7. Not used.
8. Not used.
9. WCAP-8567-P-A, "Improved Thermal Design Procedure,"
10. WCAP-16045-P-A, "Qualification of the Two Dimensional Transport Code PARAGON," and
11. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology."

(continued)

ENCLOSURE 4
(NON-PROPRIETARY)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 234 AND 236 TO

FACILITY OPERATING LICENSE NOS. DPR-80 AND DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-275 AND 50-323

Proprietary information pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* has been redacted from this document.

Redacted information is identified by blank space enclosed within [[double brackets]].

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 234 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 236 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By letter dated December 26, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19003A196), as supplemented by letters dated September 23, 2019, and October 24, 2019 (ADAMS Accession Nos. ML19266A657 and ML19297H634, respectively), Pacific Gas and Electric Company (PG&E, the licensee) submitted a license amendment request (LAR) to Facility Operating License Nos. DPR-80 and DPR-82 for the Diablo Canyon Nuclear Power Plant (Diablo Canyon, DCPP), Units 1 and 2, respectively.

The amendments would revise Technical Specification (TS) 5.6.5b, "Core Operating Limits Report (COLR)," to replace the existing loss-of-coolant accident (LOCA) methodologies with the U.S. Nuclear Regulatory Commission (NRC, the Commission)-approved LOCA methodology contained in WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," dated November 2016 (ADAMS Package Accession No. ML17277A130), which was used for LOCA reanalysis for Diablo Canyon, Units 1 and 2.

The supplemental letters dated September 23, 2019, and October 24, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 5, 2019 (84 FR 7941).

2.0 REGULATORY EVALUATION

2.1 Description of the Licensee's Proposed Changes

The licensee proposed changes to TS 5.6.5b for Diablo Canyon, Units 1 and 2, to reflect the use of the NRC-approved LOCA methodology.

The current TS 5.6.5b includes the following five NRC-approved LOCA methodologies (numbered as 5.6.5b.4 through 5.6.5b.8):

4. WCAP-10054-P-A, Westinghouse [Westinghouse Electric Company, LLC] Small Break LOCA ECCS [Emergency Core Cooling System] Evaluation Model Using the NOTRUMP Code, August 1985 (Westinghouse Proprietary),

5. WCAP-10054-P-A, Addendum 2, Revision 1, Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model, July 1997 (Westinghouse Proprietary),

6. WCAP-12945-P-A, Westinghouse Code Qualification Document for Best-Estimate Loss of Coolant Analysis, June 1996 (Westinghouse Proprietary),

7. WCAP-12945-P-A, Addendum 1-A, Revision 0, "Method for Satisfying 10 CFR 50.46 Reanalysis Requirements for Best Estimate LOCA Evaluation Models," December 2004. (Westinghouse Proprietary) (Unit 1 Only),

8. WCAP-16009-P-A, Revision 0, Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), January 2005. (Westinghouse Proprietary) (Unit 2 Only),

The proposed revision to TS 5.6.5b would replace these five LOCA methodologies with WCAP-16996-P-A, Revision 1, as shown below.

4. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology),"

5. Not used.

6. Not used.

7. Not used.

8. Not used.

2.2 Regulatory Review

The NRC staff considered the following regulations and guidance during its review of the proposed changes.

Regulations

Under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.92(a), determinations on whether to issue a license amendment request are guided by the considerations that govern the issuance of initial licenses or construction permits to the extent applicable and appropriate. Both the common standards for licenses and construction permits in 10 CFR 50.40(a), and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3), provide that there must be reasonable assurance that the activities at issue will not endanger the health and safety of the public.

The regulations at 10 CFR 50.36 require that TSs include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports.

The regulation at 10 CFR 50.90 requires NRC approval for any modification to, addition to, or deletion from the TSs.

The regulation at 10 CFR 50.36(c)(5) states that administrative controls "are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner."

The following paragraphs of 10 CFR 50.46(b) require in part that:

(1) "Peak cladding temperature." The calculated maximum fuel element cladding temperature shall not exceed 2200 °F [degrees Fahrenheit].

(2) "Maximum cladding oxidation." The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) "Maximum hydrogen generation." The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) "Coolable geometry." Calculated changes in core geometry shall be such that the core remains amenable to cooling.

Guidance Documents

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Section 15.6.5,

Revision 3, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," dated March 2007 (ADAMS Accession No. ML070550016).

- NRC Regulatory Guide (RG) 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," dated May 1989 (ADAMS Accession No. ML003739584).

- NRC RG 1.203, "Transient and Accident Analysis Methods," dated December 2005 (ADAMS Accession No. ML053500170).

- NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 4, 1988 (ADAMS Accession No. ML031200485).

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine whether the proposed changes are consistent with the regulations and guidance discussed in Section 2.0 of this safety evaluation. The staff reviewed the proposed changes to verify that the new LOCA methodology is an approved NRC code and that all limitations and conditions are met, that the licensee appropriately applied the LOCA Evaluation Model (EM) to Diablo Canyon, Units 1 and 2, and that the results meet the acceptance criteria of 10 CFR 50.46(b)(1) through (4).

3.1 Description of FULL SPECTRUM LOCA Methodology

As described in WCAP-16996-P-A, Revision 1, the purpose of the Full Spectrum LOCA (FSLOCA) Methodology EM is to build on the ASTRUM EM, by extending the applicability of the WCOBRA/TRAC Code to include the treatment of small break LOCA (SBLOCA) and intermediate break LOCA (IBLOCA) scenarios. The term "Full Spectrum" specifies that the new EM is intended to resolve the full spectrum of LOCA scenarios that result from a postulated break in the cold leg of a pressurized water reactor (PWR). The break sizes considered in the Westinghouse FSLOCA methodology include any break size in which break flow is beyond the capacity of the normal charging pumps, up to and including a double ended guillotine rupture with a break flow area equal to 2 times the pipe area.

3.2 Limitations and Conditions

The safety evaluation for WCAP-16996-P-A, Revision 1 (ADAMS Package Accession No. ML17207A124) contains 15 limitations and conditions that must be met in order to implement the NRC-approved FSLOCA EM.

A summary of each limitation and condition and how it has been met as stated by the licensee in its application dated December 26, 2018, and the associated NRC staff findings are provided below.

Limitation and Condition 1 – Applicability with Regard to LOCA Transient Phases

> The FSLOCA EM is not approved to demonstrate compliance with 10 CFR 50.46 acceptance criterion (b)(5) related to the long-term cooling.

> The analyses for [DCPP] Units 1 and 2 with the FSLOCA EM are only being used to demonstrate compliance with 10 CFR 50.46 (b)(1) through (b)(4).

Given that the licensee is not using the FSLOCA EM to demonstrate compliance with 10 CFR 50.46(b)(5), the NRC staff finds that the licensee has met the requirements of Limitation and Condition 1.

<u>Limitation and Condition 2 – Applicability with Regard to Type of PWR Plants</u>

> The FSLOCA EM is approved for the analysis of Westinghouse-designed 3-loop and 4-loop pressurized water reactors (PWRs) with cold-side injection. Analyses should be executed consistent with the approved method, or any deviations from the approved method should be described and justified.
>
> DCPP Units 1 and 2 are Westinghouse-designed 4-loop PWRs with cold-side injection, so they are within the NRC approved methodology. The analyses for DCPP Units 1 and 2 utilize the NRC approved FSLOCA methodology, except for the changes discussed in Section 3.2 [of the Enclosure to the letter dated December 26, 2018[1]] which were previously transmitted to the NRC pursuant to 10 CFR 50.46 in [Westinghouse Letter dated July 18, 2018, No.] LTR-NRC-18-30 [ADAMS Accession No. ML19288A174].

These changes are reviewed in Section 3.3 of this safety evaluation.

Given that Diablo Canyon, Units 1 and 2, are Westinghouse-designed 4-loop PWRs with cold-side injection, the NRC staff finds that the FSLOCA EM is applicable to them. In addition, the NRC staff finds that the licensee has appropriately applied the FSLOCA EM with the changes described in Section 3.3 of this safety evaluation. Therefore, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 2.

<u>Limitation and Condition 3 – Applicability for Containment Pressure Modeling</u>

> For Region II, the containment pressure calculation will be executed in a manner consistent with the approved methodology (i.e., the COCO or LOTIC2 model will be based on appropriate plant-specific design parameters and conditions, and engineered safety features which can reduce pressure are modelled). This includes utilizing a plant-specific initial containment temperature, and only taking credit for containment coatings which are qualified and outside of the break zone-of-influence.
>
> The containment pressure calculation for the DCPP Unit 1 and 2, analyses was performed consistent with the NRC approved methodology. [The containment pressure is calculated for each LOCA transient in the analysis using the COCO code. The COCO containment code is integrated into the <u>W</u>COBRA/TRAC-TF2 thermal-hydraulic code.] Appropriate design parameters and conditions were modelled, as were the engineered safety features which can reduce the containment pressure. A minimum initial temperature associated with normal

[1] Referred to as the Enclosure to the License Amendment Request (LAR), hereafter.

full-power operating conditions was modelled, and no coatings were credited on any of the containment structures.

The NRC staff finds that the licensee used the NRC-approved methodology for the Region II containment pressure calculation with appropriate design parameters and conditions. Therefore, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 3.

Limitation and Condition 4 – Decay Heat Modeling

> The decay heat uncertainty multiplier will be **[[** **]]**. The analysis simulations for the FSLOCA EM will not be executed for longer than 10,000 seconds following reactor trip unless the decay heat model is appropriately justified. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region I and Region II analysis results will be provided in the analysis submittal in units of sigma and absolute units.
>
> Consistent with the NRC approved methodology, the decay heat uncertainty multiplier was **[[** **]]** for the DCPP Units 1 and 2, analyses. The analysis simulations were all executed for less than 10,000 seconds following reactor trip. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region I and Region II analysis results have been provided in units of sigma and approximate absolute units in Table 1 [of the Enclosure to the LAR] for DCPP Unit 1 and Table 2 [of the Enclosure to the LAR] for DCPP Unit 2.

The NRC staff finds that the licensee appropriately modeled decay heat per the limitation and condition and reported the resulting sampled values in units of sigma and absolute units for the limiting cases. Therefore, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 4.

Limitation and Condition 5 – Fuel Burnup Limits

> The maximum assembly and rod length-average burnup is limited to **[[** **]]** respectively.
>
> The maximum analyzed assembly and rod length-average burnup is less than or equal to **[[** **]]** respectively, for DCPP Units 1 and 2.

Based on the above, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 5.

Limitation and Condition 6 –WCOBRA/TRAC-TF2 Interface with PAD 5.0

> The fuel performance data for analyses with the FSLOCA EM should be based on the [latest version of an NRC-approved fuel performance code, which is the]

> PAD5 code (at present),[2] which includes the effect of thermal conductivity degradation. The nominal fuel pellet average temperatures and rod internal pressures should be the maximum values, and the generation of all the PAD5 fuel performance data should adhere to the NRC approved PAD5 methodology.
>
> PAD5 fuel performance data is utilized in the DCPP Units 1 and 2 analyses with the FSLOCA EM. The analyzed fuel pellet average temperatures bound the maximum values calculated in accordance with Section 7.5.1 of WCAP-17642-A, Revision 1 [("Westinghouse Performance Analysis and Design Model (PAD5)," November 2017)] and the analyzed rod internal pressures were calculated in accordance with Section 7.5.2 of WCAP-17642-A, Revision 1.

Given that the licensee used the latest NRC-approved fuel performance code (i.e., PAD5) and used appropriate conservative inputs, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 6.

<u>Limitation and Condition 7 – Interfacial Drag Uncertainty in Region I Analyses</u>

> The YDRAG uncertainty parameter should be **[[**
>
> **]]**
>
> Consistent with the NRC approved methodology, the YDRAG uncertainty parameter was **[[**
>
> **]]** for the DCPP Units 1 and 2, Region I analyses.

The NRC staff finds that the licensee appropriately used the specified interfacial drag uncertainty parameter as noted above and, therefore, meets the requirement of Limitation and Condition 7.

<u>Limitation and Condition 8 – Biased Uncertainty Contributors in Region I Analyses</u>

> The **[[**
>
> **]]**
>
> Consistent with the NRC approved methodology, the **[[**
>
> **]]** for the DCPP Units 1 and 2 Region I analyses.

The NRC staff finds that the licensee appropriately used the specified biased uncertainty parameters as noted above and, therefore, meets the requirement of Limitation and Condition 8.

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[2] WCAP-17642-NP-A, Revision 1, "Westinghouse Performance Analysis and Design Model (PAD5)," November 2017 (ADAMS Accession No. ML17338A396).

Limitation and Condition 9 – Effect of Bias in Applications for Region I

> For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to confirm that the **[[**
> **]]** for the plant design being analyzed. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.
>
> DCPP Units 1 and 2, are both Westinghouse-designed 4-loop PWRs. The requested sensitivity study was performed for a 4-loop Westinghouse-designed PWR and is discussed in [Westinghouse letter dated July 13, 2018, No.] LTR-NRC-18-50....

The NRC staff reviewed the attachment to Westinghouse letter No. LTR-NRC-18-50, "Information to Satisfy the FULL SPECTRUM LOCA (FSLOCA) Evaluation Methodology Plant Type Limitations and Conditions for 4-loop Westinghouse Pressurized Water Reactors (PWRs)" (not publicly available). This document describes the sensitivity studies done on the selected parameters and demonstrates that **[[**

]] Therefore, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 9.

Limitation and Condition 10 – Boundary Between Region I and Region II Breaks

> For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to: 1) demonstrate that no unexplained behavior occurs in the predicted safety criteria across the region boundary, and 2) ensure that the **[[**
> **[[**
> **]]** must cover the equivalent 2 to 4-inch break range using reactor coolant system (RCS)-volume scaling relative to the demonstration plant. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.
>
> Additionally, the minimum sampled break area for the analysis of Region II should be 1 ft^2 [square foot].
>
> DCPP Units 1 and 2, are both Westinghouse-designed 4-loop PWRs. The requested sensitivity study was performed for a 4-loop Westinghouse-designed PWR and is discussed in [Westinghouse letter dated July 13, 2018, No.] LTR-NRC-18-50.
>
> The minimum sampled break area for the DCPP Units 1 and 2 Region II analyses was 1 ft^2.

The NRC staff reviewed the attachment to Westinghouse letter No. LTR-NRC-18-50, "Information to Satisfy the FULL SPECTRUM LOCA (FSLOCA) Evaluation Methodology Plant Type Limitations and Conditions for 4-loop Westinghouse Pressurized Water Reactors (PWRs)" (not publicly available). This document describes the sensitivities performed to demonstrate

that the boundary between Region I and Region II breaks is appropriate for a 4-loop Westinghouse-designed plant.

The limitation and condition specifies that plants with larger RCS fluid volumes than the 3-loop plant test example in WCAP-16996-P-A, Revision 1, should cover the same 2-to-4 inch range using break area to RCS volume scaling to ensure that the break range is preserved and not artificially truncated. The licensee applied **[[**

]]

In order to demonstrate that no unexplained behavior occurs in the predicted safety criteria across the region boundary, the analysis examined breaks **[[**

]] were not considered, the NRC staff finds that these would be bounded by the LBLOCA (Region II) similar to the IBLOCAs.

In addition, the Region II analysis considers a minimum break area of 1.0 ft^2 consistent with the requirement in the limitation and condition.

Therefore, the NRC staff finds that the requirements of Limitation and Condition 10 are met as the licensee performed the necessary sensitivity study to determine the appropriate break size range for Region I and boundary between Region I and Region II.

<u>Limitation and Condition 11 - [[ ]] in Uncertainty Analyses for Region II and Documentation of Reanalysis Results for Region I and Region II</u>

There are various aspects of this Limitation and Condition, which are summarized below:

1) The **[[** **]]** the Region I and Region II analysis seeds, and the analysis inputs will be declared and documented prior to performing the Region I and Region II uncertainty analyses. The **[[** **]]** and the Region I and Region II analyses seeds will not be changed throughout the remainder of the analysis once they have been declared and documented.

2) If the analysis inputs are changed after they have been declared and documented, for the intended purpose of demonstrating compliance with the applicable acceptance criteria, then the changes and associated

rationale for the changes will be provided in the analysis submittal. Additionally, the preliminary values for peak cladding temperature (PCT), maximum local oxidation (MLO), and core-wide oxidation (CWO) which caused the input changes will be provided. These preliminary values are not subject to Appendix B verification, and archival of the supporting information for these preliminary values is not required.

3) Plant operating ranges which are sampled within the uncertainty analysis will be provided in the analysis submittal for both regions.

This Limitation and Condition was met for the DCPP Units 1 and 2 analyses as follows:

1) The **[[ ]]** the Region I and Region II analysis seeds, and the analysis inputs were declared and documented prior to performing the Region I and Region II uncertainty analyses. The **[[ ]]** and the Region I and Region II analyses seeds were not changed once they were declared and documented.

2) The analysis inputs were not changed once they were declared and documented.

3) The plant operating ranges which were sampled within the uncertainty analyses are provided for DCPP Units 1 and 2, in Table 3 [of the Enclosure to the LAR].

During the regulatory audit[3], the NRC staff reviewed the documentation containing the analysis seeds and inputs. Given that the licensee has declared and documented the appropriate inputs and did not change these values once declared and documented, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 11.

<u>Limitation and Condition 12 – Steam Generator Heat Removal During SBLOCAs</u>

The plant-specific dynamic pressure loss from the steam generator secondary-side to the main steam safety valves must be adequately accounted for in analysis with the FSLOCA EM.

A bounding plant-specific dynamic pressure loss from the steam generator secondary-side to the main steam safety valves was modelled in the DCPP Units 1 and 2 analyses.

During the regulatory audit[3], the NRC staff observed the use of a constant value to account for the dynamic pressure loss. Given the use of a bounding dynamic pressure loss, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 12.

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[3] An on-line regulatory audit was conducted from April 29, 2019, to October 2, 2019, in support of the NRC staff's review of the subject LAR. An audit summary, documenting the results of the review, was issued by letter dated October 11, 2019 (ADAMS Accession No. ML19276F243).

Limitation and Condition 13 – Upper Head Spray Nozzle Loss Coefficient

> In plant-specific models for analysis with the FSLOCA EM: 1) the [[ ]] and 2) the [[ ]]

> The [[ ]] in the analyses for DCPP Units 1 and 2. The [[ ]] in the analyses.

Based on the above, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 13.

Limitation and Condition 14 – Correlation for Oxidation

> For analyses with the FSLOCA EM to demonstrate compliance against the current 10 CFR 50.46 oxidation criterion, the transient time-at-temperature will be converted to an equivalent cladding reacted (ECR) using either the Baker-Just or the Cathcart-Pawel correlation. In either case, the pre-transient corrosion will be summed with the LOCA transient oxidation. If the Cathcart-Pawel correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 13 percent limit. If the Baker-Just correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 17 percent limit.

> For the DCPP Unit[s] 1 and 2, analyses, the Baker-Just correlation was used to convert the LOCA transient time-at-temperature to an ECR. The resulting LOCA transient ECR was then summed with the pre-existing corrosion for comparison against the 10 CFR 50.46 local oxidation acceptance criterion of 17 percent.

The NRC staff finds that by using the Baker-Just correlation, converting to an ECR, and accounting for pre-existing corrosion, the licensee has met the requirements of Limitation and Condition 14.

Limitation and Condition 15 – LOOP versus Offsite Power Available (OPA) Treatment in Uncertainty Analyses for Region II

> The Region II analysis will be executed twice; once assuming loss-of-offsite power (LOOP) and once assuming offsite power available (OPA). The results from both analysis executions should be shown to be in compliance with the 10 CFR 50.46 acceptance criteria.

> The [[ ]]

> The Region II uncertainty analyses for DCPP Units 1 and 2, were performed twice; once assuming a LOOP and once assuming OPA. The results from both analyses that were performed are in compliance with the 10 CFR 50.46 acceptance criteria, and the LOOP configuration is the limiting configuration (see Tables 4 and 5 [of the Enclosure to the LAR]).

The [[

]]

Given that the licensee has performed the Region II analysis for both LOOP and OPA and that the results from both are in compliance with the acceptance criteria in 10 CFR 50.46(b)(1) through (b)(4), the NRC staff finds that the licensee has met the requirements of Limitation and Condition 15.

3.3 Changes and Corrections to the Approved FSLOCA EM

Since the safety evaluation was issued on the FSLOCA methodology (WCAP-16996-P-A, Revision 1), several changes and corrections have been made to the FSLOCA EM. The final code version in the NRC-approved FSLOCA EM was WCOBRA/TRAC-TF2 Version 1.3, however, Version 1.4 was utilized in the analyses for Diablo Canyon, Units 1 and 2. The changes from Version 1.3 to Version 1.4 were described by the licensee in the LAR as follows:

> General Code Maintenance
>
> Various updates were made to enhance the usability of the code. These updates do not impact the code-calculated results.
>
> Enhancement to Pump Momentum Equation at Low Pump Speed
>
> A change was made to allow for transition from a homogeneous flow condition to a condition that allows slip at the pump impeller (and counter-current flow) when reactor coolant pump speed approaches zero. Under this condition, the PIPE momentum equations are solved rather than the simplified PUMP momentum equation at the pump-impeller cell face. A smooth transition between the two sets of momentum equations is ensured by ramping the interfacial drag during the transition. This improvement in the pump component momentum equations can allow liquid to flow back from the cold leg into the crossover leg, which has a negligible impact on the Region II analyses and a penalizing (i.e., conservative) impact on the Region I analyses.
>
> Conservation of Non-Condensable Gas
>
> An imbalance in the non-condensable gas mass that could occur under certain conditions (most likely in loop components during accumulator injection) in WCOBRA/TRAC-TF2 Version 1.3 was identified. Since a vapor/liquid mixture cannot exist below 32°F and WCOBRA/TRAC-TF2 does not implement the vapor property functions for temperatures below 32°F, whenever the energy content of the combined gas phase falls below 32°F, the Version 1.3 code reset the gas phase temperature to be 32°F, causing unbalanced mass and energy for the combined gas phase. This imbalance was corrected in WCOBRA/TRAC-TF2 Version 1.4, and the analyses for DCPP Units 1 and 2 were performed with the corrected code version.

The licensee also noted in the LAR that there were various errata identified in WCAP-16996-P-A, Revision 1. These errata do not impact the WCOBRA/TRAC-TF2 coding.

As such, there is no impact on the Diablo Canyon, Units 1 and 2 analyses.

The NRC staff finds that changes made due to general code maintenance are not expected to impact code-calculated results, and that changes to the pump momentum equation and the conservation of non-condensable gas are appropriate and acceptable.

By letter dated September 23, 2019, the licensee updated the LAR to correct two errors that were discovered by Westinghouse after the submittal of the LAR. The licensee stated that the first error was in the gamma energy redistribution multiplier on the hot rod and hot assembly power used for the Diablo Canyon, Units 1 and 2, analyses. The second error involved several inputs in the containment model used for the Diablo Canyon, Unit 1, Region II analysis that were improperly aligned in the computer code input file and were read by the WCOBRA/TRAC-TF2 computer code incorrectly. For both errors, the licensee applied a penalty on the calculated PCT. These errors and penalties are discussed below.

For the first error, the licensee stated that the error is in the gamma redistribution multiplier on the hot rod and hot assembly power (FGAMMA) used for the Diablo Canyon, Units 1 and 2, analyses. The error results in an underestimation of the hot rod and hot assembly power by up to 5 percent. The underestimation of the hot rod and hot assembly power results in an underprediction in the calculated PCT results for the Diablo Canyon, Units 1 and 2, analyses. The error results in a 0 percent to 5 percent deficiency in the modeled hot rod and hot assembly rod linear heat rates on a run specific basis, depending on the as-sampled value for the uncertainty. The correction of this error was assessed by Westinghouse for Diablo Canyon, Units 1 and 2, for both SBLOCA (Region 1) and LBLOCA (Region II). For SBLOCA, the PCT impact was assessed using a run-specific PCT versus linear heat rate relationships and the run-specific hot rod and hot assembly linear heat rate increase that would result from the error correction. For LBLOCA, parametric sensitivity studies, derived from a subset of uncertainty analysis simulations covering various design features and fuel arrays, were examined to determine the sensitivity of the analysis results to the error correction. The assessment of the error resulted in penalty factors applied to Diablo Canyon, Units 1 and 2, for both small- and large-break LOCAs. The penalties ranged from 9 °F (Unit 2 SBLOCA) to 31 °F (Units 1 and 2 LBLOCA). The NRC staff determined that the PCT penalties were reasonable. In addition, given the large margin (487 °F for the limiting case) to the 10 CFR 50.46(b)(1) PCT limit of 2,200 °F, the staff finds the PCT penalties acceptable in this case. If the FSLOCA calculations are performed again in the future, a corrected version of FSLOCA should be applied or the PCT penalties should be reassessed.

The licensee stated that the MLO and CWO were confirmed to maintain compliance with the 10 CFR 50.46 acceptance criteria with the error correction. In terms of CWO, the NRC staff finds this acceptable as the error correction has only a limited impact on the power modeled for a single assembly in the core and, as such, there is a negligible impact of the error correction on the system thermal-hydraulic response during the postulated LOCA. For the MLO, the NRC staff finds this acceptable as the increased power to the hot rod and hot assembly is small and the overall temperatures are low enough that any additional oxidation due to the error correction would be expected to be minimal.

The second error is related to inputs in the containment model used for the Diablo Canyon, Unit 1, LBLOCA. The licensee stated that these input errors impact the initial temperature for an unheated structure as well as the spray modeling which influence the back-pressure boundary condition of a LBLOCA thermal-hydraulic response. These errors were evaluated by

Westinghouse by re-executing a subset of the Diablo Canyon, Unit 1, LBLOCA (Region II) analysis with the input issues corrected. The correction of the input errors was estimated to increase the Diablo Canyon, Unit 1, LBLOCA (Region II) analysis PCT by 6 °F.

The NRC staff determined that the PCT penalty was reasonable. In addition, given the large margin (487 °F for the limiting case) to the 10 CFR 50.46(b)(1) PCT limit of 2,200 °F, the NRC staff finds the PCT penalty acceptable in this case. If the FSLOCA calculations are performed again in the future, a corrected version of FSLOCA should be applied, or the PCT penalties should be reassessed.

3.4 Results and Compliance with 10 CFR 50.46

The licensee presented the results for PCT, MLO, and CWO in Tables 4 and 5 in the Enclosure to the LAR for Diablo Canyon, Units 1 and 2, respectively. More details on the analysis results were provided for information only in the Diablo Canyon Updated Final Safety Analysis Report (UFSAR) update pages in Attachment 4 to the Enclosure to the LAR.

To demonstrate compliance with 10 CFR 50.46(b)(1) through (b)(4), the following criteria must be met:

1. PCT;
2. Maximum cladding oxidation;
3. Maximum hydrogen generation; and
4. Coolable geometry.

Each of the above four 10 CFR 50.46 criteria is discussed below.

Note that the FSLOCA EM does not address 10 CFR 50.46 (b)(5), "Long-term cooling." Long-term cooling is dependent on the demonstration of the continued delivery of cooling water to the core. The licensee stated that the actions that are currently in place to maintain long-term cooling are not impacted by the application of the NRC-approved FSLOCA EM.

Peak Cladding Temperature

The requirement of 10 CFR 50.46(b)(1) states that "The calculated maximum fuel element cladding temperature shall not exceed 2200 °F." The licensee stated that the analysis for PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level, and given that the resulting PCT is less than 2,200 °F, the analyses with the FSLOCA EM confirm that 10 CFR 50.46 acceptance criterion (b)(1) is satisfied. The licensee presented the results in Tables 4 and 5 of Attachment 1 to the Enclosure to the letter dated September 23, 2019 for Diablo Canyon, Units 1 and 2, respectively.

Given that the maximum calculated PCT is below the 2,200 °F PCT limit, the NRC staff finds that the acceptance criterion of 10 CFR 50.46(b)(1) is met.

Maximum Cladding Oxidation

The requirements of 10 CFR 50.46(b)(2) state, in part, that "The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

The licensee stated that the analysis for MLO corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Since the resulting MLO is less than 17 percent when converting the time-at-temperature to an equivalent cladding reacted using the Baker-Just correlation and adding the pre-transient corrosion, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2) is satisfied. The licensee presented the results in Tables 4 and 5 of the Enclosure to the LAR for Diablo Canyon, Units 1 and 2, respectively.

Given that the resulting MLO is below the 17 percent limit, the NRC staff finds that the acceptance criterion of 10 CFR 50.46(b)(2) is met.

Maximum Hydrogen Generation

The requirement of 10 CFR 50.46(b)(3) states that "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

The licensee stated that the analysis for CWO corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. Since the resulting CWO is less than 1 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3) is satisfied. The licensee presented the results in Tables 4 and 5 of the Enclosure to the LAR for Diablo Canyon, Units 1 and 2, respectively.

Given that the resulting CWO is below the 1 percent limit, the NRC staff finds that the acceptance criterion of 10 CFR 50.46(b)(3) is met.

Coolable Geometry

The requirement of 10 CFR 50.46(b)(4) states that "Calculated changes in core geometry shall be such that the core remains amenable to cooling." The licensee stated that this criterion is met by demonstrating compliance with criteria 10 CFR 50.46(b)(1), (b)(2), and (b)(3), and by ensuring that fuel assembly grid deformation due to combined LOCA and seismic loads is specifically addressed.

Section 32.1 of the NRC-approved FSLOCA EM documents that the effects of LOCA and seismic loads on the core geometry do not need to be considered unless fuel assembly grid deformation extends beyond the core periphery (i.e., deformation in a fuel assembly with no sides adjacent to the core baffle plates). Inboard grid deformation due to the combined LOCA and seismic loads was calculated to not occur for Diablo Canyon, Units 1 and 2, based on performance of the fuel assembly structural analysis that, in turn, was based on a pipe break that considers the application of leak-before-break as approved in License Amendment Nos. 221 and 223 for Diablo Canyon, Units 1 and 2, respectively (ADAMS Accession No. ML15281A164).

Given that the criteria in 10 CFR 50.46(b)(1), (b)(2), and (b)(3) are met and that fuel assembly grid deformation due to combined LOCA and seismic loads is specifically addressed, the NRC staff finds that the acceptance criterion of 10 CFR 50.46(b)(4) is met.

3.5 Comparison to Results from Analyses of Record

The licensee provided a comparison between the results from the Analysis of Record (AOR) and the proposed FSLOCA EM.

The SBLOCA AOR results for Diablo Canyon, Units 1 and 2, originate from analyses performed with an EM developed according to 10 CFR Part 50, Appendix K, "ECCS Evaluation Models." The FSLOCA EM is a best-estimate plus uncertainty method, which relaxes some of the conservatisms required for EMs developed to 10 CFR Part 50, Appendix K. The licensee stated that the reduction in the SBLOCA analysis PCT results is primarily attributed to the more realistic treatment of various phenomena within the FSLOCA EM, most notably the decay heat modeling; the prior analyses assumed decay heat based on 1.2 times the American National Standards Institute/American Nuclear Society (ANSI/ANS) 5.1-1971, whereas the analyses with the FSLOCA EM are based on ANSI/ANS 5.1-1979.

The licensee stated that the SBLOCA MLO results are substantially higher for the analyses with the FSLOCA EM than in the AORs. This is primarily attributed to the AOR results only considering the oxidation accrued during the LOCA transient, whereas the results from the FSLOCA EM include the steady-state corrosion. The CWO results for both the AORs and the analyses with the FSLOCA EM indicate little to no CWO during the SBLOCA transient.

The NRC staff agrees with the licensees' explanations for the differences in SBLOCA results between the AOR and FSLOCA EM, and finds that the differences are reasonable and expected.

The licensee stated that the reduction in the LBLOCA PCT results is attributed to differences between the FSLOCA EM and the legacy EMs, including, but not limited to, the following: improvements to the statistical analysis method (elimination of the superposition penalty (DCPP Unit 1 only) and larger sample size (both units)); improvements to the fuel temperature calculation; improvements to the axial power shape methodology; and improvements to the swelling, burst, and blockage models.

The licensee stated that the LBLOCA MLO results for the AOR versus the results from analysis with the FSLOCA EM for Diablo Canyon, Unit 1, are relatively similar, while the results for Diablo Canyon, Unit 2, are much higher for the analysis with the FSLOCA EM. The AOR results only considered the oxidation accrued during the LOCA transient, whereas the results from the FSLOCA EM included the steady-state corrosion. The increase in the MLO for Diablo Canyon, Unit 2, is primarily attributed to the contribution of the steady-state corrosion. The Diablo Canyon, Unit 1, AOR MLO result is based on a LOCA transient with a cladding temperature in excess of 2200 °F for which the time-at-temperature has been artificially increased (both artifacts of the AOR evaluation model). The excessive transient cladding temperature and artificial increase in time-at-temperature from the AOR EM tend to offset the inclusion of the steady-state corrosion within the FSLOCA EM, and the resulting MLO difference is therefore much smaller than that observed for Diablo Canyon, Unit 2.

The LBLOCA CWO results for the AOR and the analysis with the FSLOCA EM are relatively low for Diablo Canyon, Unit 2. For Diablo Canyon, Unit 1, the AOR CWO result is based on the same transient as the local oxidation result with a cladding temperature in excess of 2200 °F. As such, the reduction in CWO for the analysis with the FSLOCA EM is largely attributed to the use of a transient with lower temperature, as predicted by differences in the EMs.

Overall, the NRC staff finds that the differences in results (PCT, MLO, and CWO) between the AOR and FSLOCA for the LBLOCA are reasonable and as expected.

3.6 Technical Evaluation Summary

The licensee proposed to modify TS 5.6.5b, "Core Operating Limits Report (COLR)," to replace the existing NRC-approved LOCA methodologies with the NRC-approved LOCA methodology contained in WCAP-16996-P-A, Revision 1. The NRC staff concludes that the proposed TS change is acceptable as it changes from one set of NRC-approved methods to another NRC-approved method. The NRC staff's review has determined that the licensee appropriately applied the FSLOCA EM to Diablo Canyon, Units 1 and 2, and finds that the resulting analysis meets the criteria in 10 CFR 50.46(b)(1) through (4).

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendments on November 26, 2019. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration published in the *Federal Register* on March 5, 2019 (84 FR 7941), and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Robert Beaton, NRR

Date: January 9, 2020

SUBJECT: DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 234 AND 236 TO REVISE TECHNICAL SPECIFICATION 5.6.5b, "CORE OPERATING LIMITS REPORT (COLR)," FOR FULL SPECTRUM LOSS-OF-COOLANT ACCIDENT METHODOLOGY (EPID L-2018-LLA-0730) DATED JANUARY 9, 2020

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ADAMS Accession Nos.
ML19312C440 Non-public (Proprietary)
ML19316A109-Public (Non-proprietary) ***By memo dated 11/14/19**

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DATE	11/26/2019	11/26/2019	12/03/19	11/14/2019
OFFICE	NSS/DSS/SFNB/BC	OGC*	NRR/DORL/LPL4/BC	NRR/DORL/LPL4/PM
NAME	RLukes	RAugustus	JDixon-Herrity	BSingal
DATE	12/09/2019	01/08/2020	01/09/2020	01/09/2020

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