



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

March 9, 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, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING REVISION TO UPDATED FINAL SAFETY ANALYSIS REPORT FOR BEST ESTIMATE ANALYZER FOR THE CORE OPERATIONS-NUCLEAR POWER DISTRIBUTION MONITORING SYSTEM METHODOLOGY AND TECHNICAL SPECIFICATION 5.6.5 (CAC NOS. MF6120 AND MF6121)

Dear Mr. Halpin:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 224 to Facility Operating License No. DPR-80 and Amendment No. 226 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant (DCPP), Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Updated Final Safety Analysis Report (UFSAR) and Technical Specifications (TSs) in response to your application dated April 16, 2015.

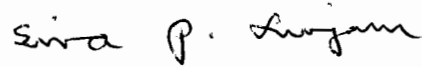
The amendments revise the Best Estimate Analyzer for the Core Operations-Nuclear (BEACON) power distribution monitoring system methodology described in the DCPP UFSAR Section 4.3.2.2, "Power Distribution," to the method described in the Westinghouse Electric Company LLC (Westinghouse) proprietary topical report (TR) WCAP-12472-P-A, Addendum 4, "BEACON Core Monitoring and Operation Support System." The amendments also revise TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," Section b to replace Westinghouse proprietary TR WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," with NRC-approved proprietary TR WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," and NRC-approved proprietary TR WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology."

E. Halpin

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A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,



Siva P. Lingam, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosures:

1. Amendment No. 224 to DPR-80
2. Amendment No. 226 to DPR-82
3. Safety Evaluation

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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 No. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 224
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 April 16, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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.

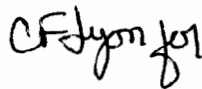
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. DPR-80 is hereby amended to read as follows:

- (2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 224, 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 prior to MODE 4 at the start of Cycle 21. Implementation of the amendment shall also include revision of the Updated Final Safety Analysis Report as described in the licensee's letter dated April 16, 2015.

FOR THE NUCLEAR REGULATORY COMMISSION



Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
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: March 9, 2016



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 No. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 226
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 April 16, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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.

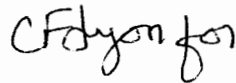
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. DPR-82 is hereby amended to read as follows:

- (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. 226, 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 prior to MODE 4 at the start of Cycle 20. Implementation of the amendment shall also include revision of the Updated Final Safety Analysis Report as described in the licensee's letter dated April 16, 2015.

FOR THE NUCLEAR REGULATORY COMMISSION



Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
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: March 9, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 224

TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 226 TO FACILITY OPERATING LICENSE NO. DPR-82

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-

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Facility Operating License No. DPR-82

REMOVE

INSERT

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Technical Specifications

REMOVE

INSERT

5.0-20

5.0-20

5.0-21

5.0-21

- (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. 224 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. 226, 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_0 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-10054-P-A, Westinghouse Small Break LOCA ECCS 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),
 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)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

- a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing, Low Temperature Overpressure Protection (LTOP) arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
 - 1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and
 - 2. Specification 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

(continued)



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 224 TO FACILITY OPERATING LICENSE NO. DPR-80
AND AMENDMENT NO. 226 TO FACILITY OPERATING LICENSE NO. DPR-82
PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By letter dated April 16, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15107A333), Pacific Gas and Electric Company (the licensee) submitted a license amendment request (LAR) to revise the Best Estimate Analyzer for the Core Operations-Nuclear (BEACON) power distribution monitoring system methodology described in the Updated Final Safety Analysis Report (UFSAR) Section 4.3.2.2, "Power Distribution," to the method described in the Westinghouse Electric Company LLC (Westinghouse) proprietary topical report (TR) WCAP-12472-P-A, Addendum 4, "BEACON Core Monitoring and Operation Support System," September 2012 (ADAMS Accession No. ML12270A386 for non-proprietary version) for Diablo Canyon Power Plant (DCPP), Units 1 and 2. This LAR also proposes to revise Technical Specification (TS) 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," Section b to replace Westinghouse proprietary TR WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988, with U.S. Nuclear Regulatory Commission (NRC)-approved proprietary TR WCAP-16045-P-A, Revision 0, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004 (ADAMS Accession No. ML042250322 for non-proprietary version), and NRC-approved proprietary TR WCAP-16045-P-A, Addendum 1-A, Revision 0, "Qualification of the NEXUS Nuclear Data Methodology," August 2007 (ADAMS Accession No. ML072570329 for non-proprietary version).

2.0 REGULATORY EVALUATION

The regulatory requirements and criteria specifically applicable to the proposed license amendment are summarized and addressed below:

NRC Generic Letter 88-16

The guidance in NRC Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988 (ADAMS Accession No. ML031130447), indicates that it is acceptable for licensees to control reactor physics parameter limits by

specifying the calculation methodology. The generic letter indicates that such parameter limits may be removed from TSs and placed in a cycle-specific COLR. The COLR is defined in the TSs, and the Reporting Requirements in the TSs require that a COLR be submitted to the NRC each operating cycle, or each time the COLR is revised. Generic Letter 88-16 also recommends that licensees include, in the TSs, a list of references for the NRC-approved methodologies, which are used to generate the cycle-specific parameter operating limits.

10 CFR 50.34

Section 50.34, "Contents of applications; technical information," of Title 10 to the *Code of Federal Regulations* (10 CFR) Part 50 requires that safety analysis reports include analysis and evaluation of the design and performance of structures, systems, and components for the prevention of accidents and the mitigation of the consequences of accidents. As part of the core reload design process, licensees (or vendors) perform reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle.

To confirm that the analyses remain bounding, licensees confirm that key inputs to the safety analyses (such as the critical power ratio) are conservative with respect to the current design cycle. If key safety analysis parameters are not bounded, a reanalysis or reevaluation of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

10 CFR 50.36

In Section 50.36, "Technical specifications," of 10 CFR Part 50, the Commission established its regulatory requirements related to the content of the TSs. Consistent with 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: 1) safety limits, limiting safety system settings (LSSS), and limiting control settings; 2) limiting conditions for operations (LCOs); 3) surveillance requirements; 4) design features; and 5) administrative controls.

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes

Current TS 5.6.5 Section b states, in part:

10. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores."

Revised TS 5.6.5 Section b would state, in part:

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."

3.2 Technical Evaluation of the Application of the Revised BEACON Methodology

The NRC-approved BEACON methodology described in Westinghouse TR WCAP-12472-P-A Addendum 4, "BEACON Core Monitoring and Operation Support System, Addendum 4", has been reviewed for application to DCP, Units 1 and 2, as summarized below.

The BEACON core monitoring system currently in use at DCP, Units 1 and 2, WCAP-12472-NP-A, Addendum 1-A, "BEACON Core Monitoring and Operations Support System," January 2000 (ADAMS Accession No. ML003678155), employs a predictive three-dimensional nodal model of the reactor core calibrated to an in-core flux map, which is periodically obtained utilizing the movable in-core detector system (MIDS). The resultant nodal power distribution is frequently updated as part of the core follow calculation, and is adjusted for the axial offset measured by the ex-core detectors; the radial distribution of power is further corrected using a calorimetric process based on core exit thermocouple measurements. Updating of the power distribution based on ex-core detector and core exit thermocouple readings is performed on a nearly continuous basis. A reasonably accurate representation of the core power distribution is determined through use of the MIDS. The flux traces obtained with MIDS are used to calibrate the BEACON predictive model, and also to calibrate both the ex-core detector response, and the core exit thermal couple readings. Between MIDS in-core flux scans, the ex-core and core exit thermocouple calibration factors are utilized in the BEACON power distribution calculation.

A key factor in the BEACON methodology is the uncertainties applied to the BEACON measured power distributions. One component of the measurement uncertainty is the variability of the thermocouple calibration factors (also referred to as mixing factors). The mixing factor uncertainty (standard deviation) is shown to exhibit a power level dependency due to the separate effects of decreased temperature difference (ΔT_{RPL}) at lower power levels and changes in cross-flow patterns at reduced power levels.

In the current methodology, the mixing factor uncertainty is determined as a function of power level by accounting for the effects of decreased ΔT_{RPL} and changes in cross-flow patterns separately. Additionally, an average thermocouple standard deviation at hot full-power conditions is determined from past performance of thermocouples from previous cycles and utilized to account for measurement uncertainty.

The updated methodology described in WCAP-12472-P-A, Addendum 4, utilizes core exit thermocouple data collected during initial power ascension following a refueling outage. This process simultaneously considers the effects of both ΔT_{RPL} and cross-flow patterns on mixing factor uncertainty, and eliminates the reliance on previous cycle thermocouple data, thus resulting in a more accurate and reliable evaluation of thermocouple uncertainties. Additionally, the use of current cycle data allows for the inclusion of any changes to the thermocouples since the previous cycle. These changes include the failure or replacement of individual thermocouples, relocation of a thermocouple or instrumentation system changes.

The NRC Safety Evaluation Report (SER) dated August 9, 2012, approving WCAP-12472-P-A, Addendum 4 (ADAMS Accession No. ML12150A147), states that the updated BEACON thermocouple methodology is acceptable. The NRC staff's review of the applicant's LAR shows that the licensee has considered the specific application of the NRC-approved methodology to

DCPP, Units 1 and 2, including the plant-specific configuration of the core exit thermocouple design and treatment of thermocouple failures.

In the NRC staff's audit report dated February 23, 2016 (ADAMS Accession No. ML16053A323), the staff noted that the licensee stated and confirmed that WCAP-12472-P-A, Addendum 4 may replace WCAP-12472-P-A, Addendum 1-A without loss of applicable licensing basis.

The proposed change in this LAR does not impact DCPP's compliance with the NRC SER conditions associated with the approval of the original BEACON WCAP-12472 and WCAP-12472-P-A, Addendum 1-A. Therefore, the three licensing conditions cited in the NRC SER documenting the staff's approval of the original BEACON topical report WCAP-12472-P-A, Addendum 1-A remains applicable with the implementation of Addendum 4 of WCAP-12472-P-A.

Based on the above, the NRC staff finds that the methodology described in WCAP-12472-NP-A, Addendum 4 may be implemented at DCPP, Units 1 and 2 as proposed by the licensee.

3.3 Technical Evaluation of the Application of WCAP-16045-P-A and addendum

3.3.1 PARAGON and NEXUS Code Descriptions

The PARAGON code, described in WCAP-16045-P-A, is a neutron transport code, intended to replace PHOENIX-P for use in providing nuclear input data for the Westinghouse proprietary Advanced Nodal Code (ANC), which is a core simulator code. NEXUS (WCAP-16045-NP-A, Addendum 1A, Revision 0, "Qualification of the NEXUS Nuclear Data Methodology," August 2007) is an improvement to the PARAGON code system.

WCAP-16045-NP-A, Revision 0, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004 and WCAP-16045-NP-A, Addendum 1-A are NRC-approved, based on extensive validation and verification. As described in the topical report and addendum, the data set used for validation and verification includes numerous criticality experiments and plant data. The plant data reflect a widely varied set of operating conditions that includes numerous fuel designs, lattice geometries, and burnable absorber loadings in use in the Westinghouse- and Combustion Engineering-designed nuclear steam supply systems. The verification was expanded to include comparison of NEXUS results for critical boron concentration to those obtained using the previous PARAGON methodology. The results showed excellent agreement between NEXUS predictions and available data.

3.3.2 Applicability to DCPP, Units 1 and 2

The NRC staff reviewed information contained in the UFSAR to verify that the database discussed in Section 3.3.1 of this SE includes fuel with similar characteristics to that used at DCPP. Chapter 4, "Reactor," of the UFSAR indicates that DCPP, Units 1 and 2, use 17x17 Westinghouse optimized fuel. This design feature is represented in the database described above; therefore, the NRC staff determined that the NEXUS code system is acceptable for use at DCPP. The only condition or limitation established in the SERs approving PARAGON and NEXUS precludes their use for mixed-oxide (MOX) cores. The NRC staff verified and

documented in its audit report dated February 23, 2016, that DCPD does not use MOX fuel in the core reload design.

The NRC staff determined, therefore, that a separate licensing action would be required in order to use MOX fuel at DCPD, so the MOX limitation in the SERs approving PARAGON and NEXUS is satisfied.

3.4 Additional Review Topics

The addendum to WCAP-16045-P-A brings about a change to the way boron letdown curves are calculated and provide input into the overall nuclear design method. Because of this, in previous reviews of PARAGON-NEXUS/ANC applications, the NRC staff has found it necessary to verify that no changes were made to the analysis methods for post loss-of-coolant accident (LOCA) subcriticality and boric acid precipitation behavior.

The PARAGON code, as described in WCAP-16045-P-A, is a neutron transport code intended to replace PHOENIX-P for use in providing cross-section data to the Westinghouse proprietary ANC, a core simulator code. NEXUS, described in WCAP-16045-P-A, Addendum 1-A, is an improvement to the PARAGON/ANC code system that changes the method of communicating the nuclear data output from PARAGON to ANC. Rather than using boron letdown curves, NEXUS accounts for variations in the neutron spectrum by parameterizing the PARAGON cross-section output and reconstructing it within ANC.

WCAP-16045-P-A and addendum are NRC-approved, and the NRC staff has previously determined that they have been extensively validated and verified using a variety of experiments and plant data that account for a wide range of differences in operating conditions. In Addendum 1-A, the verification was expanded to include the comparison of NEXUS results for critical boron concentration to those obtained using the previous PARAGON methodology, showing excellent agreement between NEXUS predictions and available data.

The use of WCAP-16045-NP-A, Addendum 1, for DCPD does not affect the inputs or method(s) for ensuring core subcriticality, both short- and long-term post-LOCA, thereby precluding the potential for return to power following a large break LOCA. Since neither the post-LOCA boron source concentration nor heat generation are impacted by the use of Addendum 1, the current emergency operating procedure timing for boric acid precipitation and the action time for switching to simultaneous injection will continue to remain valid. Core design specific parameters that are verified each cycle to be conservative with respect to the LOCA inputs and refueling boron concentration, will continue to be calculated using NRC approved methods.

The NRC staff concludes this is acceptable, because it confirms that the implementation of PARAGON and NEXUS at DCPD will not affect the calculation of post-LOCA boron requirements or emergency procedures to mitigate post-LOCA boric acid precipitation. Based on the above, the staff concludes that the existing post-LOCA analyses for subcriticality and long-term cooling remain applicable.

3.5 Summary

The NRC staff has determined that the proposed changes in the UFSAR to the BEACON core monitoring system methodology described in WCAP-12472-P-A, Addendum 4 and the TS changes required to replace the PHOENIX-P code system with the PARAGON/NEXUS code system are acceptable for DCP. This determination is based on 1) that the original BEACON TR WCAP-12472-P-A remains applicable, 2) that PARAGON and NEXUS are NRC-approved methods, which have been determined to be applicable to DCP, and 3) that the DCP post-LOCA subcriticality and boric acid precipitation analyses will be unaffected by the proposed changes. Based on the above, the staff concludes that the proposed TS revision is consistent with the guidance provided in GL 88-16, and that the generic qualification and validation of the PARAGON-NEXUS/ANC system satisfies the requirements of 10 CFR 50.34 and 10 CFR 50.36.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component 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 and there has been no public comment on such finding published in the *Federal Register* on June 9, 2015 (80 FR 32628). 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: F. Forsaty, NRR

Date: March 9, 2016

E. Halpin

- 2 -

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

Siva P. Lingam, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosures:

1. Amendment No. 224 to DPR-80
2. Amendment No. 226 to DPR-82
3. Safety Evaluation

cc w/encls: Distribution via Listserv

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LPL4-1 Reading

RidsACRS_MailCTR Resource

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RidsNrrDssStsb Resource

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KWest, NRR/DSS/STSB

FForsaty, NRR/DSS/SRXB

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*SE via memo

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DSS/SRXB/BC (A)*	NRR/DSS/SNPB/BC
NAME	SLingam	JBurkhardt	CJackson (EOesterle for)	JDean
DATE	2/26/16	2/25/16	2/18/16	3/7/16
OFFICE	NRR/DSS/STSB/BC	OGC – NLO w/comments	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM
NAME	RElliott	JLindell	RPascarelli (FLyon for)	SLingam
DATE	3/1/16	3/4/16	3/9/16	3/9/16

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