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

March 4, 2020

Mrs. Maria L. Lacal
Executive Vice President/
Chief Nuclear Officer
Mail Station 7602
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 - ISSUANCE OF AMENDMENT NOS. 212, 212, AND 212 TO REVISE TECHNICAL SPECIFICATIONS TO SUPPORT THE IMPLEMENTATION OF FRAMATOME HIGH THERMAL PERFORMANCE FUEL (EPID L-2018-LLA-0194)

Dear Mrs. Lacal:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment Nos. 212, 212, and 212, to Renewed Facility Operating License Nos. NPF-41, NPF-51, and NPF-74 for the Palo Verde Nuclear Generating Station (Palo Verde), Units 1, 2, and 3, respectively. The amendments requested changes to the Technical Specifications (TSs) in response to your application dated July 6, 2018, as supplemented by letters dated October 18, 2018, March 1, 2019, May 17, 2019, October 4, 2019, November 26, 2019, and December 19, 2019.

The licensee requested revisions to the TSs to support the implementation of Framatome Advanced Combustion Engineering 16x16 High Thermal Performance fuel design with M5® as a fuel rod cladding material and gadolinia as a burnable absorber for Palo Verde, Units 1, 2, and 3. In addition to this license amendment request, the licensee is requesting an exemption from certain requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.46, "Acceptance criteria for emergency core cooling systems [ECCS] for light-water nuclear power reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," to allow the use of Framatome M5® alloy as a fuel cladding material. In addition, the proposed amendments would revise TS 2.1.1, "Reactor Core SLs [Safety Limits]"; TS 4.2.1, "Fuel Assemblies"; and TS 5.6.5, "Core Operating Limits Report (COLR)."

The amendments would adapt the approved Palo Verde reload analysis methodology to address both Westinghouse and Framatome fuel, including the implementation of Framatome methodologies, parameters and correlations. The ability to use either Westinghouse or Framatome fuel will ensure security of the Palo Verde fuel supply by providing for multiple fuel vendors with reliable fuel designs and geographically diverse manufacturing facilities.

Enclosure 4 to this letter contains proprietary information. When separated from Enclosure 4, this document is DECONTROLLED.

The NRC staff has determined that the related safety evaluation contains proprietary information pursuant to 10 CFR 2.390, "Public inspections, exemptions, request for withholding." The proprietary information is indicated by bolded text enclosed within **[[double brackets]]**. The proprietary version of the safety evaluation is provided as Enclosure 4. Accordingly, the NRC staff has also prepared a non-proprietary version of the safety evaluation, which is provided as Enclosure 5.

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

Sincerely,

/RA/

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

Docket Nos. STN 50-528, STN 50-529,
and STN 50-530

Enclosures:
1. Amendment No. 212 to NPF-41
2. Amendment No. 212 to NPF-51
3. Amendment No. 212 to NPF-74
4. Safety Evaluation (Proprietary)
5. Safety Evaluation (Non-Proprietary)

cc w/o Enclosure 4: Listserv

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-528

PALO VERDE NUCLEAR GENERATING STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 212
License No. NPF-41

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

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (collectively, the licensees) dated July 6, 2018, as supplemented by letters dated October 18, 2018, March 1, 2019, May 17, 2019, October 4, 2019, November 26, 2019, and December 19, 2019, 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 paragraphs 2.C(2) and 2.C(14) of Renewed Facility Operating License No. NPF-41 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this renewed operating license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

   (14) The Additional Conditions contained in Appendix D, as revised through Amendment No. 212, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of issuance and shall be implemented within 90 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 Renewed Facility
Operating License No. NPF-41
and Technical Specifications

Date of Issuance: March 4, 2020

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-529

PALO VERDE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 212
License No. NPF-51

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

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (collectively, the licensees) dated July 6, 2018, as supplemented by letters dated October 18, 2018, March 1, 2019, May 17, 2019, October 4, 2019, November 26, 2019, and December 19, 2019, 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 paragraphs 2.C(2) and 2.C(9) of Renewed Facility Operating License No. NPF-51 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this renewed operating license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

    (9) The Additional Conditions contained in Appendix D, as revised through Amendment No. 212, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of issuance and shall be implemented within 90 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 Renewed Facility Operating License No. NPF-51 and Technical Specifications

Date of Issuance: March 4, 2020

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-530

PALO VERDE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 212
License No. NPF-74

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

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (collectively, the licensees) dated July 6, 2018, as supplemented by letters dated October 18, 2018, March 1, 2019, May 17, 2019, October 4, 2019, November 26, 2019, and December 19, 2019, 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 paragraphs 2.C(2) and 2.C(5) of Renewed Facility Operating License No. NPF-74 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this renewed operating license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

   (5) The Additional Conditions contained in Appendix D, as revised through Amendment No. 212, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of issuance and shall be implemented within 90 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 Renewed Facility
Operating License No. NPF-74
and Technical Specifications

Date of Issuance: March 4, 2020

ATTACHMENT TO LICENSE AMENDMENT NOS. 212, 212, AND 212 TO

RENEWED FACILITY OPERATING LICENSE NOS. NPF-41, NPF-51, AND NPF-74

PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3

DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

Replace the following pages of the Renewed Facility Operating License Nos. NPF-41, NPF-51, and NPF-74, 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.

Renewed Facility Operating License No. NPF-41

REMOVE	INSERT
5	5
6	6

Renewed Facility Operating License No. NPF-51

REMOVE	INSERT
6	6
7	7

Renewed Facility Operating License No. NPF-74

REMOVE	INSERT
4	4

Appendix A - Technical Specifications

REMOVE	INSERT
2.0-1	2.0-1
4.0-1	4.0-1
5.6-3	5.6-3
5.6-7	5.6-7
5.6-8	5.6-8

Appendix D – Additional Conditions

REMOVE	INSERT
1	1
--	4

(1) Maximum Power Level

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3990 megawatts thermal (100% power), in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this renewed operating license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Antitrust Conditions

This renewed operating license is subject to the antitrust conditions delineated in Appendix C to this renewed license.

(4) Operating Staff Experience Requirements

Deleted

(5) Post-Fuel-Loading Initial Test Program (Section 14, SER and SSER 2)*

Deleted

(6) Environmental Qualification

Deleted

(7) Fire Protection Program

APS shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility, as supplemented and amended, and as approved in the SER through Supplement 11, subject to the following provision:

APS may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

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*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.

(8) Emergency Preparedness

Deleted

(9) Results of Piping Vibration Test Program (Section 3.9.2, SER)

Deleted

(10) Response to Salem ATWS Event (Section 7.2, SSER 7, and Section 1.11, SSER 8)

Deleted

(11) Supplement No. 1 to NUREG-0737 Requirements

Deleted

(12) Radiochemistry Laboratory (Section 7.3.1.5(3), Emergency Plan)

Deleted

(13) RCP Shaft Vibration Monitoring Program (Section 5.4.1, SSER 12)

Deleted

(14) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 212, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Additional Conditions.

(15) Mitigation Strategy License Condition

APS shall develop and maintain strategies for addressing large fires and explosions and that includes the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance.
2. Assessment of mutual aid fire fighting assets.
3. Designated staging areas for equipment and materials.
4. Command and control.
5. Training of response personnel.

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets.
2. Communications.

(1) Maximum Power Level

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3990 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this renewed operating license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Antitrust Conditions

This renewed operating license is subject to the antitrust conditions delineated in Appendix C to this renewed operating license.

(4) Operating Staff Experience Requirements (Section 13.1.2, SSER 9)*

Deleted

(5) Initial Test Program (Section 14, SER and SSER 2)

Deleted

(6) Fire Protection Program

APS shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility, as supplemented and amended, and as approved in the SER through Supplement 11, subject to the following provision:

APS may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(7) Inservice Inspection Program (Sections 5.2.4 and 6.6, SER and SSER 9)

Deleted

*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.

(8) Supplement No. 1 to NUREG-0737 Requirements

Deleted

(9) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 212, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Additional Conditions.

(10) Mitigation Strategy License Condition

APS shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance.
2. Assessment of mutual aid fire fighting assets.
3. Designated staging areas for equipment and materials.
4. Command and control.
5. Training of response personnel.

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets.
2. Communications.
3. Minimizing fire spread.
4. Procedures for implementing integrated fire response strategy.
5. Identification of readily-available pre-staged equipment.
6. Training on integrated fire response strategy.
7. Spent fuel pool mitigation measures.

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing.
2. Dose to onsite responders.

(4) Pursuant to the Act and 10 CFR Part 30, 40, and 70, APS to receive, possess, and use in amounts 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, APS to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating 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

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3990 megawatts thermal (100% power), in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this renewed operating license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Antitrust Conditions

This renewed operating license is subject to the antitrust conditions delineated in Appendix C to this renewed operating license.

(4) Initial Test Program (Section 14, SER and SSER 2)

Deleted

(5) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 212, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Additional Conditions.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, Departure from Nucleate Boiling Ratio (DNBR) shall be maintained at ≥ 1.34.

2.1.1.2 In MODES 1 and 2,

2.1.1.2.1 The peak fuel centerline temperature for Westinghouse supplied fuel using erbium as a burnable poison shall be maintained < 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-382-P-A).

2.1.1.2.2 The peak fuel centerline temperature for Westinghouse supplied fuel using zirconium-diboride as a burnable poison, or not using a burnable poison integral to the fuel pellet, shall be maintained < 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup).

2.1.1.2.3 The peak fuel centerline temperature for Framatome supplied fuel using gadolinium as a burnable poison, or not using a burnable poison integral to the fuel pellet, shall be maintained < 4901°F (decreasing by 13.7°F per 10,000 MWD/MTU for burnup).

2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained at ≤ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1.1 or SL 2.1.1.2 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

4.0 DESIGN FEATURES

4.1 Site Location

The Palo Verde Nuclear Generating Station is located in Maricopa County, Arizona, approximately 50 miles west of the Phoenix metropolitan area. The site is comprised of approximately 4,050 acres. Site elevations range from 890 feet above mean sea level at the southern boundary to 1,030 feet above mean sea level at the northern boundary. The minimum distance from a containment building to the exclusion area boundary is 871 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies.

a. Each assembly shall consist of a matrix of fuel rods with an NRC approved cladding material with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. Each unit-specific COLR shall contain an identification of the fuel types and cladding material in the reactor, and the associated COLR methodologies.

b. A limited number of lead test assemblies not meeting 4.2.1.a may be placed in nonlimiting core regions. Each unit-specific COLR shall contain an identification of any lead test assemblies in the reactor.

4.2.2 Control Element Assemblies

The reactor core shall contain 76 full strength and 13 part strength control element assemblies (CEAs).

The control section for the full strength CEAs shall be either boron carbide with Alloy 625 cladding, or a combination of silver-indium-cadmium and boron carbide with Alloy 625 cladding.

The control section for the part strength CEAs shall be solid Alloy 625 slugs with Alloy 625 cladding.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.

2. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.2.

3. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.4.

4. Boron Dilution Alarm System for Specification 3.3.12.

5. CEA Alignment for Specification 3.1.5.

6. Regulating CEA Insertion Limits for Specification 3.1.7.

7. Part Strength CEA Insertion Limits for Specification 3.1.8.

8. Linear Heat Rate for Specification 3.2.1.

9. Azimuthal Power Tilt - T_q for Specification 3.2.3.

10. DNBR for Specification 3.2.4.

11. Axial Shape Index for Specification 3.2.5.

12. Boron Concentration (Mode 6) for Specification 3.9.1.

13. Fuel types and cladding material in the reactor for Specification 4.2.1.a and 4.2.1.b, and the associated COLR methodologies for Specification 4.2.1.a.

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:

--------------------------------NOTE--------------------------------

The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

(continued)

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

20. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers." [Methodology for Specifications 3.1.1, Shutdown Margin-Reactor Trip Breakers Open; 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; and 3.1.4, Moderator Temperature Coefficient.]

21. CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16 x 16 PWR Fuel." [Methodology for Specifications 3.1.1, Shutdown Margin-Reactor Trip Breakers Open; 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; and 3.1.4, Moderator Temperature Coefficient.]

22. WCAP-16500-P-A, "CE 16x16 Next Generation Fuel Core Reference Report." [Methodology for Specifications 2.1.1, Reactor Core SLs; 3.2.4, DNBR]

23. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis." [Methodology for Specifications 2.1.1, Reactor Core SLs; 3.2.4, DNBR]

24. CENPD-387-P-A, "ABB Critical Heat Flux Correlations for PWR Fuel." [Methodology for Specifications 2.1.1, Reactor Core SLs; 3.2.4, DNBR]

25. WCAP-16523-P-A, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes." [Methodology for Specifications 2.1.1, Reactor Core SLs; 3.2.4, DNBR]

26. WCAP-16072-P-A, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs." [Methodology for Specifications 2.1.1, Reactor Core SLs; 3.2.4, DNBR]

27. EMF-2103P-A, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors." [Methodology for Specification 3.2.1, Linear Heat Rate]

28. EMF-2328 (P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." [Methodology for Specification 3.2.1, Linear Heat Rate]

29. BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code." [Methodology for Specification 3.2.1, Linear Heat Rate]

30. BAW-10241(P)(A), "BHTP DNB Correlation Applied with LYNXT." [Methodology for Specification 3.2.4, DNBR]

31. EPRI-NP-2511-CCM-A, "VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores." [Methodology for Specification 3.2.4, DNBR]

(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 mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B or F of LCO 3.3.10, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG.

b. Active degradation mechanisms found.

c. Nondestructive examination techniques utilized for each degradation mechanism.

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications.

e. Number of tubes plugged during the inspection outage for each active degradation mechanism.

f. Total number and percentage of tubes plugged to date.

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

(continued)

APPENDIX D

ADDITIONAL CONDITIONS

RENEWED FACILITY OPERATING LICENSE NOS. NPF-41, NPF-51, AND NPF-74

The licensee shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Conditions	Implementation Date
205	APS shall apply a radial power fall off (RFO) curve penalty, equivalent to the fuel centerline temperature reduction in Section 4 of Attachment 8 to the Palo Verde license amendment request dated July 1, 2016, to accommodate the anticipated impacts of thermal conductivity degradation (TCD) on the predictions of FATES3B at high burnup for Westinghouse Next Generation Fuel or to future Westinghouse-supplied fuel designs introduced at PVNGS to which the FATES3B fuel performance code would be applied. To ensure the adequacy of this RFO curve penalty, as part of its normal reload process for each cycle that analysis using FATES3B is credited, APS shall verify that the FATES3B analysis is conservative with respect to an applicable confirmatory analysis using an acceptable fuel performance methodology that explicitly accounts for the effects of TCD. The verification shall confirm satisfaction of the following conditions: i. The maximum fuel rod stored energy in the confirmatory analysis is bounded by the maximum fuel rod stored energy calculated in the FATES3B and STRIKIN-II analyses with the RFO curve penalty applied. ii. All fuel performance design criteria are met under the confirmatory analysis. If either of the above conditions cannot be satisfied initially, APS shall adjust the RFO curve penalty or other core design parameters such that both conditions are met.	The license amendment shall be implemented within 90 days of the date of issuance.

Amendment Number	Additional Conditions	Implementation Date
212	Prior to use of fresh fuel from multiple fuel vendors in a single reload batch, APS will obtain NRC approval of the methodology used to perform the associated reload safety analyses. Lead Test assemblies per Technical Specification (TS) 4.2.1.b are not considered mixed fresh fuel.	The license amendment shall be implemented within 90 days of the date of issuance.

ENCLOSURE 5
(NON-PROPRIETARY)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 212, 212, AND 212 TO RENEWED

FACILITY OPERATING LICENSE NOS. NPF-41, NPF-51, AND NPF-74

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3

DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

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 NOS. 212, 212, AND 212 TO RENEWED

FACILITY OPERATING LICENSE NOS. NPF-41, NPF-51, AND NPF-74

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

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DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

1.0 INTRODUCTION

By letter dated July 6, 2018 (Reference 1), as supplemented by letters dated October 18, 2018, March 1, 2019, May 17, 2019, October 4, 2019, November 26, 2019, and December 19, 2019 (References 2, 3, 4, 5, 6, and 7, respectively), Arizona Public Service Company (APS, the licensee) requested changes to the technical specifications (TSs) to support the implementation of Framatome Advanced Combustion Engineering (CE) 16x16 High Thermal Performance (HTP™) fuel design with M5® as a fuel rod cladding material and gadolinia as a burnable absorber for Palo Verde Nuclear Generating Station (Palo Verde or PVNGS), Units 1, 2, and 3. In addition, the proposed amendments would revise TS 2.1.1, "Reactor Core SLs [Safety Limits]"; TS 4.2.1, "Fuel Assemblies"; and TS 5.6.5, "Core Operating Limits Report (COLR)." The proposed TS changes included the addition of several topical reports (TR) that describe the analytical methods to be used in the determination of core operating limits, including Framatome large-break and small-break loss-of-coolant accident (LOCA) evaluation models. Adoption of the Framatome small-break LOCA (SBLOCA) methodology required consideration of reactor coolant pump (RCP) trip timing, as continued operation of RCPs following a SBLOCA may have a detrimental effect upon core uncovery and peak cladding temperature for certain break sizes and break locations, due to redistribution of coolant inventory within the primary system. Consequently, APS proposed a new time critical action, which requires tripping RCPs within 5 minutes following the loss of subcooled margin, in this license amendment request (LAR).

The proposed amendments would adapt the approved Palo Verde reload analysis methodology to address both Westinghouse and Framatome fuel, including the implementation of Framatome methodologies, parameters, and correlations (Reference 87). The ability to use either Westinghouse or Framatome fuel will ensure security of the Palo Verde fuel supply by providing for multiple fuel vendors with reliable fuel designs and geographically diverse manufacturing facilities.

In addition to this LAR, APS is requesting an exemption from certain requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46, "Acceptance criteria for emergency

core cooling systems [ECCS] for light-water nuclear power reactors," and 10 CFR Part 50, Appendix K, to allow the use of Framatome M5® alloy as a fuel cladding material. The cladding material exemption supports the fuel transition to Framatome CE 16x16 HTP™ and has been documented separately (Reference 8).

The supplemental letters dated March 1, 2019, May 17, 2019, October 4, 2019, November 26, 2019, and December 19, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on January 8, 2019 (84 FR 90).

To clarify the outstanding issues with the LAR and requested exemption, an initial regulatory audit was conducted from January 22–23, 2019, in Rockville, Maryland (References 9 and 10).

A second regulatory audit was conducted from June 17–20, 2019, at the Palo Verde site in Wintersburg, Arizona (References 11 and 12). The purpose of the second regulatory audit was to continue discussion of outstanding issues and permit the NRC staff to perform an audit of calculations and other documentation supporting the LAR/exemption submittal.

A third regulatory audit was conducted from November 7–8, 2019, in Rockville, Maryland (References 13 and 14). The purpose of the third regulatory audit was to resolve outstanding issues, and to audit Westinghouse engineering calculations supporting the introduction of Framatome HTP™ fuel.

By e-mail dated April 5, 2019 (Reference 15) and letter dated August 29, 2019 (Reference 16), the NRC sent the licensee requests for additional information (RAIs). By letters dated May 17, 2019, October 4, 2019, November 26, 2019, and December 19, 2019, the licensee responded to the RAIs.

2.0 REGULATORY EVALUATION

The NRC staff considered the following regulatory requirements and guidance during its review of the LAR.

Regulatory Requirements

The regulations under 10 CFR 50.36, "Technical specifications," provide regulatory requirements related to the content of TSs. Section 50.36(b) of 10 CFR requires that each license authorizing the operation of a facility will include TSs and that the TSs will be derived from the safety analysis. Section 50.36(c) of 10 CFR specifies the categories that are to be included in the TSs including (1) SLs, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

Section 50.36(c)(1)(i)(A) of 10 CFR states, in part: "Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down."

Section 50.36(c)(1)(ii)(A) of 10 CFR states, in part: "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

Section 50.36(c)(4), "Design features," of 10 CFR, states, in part: "Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety...."

Section 50.36(c)(5), "Administrative controls," of 10 CFR, states, in part: "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."

Requirements for analyzing the design-basis LOCA are provided in 10 CFR 50.46 and 10 CFR Part 50, Appendix K, and Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 35, "Emergency core cooling."

Section 50.46(a)(1)(i) of 10 CFR states, in part:

> Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in [Section 50.46(b)]. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.

Appendix K to 10 CFR Part 50, sets forth the documentation requirements for each evaluation model, and establishes required and acceptable features of evaluation models for heat removal by the ECCS.

GDC 35 requires abundant core cooling sufficient to (1) prevent fuel and cladding damage that could interfere with effective core cooling and (2) limit the metal-water reaction on the fuel cladding to negligible amounts. GDC 35 further requires suitable redundancy of the ECCS, such that it can accomplish its design functions, assuming a single failure, irrespective of whether its electrical power is supplied from offsite or onsite sources.

Furthermore, the Palo Verde Updated Final Safety Analysis Report (UFSAR) (Reference 17) describes compliance with the following additional GDC of Appendix A to 10 CFR Part 50 relevant to this review:

- GDC 10, "Reactor design," states: "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

- GDC 12, "Suppression of reactor power oscillations," states: "The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed."

- GDC 13, "Instrumentation and control," states: "Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges."

- GDC 15, "Reactor coolant system design," states: "The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

- GDC 16, "Containment design," states: "Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

- GDC 19, "Control room," states, in part: "A control room shall be provided from which actions can be taken to operate the [plant] safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents."

- GDC 20, "Protection system functions," states: "The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety."

- GDC 24, "Separation of protection and control systems," states: "The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired."

- GDC 38, "Containment heat removal," states: "A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment

pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels."

- GDC 50, "Containment Design Basis," states: "The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters."

<u>Regulatory Guidance</u>

NUREG-0711, "Human Factors Engineering Program Review Model" Revision 3 (Reference 18).

The following sections of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP):

- Chapter 4, Section 4.2, "Fuel System Design," Revision 3 (Reference 19).

- Chapter 4, Section 4.3, "Nuclear Design," Revision 3 (Reference 20).

- Chapter 4, Section 4.4, "Thermal and Hydraulic Design," Revision 2 (Reference 21).

- Chapter 15, Section 15, "Introduction - Transient and Accident Analyses," Revision 3 (Reference 22).

- Chapter 15, Section 15.0.2, "Review of Transient and Accident Analysis Methods (Reference 23).

- Chapter 18, Section 18, "Human Factors Engineering," Revision 3 (Reference 24).

NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1 (Reference 25).

Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods," Revision 0 (Reference 26).

The NRC staff reviewed the licensee's submittal to evaluate the applicability of Framatome methodology for Palo Verde to confirm that the use of the methodologies is within the NRC-approved ranges of applicability and the results of the analyses are in compliance with the applicable requirements of the regulatory requirements listed above.

The proposed license amendments would add five TRs describing analytical methods for determining the core operating limits to TS 5.6.5.b. Addition of these TRS to the referenced core operating analytical reports would permit the licensee to exercise consistency with the approved analytical methods to determine the core operating limits for Framatome HTP™ fuel on a cycle-specific basis. The following five TRS are:

1. EMF-2103(P)(A), Revision 3, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," June 2016 (Reference 27).

   TR EMF-2103(P)(A) describes the Framatome methodology developed for the realistic evaluation of a large-break LOCA (LBLOCA) for pressurized-water reactors (PWRs) with recirculation (U-tube) steam generators. Specifically, Westinghouse 3- and 4-loop designs; CE, all with fuel assembly lengths of 14 feet or less and ECCS injection to the cold legs, are covered.

   The licensee has demonstrated that the conditions and limitations specified in the NRC safety evaluation (SE) report for Topical Report EMF-2103(P)(A) (Reference 28) are met as described in licensing report, ANP-3639, "Palo Verde Units 1, 2, and 3 Realistic Large Break LOCA Summary Report" (Reference 29).

2. EMF-2328(P)(A) Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001 (Reference 67) and EMF-2328(P)(A) Supplement 1(P)(A) Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," December 2016 (Reference 86).

   TR EMF-2328(P)(A) documents the use of S-RELAP5 thermal-hydraulic analysis code to analyze the SBLOCA for Westinghouse and CE PWR plants. Conditions and limitations listed in the NRC SE were negated with the approval of the supplement to this topical report. There are no other conditions and limitations for this topical report.

3. BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code," January 2004 (Reference 31).

   TR BAW-10231P-A, Revision 1 describes the COPERNIC fuel rod design computer code that performs the thermal/mechanical analyses necessary to simulate the fuel rod behavior during irradiation. The COPERNIC code is approved for uranium dioxide (UO_2) licensing applications with M5® clad material to a peak rod average burnup of 62 gigawatt days per metric ton of uranium (GWd/MTU).

   The licensee has demonstrated that the conditions and limitations listed in the NRC SE for TR BAW-10231P-A have been met.

4. BAW-10241(P)(A), Revision 1, "BHTP [Designation for Framatome] DNB [Departure from Nucleate Boiling] Correlation Applied with LYNXT," July 2005 (Reference 32).

   TR BAW-10241, Revision 1 describes the BHTP DNB correlation proposed to be used with the LYNXT computer code. BAW-10241(P)(A) addresses the extension of the range of applicability of the independent variables, pressure, local mass flux, inlet enthalpy, and local quality in the BHTP critical heat flux (CHF) correlation.

The licensee has stated that the conditions and limitations contained in the NRC SE for this TR are met.

5. EPRI-NP-2511-CCM-A, Mod 02, Revision 3, "VIPRE-01: A Thermal Hydraulic Code for Reactor Cores," March 1988 (References 33, 34, and 35).

   TR EPRI-NP-2511-CCM-A describes the Electric Power Research Institute's (EPRI's) VIPRE [Versatile Internals and Component Program for Reactors] computer code for performing detailed thermal-hydraulic analyses of reactor cores.

3.0 TECHNICAL EVALUATION

The reactor core of each unit at Palo Verde contains a total of 241 fuel assemblies. Each unit currently operates with either CE 16x16 Standard (STD) Fuel, CE 16x16 Next Generation Fuel (NGF), or a mixture of the two fuel designs with a total of 241 fuel assemblies. Each of these fuel assemblies consists of 236 fuel rods arranged in 16x16 square arrays, four outer guide tubes and one center instrumentation tube.

The proposed license amendments would support the loading of Framatome Advanced CE 16x16 HTP™ fuel design with M5® fuel rod cladding and gadolinia (Gd_2O_3) as a burnable neutron absorber (burnable poison) at Palo Verde. The proposed license amendments would also support implementation of Framatome fuel by defining methodologies for fuel mechanical design analysis, fuel rod behavior and performance analyses, ECCS performance analysis, nuclear design analysis, core thermal-hydraulic design analysis, selected non-LOCA transient analyses, and core operating limits supervisory system/core protection calculator system (COLSS/CPCS) setpoints analysis. This LAR does not affect existing Westinghouse analysis methods used for analyzing Westinghouse fuel designs at Palo Verde. This section presents the technical evaluation of the methodologies and analyses performed to evaluate the above listed aspects of core and fuel design.

The NRC staff's technical evaluation is organized into the following sections:

Section	Topic
3.1	Fuel Design and Performance
3.2	Nuclear Design
3.3	Thermal-Hydraulic Methodologies
3.4	Non-LOCA Transient Analysis
3.5	LOCA Analysis
3.6	Containment Response
3.7	Proposed Technical Specification Changes
3.8	Human Performance
4.0	License Conditions

3.1 Fuel Design and Performance

3.1.1 Framatome CE 16x16 HTP™ Fuel Design

The CE 16x16 HTP™ fuel design for Palo Verde is similar to the lead fuel assemblies (LFAs) introduced in Palo Verde Unit 1 core in Cycle 15 in 2008. Framatome CE 16x16 HTP™ fuel

assembly is a 16x16 lattice with four large corner guide tubes and one large central instrument tube. The corner guide tube and center instrument tube each occupy four fuel rod lattice positions. The fuel rod array contains 236 rods some of which contain burnable poison. The fuel rods use M5® zirconium alloy cladding (Reference 36). The fuel assembly uses a cage structure with a lower high mechanical performance (HMP™) spacer grid, intermediate HTP™ spacer grids, and a top HTP™ spacer grid with a larger envelope than the intermediate HTP™ spacers. The lower tie plate (LTP) is attached to the cage structure by means of guide tube screws at the four corner guide tube locations. The upper tie plate (UTP) is attached to the fuel assembly by UTP corner locking nuts at the guide tube locations.

3.1.2 Fuel Mechanical Design

Framatome has performed mechanical compatibility evaluations to assure acceptable fit-up with Palo Verde reactor core internals, fuel handling equipment, fuel storage racks, and two co-resident fuel types. The types of fuel in the Palo Verde core are (1) CE 16 STD with ZIRLO fuel rod cladding material, (2) CE 16 NGF with Optimized ZIRLO fuel cladding material, and (3) advanced CE HTP™ fuel design with M5® as cladding material. The Framatome CE 16 HTP™ fuel assembly design was analyzed in accordance with the NRC-approved generic mechanical design criteria in TR EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs" (Reference 37). The M5® cladding material properties were previously approved by the NRC staff in BAW-10227(P)(A), "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel" (Reference 36). The Framatome HTP™ fuel design is licensed to a fuel rod burnup of 62 GWd/MTU for UO_2 rods and 55 GWd/MTU for gadolinia bearing UO_2 rods.

Mechanical compatibility of Framatome HTP™ fuel design with the resident fuel designs is assessed by using Framatome fuel assembly design details, available Palo Verde plant drawings, co-resident fuel design drawings, and LFA operating experience. Table 2-1, "Comparison of Nominal Mechanical Design Features," in Attachment 8 to the LAR dated July 6, 2018 (Reference 1), provides a comparison of the major dimensions of the CE 16 HTP™, CE 16 STD, and CE 16 NGF fuel designs. This comparison shows that with respect to fuel and clad dimensions all three fuel designs are compatible with each other. An exception to the above statement is that the CE 16 NGF design has incorporated intermediate flow mixing (IFM) grids and a reduced fuel rod pin diameter of 0.374 inch (versus 0.382 inch) as compared to the CE 16 STD and CE 16 HTP™ fuel designs. The Framatome HTP™ fuel design does not have IFM grids or reduced diameter fuel rod pins and is effectively the same as the Framatome LFA design.

3.1.2.1 Fuel Assembly

The CE HTP™ fuel assemblies are essentially the same as the LFAs except for minor design changes, such as a chamfered fuel pellet design, a small change in corner guide tube dashpot elevation and modified reaction springs to increase spring deflections that produce higher assembly holddown force to provide additional liftoff margins. The eight LFAs completed their third cycle of irradiation in March 2013 (References 38, 39, and 40). The licensee reports that post irradiation examinations were performed in January 2014 on these assemblies to confirm the incore performance of the new fuel design and to provide the empirical basis for design licensing. Inspection parameters included a detailed four face visual examination to assess overall condition, fuel assembly growth, shoulder gap closure, spacer grid growth and oxide levels, fuel rod oxide and diameter, and grid-to-rod fretting wear depth. The LFAs operated

three cycles with positive performance as demonstrated from the post irradiation examination campaign.

The introduction of CE 16 NGF fuel design affected the mechanical compatibility with the Framatome LFAs with respect to a slight change into spacer grid centerline elevations, the addition of two IFM grids, and a reduced fuel rod diameter resulting in a reduced fuel assembly weight. The assembly compatibility evaluations demonstrated acceptable structural spacer grid overlap throughout the fuel design life in a mixed-core environment. The thermal-hydraulic impact of the CE 16 NGF IFMs on the Framatome fuel design has indicated that the flow induced vibration assessment and the crossflow velocities reported are within the HTP™ fuel product experience.

Upper and Lower Tie Plates

The licensee verified that the HTP™ fuel assembly UTP and LTP to be compatible with core internals and co-resident fuel. The mechanical compatibility of the UTP is explicitly evaluated because it interfaces with the guide pipes in the upper guide structure, interfaces with all the fuel assembly grapples when moving the fuel assembly, and interfaces with the control element rods. The UTP is unchanged from the Framatome LFA program except for the reaction springs, which were modified to increase the installed spring deflections to produce higher fuel assembly hold down force. The alignment pins interface with the fuel assembly LTP at the corners where one pin is shared with four fuel assemblies. The incore instrumentation is inserted through the center of the LTP.

The LTP mates with the lower support structures such as alignment pins and allows for insertion of the incore detectors. The Framatome LTP envelope is slightly smaller than that of the co-resident fuel and is unchanged from the Framatome LFA design.

Guide Tubes

The guide tubes interface with the control rods and incore instrumentation. The corner guide tubes interface with the control rods and the center guide tube only interfaces with the incore instrumentation. The corner guide tubes are unchanged from the Framatome LFA program, except for slightly lowering the dashpot elevation, allowing a more gradual inner diameter transition. The instrument tube is dimensionally unchanged from the Framatome LFA program. The Framatome corner tubes have an expanded upper region that provides a larger annulus between the control rod outer diameter and the corner guide tube inner diameter in the parked elevation as compared to the NGF design. The axial locations of the guide tube dashpot and weep holes are also similar to the co-resident designs. The Framatome design uses MONOBLOC corner guide tubes that have a constant outer diameter. The center guide tube/instrument tube provides guidance for incore instrumentation. The Framatome instrument tube is consistent with the co-resident design.

Mechanical Design Analysis

Fuel assembly mechanical evaluations under steady-state and faulted conditions were performed using the NRC-approved generic design methods as per TR EMF-92-116(P)(A), Revision 0, consistent with the specified acceptable fuel design limits (SAFDLs) identified in Section 4.2 of the SRP (Reference 19). The use of M5® cladding and guide tubes require that the Framatome design methods be modified to incorporate M5® properties and the generic

design criteria be evaluated to assure continued applicability with TR BAW-10240(P)(A), Revision 0, "Incorporation of M5® Properties in Framatome ANP Approved Methods" (Reference 41). The COPERNIC and CROV computer codes (References 31 and 42, respectively) were used to perform the fuel rod thermal mechanical calculations. The fuel mechanical analyses are broadly separated into fuel rod analyses and fuel assembly structural analyses. For mechanical evaluation analyses, the input parameters are fuel assembly and component characteristics established by mechanical/hydraulic testing, plant parameters, fatigue duty cycles and rod power histories generated for the representative cycles.

Fuel Rod Analyses

The fuel rod analyses evaluate internal rod pressure, cladding creep collapse, transient clad strain (TCS), centerline fuel melt (CFM), cladding fatigue, and cladding corrosion. Table 2-2, "Fuel Design Evaluation Results," in Attachment 8 to the LAR dated July 6, 2018, provides the results of generic design criteria (SAFDLs) for the fuel rod and fuel assembly. Table 2-3, "CFM Rod Local LHGR Limits," and Table 2-4, "TCS Rod Local LHGR Limits," in Attachment 8 of the LAR present the CFM and TCS linear heat generation rate (LHGR) limits, respectively, from the fuel rod analyses.

The results from the analyses demonstrate that the fuel rod design criteria are satisfied for the representative fuel design under normal and faulted conditions to a UO_2 rod average burnup of 62 GWd/MTU and a gadolinia rod average burnup of 55 GWd/MTU in accordance with the safety evaluation report (SER) for the NRC-approved COPERNIC TR BAW-10231P-A. The NRC staff has reviewed the relevant calculations and reports and determined that the fuel assembly and fuel rod analyses satisfy the NRC-approved design criteria for normal and faulted conditions.

3.1.3 Fuel Seismic and LOCA External Loads

For the Palo Verde fuel transition, fuel mechanical design analyses under dynamic seismic and LOCA external loads are performed separately by Westinghouse to assess the CE 16 STD and CE 16 NGF assembly performance and by Framatome to assess the CE 16 HTP™ assembly performance. Since fuel assembly and component characteristics (e.g., stiffness) impact predictions of accelerations and impact loads, APS arranged for the two fuel vendors to share proprietary data to improve the accuracy of mixed-core modeling. Both fuel vendors employed their respective, currently approved methods.

Section 2.4, "End-of-Life Grid Crush Strength for CE 16HTP Fuel," in Attachment 8 to the LAR dated July 6, 2018, briefly describes the CE 16HTP™ structural analysis under dynamic seismic and LOCA loads. Framatome used the approved TR ANP-10337P-A, Revision 0, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations" (Reference 43), methodology, which explicitly accounts for irradiation effects. Compliance to conditions and limitations in the NRC staff's approval of ANP-10337P-A are documented in Section 2.3.2.3, "Faulted Condition Topical Report Conditions and Limitations," in Attachment 8 of the LAR. The LAR lacked sufficient information on the Framatome and Westinghouse mixed-core evaluations for the NRC staff to reach a safety finding. As a result, SNPB RAI-29 and SNPB RAI-30 were issued.

With respect to the Framatome methodology and application to the Palo Verde fuel transition, the response to SNPB RAI-29 describes the mixed-core configurations analyzed to encompass

fuel transitions from CE 16 STD to CE 16 HTP™ and CE 16 NGF to CE 16 HTP™. The response to SNPB RAI-30 describes how the fuel design analyses addressed irradiation effects. Note that much of this material was discussed during the first regulatory audit. Based on the material presented in response to these two RAIs, the NRC staff finds the Framatome CE 16 HTP™ fuel seismic and LOCA design analyses acceptable.

During the third regulatory audit, the NRC staff reviewed the draft responses to SNPB RAI-29 and SNPB RAI-30 pertaining to the Westinghouse fuel seismic analyses. The Westinghouse mixed-core configurations encompass the following fuel transitions: CE 16 STD to CE 16 HTP™, CE 16 NGF to CE 16 HTP™, and CE 16 HTP™ to CE 16 NGF. Mixed-core configurations were based on Palo Verde fuel management guidelines and controlled by restrictions related to other than seismic considerations. Hence, virtually all potential mixed fuel configurations were evaluated. During the audit, APS stated that cycle-specific verification would ensure that the upcoming fuel loading pattern is bounded by the analyzed Framatome and Westinghouse analyses. Otherwise, cycle-specific analyses would be performed using the approved methodologies.

All the Westinghouse evaluations were performed at beginning of life with still-water damping, consistent with the approved methodology. While irradiation impacts were not considered, flowing water damping has been shown to be a significant credit and is being used in TR PWR Owners Group (PWROG)-16043-P, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs" (Reference 44), to offset irradiation effects.

For the CE 16 NGF to CE 16 HTP™ and CE 16 HTP™ to CE 16 NGF fuel transitions, positive margin is maintained for all assembly components, including grids (i.e., no grid deformation is predicted). This includes operating basis earthquake and safe shutdown earthquake (SSE)+LOCA external loads.

For the CE 16 STD to CE 16 HTP™ fuel transition, positive margin is maintained for all assembly components, excluding mid-span grids within CE 16 STD assemblies located on the core periphery under SSE+LOCA loads. For existing full cores of CE 16 STD, as well as CE 16 STD to CE 16 NGF transition cores, Westinghouse predicts higher loads and more plastic deformation of these mid-span grids on the core periphery. Potential impacts on component integrity, control rod insertion, and coolability were found to be acceptable with respect to approved criteria. The proposed CE 16 STD to CE 16 HTP™ transition is in the positive direction (i.e., lower loads and deformation) and therefore, is also acceptable.

One of the NRC staff's concerns with the legacy Westinghouse fuel seismic methodology is the use of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Service Level D criteria for guide tubes. This would allow local yielding and potentially buckling of the guide tubes. Given that Palo Verde has rodded core locations along the perimeter, this concerned the NRC staff. During the third regulatory audit, the NRC staff reviewed Westinghouse engineering calculations, which demonstrated that the maximum predicted local guide tube stress under SSE+LOCA loads remained below the allowable stress for operating basis earthquake. Hence, predicted stress remained within the elastic region. Review of a second Westinghouse engineering calculation, which evaluated grid deformation mechanics with respect to guide tube location displacement provided further evidence on control rod insertion. Based upon this information, the NRC staff no longer has concerns with the level of grid deformation with respect to control rod insertion.

A second concern with the legacy Westinghouse fuel seismic methodology is that it lacks an assessment of predicted SSE grid deformation on local thermal-hydraulic conditions and predictions on departure from nucleate boiling ratio (DNBR) during anticipated operational occurrences and postulated accidents. During the third regulatory audit (Reference 14), the NRC and Westinghouse staff discussed the magnitude of plastic deformation at these peripheral locations. Given the minimal potential hydraulic channel closure and low fuel assembly power along the core periphery, the NRC staff no longer has concerns with the DNB performance of these fuel rods during anticipated operational occurrences and postulated accidents.

Based upon the information provided in response to SNPB RAI-29 and RAI-30 by letters dated October 4 and November 26, 2019 (References 5 and 6, respectively), the NRC staff finds the performance of CE 16 STD, CE 16 NGF, and CE 16 HTP™ fuel assemblies under dynamic seismic and LOCA loads acceptable.

3.2 Nuclear Design

Core nuclear design is performed in accordance with Section 4.3 of the Palo Verde UFSAR Section 4.3 (Reference 17). Core nuclear design analysis verifies the cycle-specific reload design and determines that key safety parameters are addressed in the reload design. The core design analysis establishes acceptable core design and generates input to fuel performance, thermal-hydraulic, non-LOCA transient, and core protection setpoint reload analyses. The CASMO/SIMULATE code system was used to perform mixed-core and full-core reloads for Palo Verde with Framatome fuel for the fuel cycles designated below:

- N-1 cycle is the Palo Verde U2C19 core without Framatome Fuel that is used as a starting point for the representative reactor core design physics analyses.
- N cycle is one-third Framatome Fuel.
- N+1 cycle is two-thirds Framatome Fuel.
- N+2 cycle is full core of Framatome Fuel.

For representative core designs, the presence of CE 16 HTP™ fuel in the Palo Verde core was explicitly incorporated into these neutronic models including fuel geometry and associated nuclear cross sections. The core design analysis used gadolinia as the fuel burnable absorber. For the mixed-core with three different fuel designs, the neutronic characteristics of the three associated cladding materials, M5® (HTP™ fuel), ZIRLO (CE 16 STD), and Optimized ZIRLO (CE 16 NGF), show no significant difference.

The representative fuel transition cycle loading patterns were developed based on design requirements, such as, energy, peaking, and assembly placement specified for Palo Verde Unit 2. The loading patterns were depleted at a core power level of 3990 megawatt thermal (MWt). The first representative fuel transition cycle contains fresh CE 16 HTP™ fuel with once- and twice-burnt CE 16 STD fuel. The second representative fuel transition cycle contains fresh and once-burnt CE 16 HTP™ fuel with twice-burnt CE 16 STD fuel. The third representative fuel transition cycle contains only CE 16 HTP™ fuel. Table 3-1, "Representative Fuel Transition

Cycle Core Characteristics," in Attachment 8 to the LAR dated July 6, 2018, lists key core characteristics based on representative fuel transition cycle designs. The actual cycles may contain all three fuel designs. The maximum CE 16 HTP™ fuel rod average burnup will be maintained less than the licensed burnup limits of 62 GWd/MTU for UO_2 and for 55 GWd/MTU for gadolinia rods.

Representative fuel cycle parameters such as F_R, F_{xy}, F_z, F_q, moderator temperature coefficients, scram worth are subject to cycle-to-cycle variations that occur as fuel loading patterns are changed each cycle. Also, changes to core power distributions and peaking factors vary from cycle to cycle based on actual energy requirements.

3.3 <u>Thermal-Hydraulic Methodologies</u>

The following table provides a list of the current suite of methods that have been previously approved for Palo Verde to perform safety analysis for CE 16 STD and CE 16 NGF fuel.

Table 1: Approved Thermal-Hydraulic Methodologies

Method	**Type**	**Reference**
VIPRE-W (Versatile Internals and Component Program for Reactors; Westinghouse)	Subchannel code	45
TORC (Thermal-Hydraulics of Reactor Core)	Subchannel code	46
WSSV (Westinghouse Side Supported Vane CHF Correlation)	Critical heat flux model	47
WSSV-T Westinghouse Side Supported Vane CHF Correlation for use with the TORC Code)	Critical heat flux model	47
ABB-NV (Westinghouse (ABB) Non-Vane CHF Correlation)	Critical heat flux model	48
WLOP (Westinghouse Low-Pressure CHF Correlation)	Critical heat flux model	48
Macbeth	Critical heat flux model	49
SCU (Statistical Combination of Uncertainties)	Statistical method to combine uncertainties	50, 51, 52

In order to perform safety analysis on CE 16 HTP™ fuel, two methods must be added: the VIPRE-01 subchannel code (Reference 33), and the BHTP CHF model (Reference 32). The addition of these two methods and the subsequent analysis, which must be performed for CE 16 HTP™ fuel, is considered in the following subsections.

3.3.1 Use of VIPRE-APS Subchannel Code for Reactor Safety Analysis

VIPRE-01 was developed by Battelle Pacific Northwest Laboratories under the sponsorship of EPRI and submitted to the NRC for generic review in 1984 (Reference 33). The NRC staff's review of VIPRE-01 MOD-1.0 found the code acceptable for performing PWR licensing calculations with the associated limitations and conditions stated in the SE for EPRI NP-2511-CCM (Reference 34). In 1989, VIPRE-01 MOD-2.0 was submitted for generic review. This new version included new models, error correction, and the justification for applicability to boiling water reactors (BWRs). The NRC staff's review of the updated version found that the code was acceptable for performing PWR and BWR licensing calculations with

the associated limitations and conditions stated in the SE for EPRI NP-2511-CCM, Revision 3 (Reference 35).

The original custodians of the code were the VIPRE Maintenance Group who updated the code with new features and error corrections. After MOD-2.1, the code custodians were switched from VIPRE Maintenance Group to CSA Inc., and then to Zachry Nuclear Engineering, Inc. Zachry Nuclear Engineering, Inc., is the current code custodian for VIPRE-01 and has continued to maintain the code with updates and error corrections. APS used VIPRE-01 MOD-2.6, which is abbreviated as VIPRE-APS. This is the version of the code reviewed in this SE.

The NRC staff has reviewed multiple implementations of VIPRE-01 (References 45, 53, and 54). The NRC staff based this portion of the SE on the most recent review of a VIPRE-01 implementation (Reference 55). The NRC staff followed the guidance of SRP Section 15.0.2 (Reference 23) in performing this review. The following subsections outline the criteria of the SRP and how these criteria were satisfied. All criteria can be found in SRP 15.0.2.

SRP Section 15.0.2 directs the reviewer to examine the evaluation model, which is defined as the calculational framework for evaluating the behavior of the reactor coolant system (RCS) during a postulated accident or transient, and includes the computer programs, mathematical models, assumptions, and procedures on how to treat the input and the output, as well as many other factors. While the NRC staff followed the criteria set forth in SRP Section 15.0.2, the implementation of VIPRE-APS is not a new evaluation model but a change to an approved evaluation model. Therefore, the NRC staff focused its review on ensuring that VIPRE-APS would be used in a consistent manner as the currently approved VIPRE-W, and that the modeling options chosen for VIPRE-APS would produce credible results. This resulted in criteria in SRP 15.0.2 not applying specifically to this review, as noted in their respective subsections below.

3.3.1.1 Accident Scenario Identification Process

The accident scenario identification process is a structured process used to identify the key figures of merit or acceptance criteria for the accident or anticipated operational occurrence. It is also used to identify and rank the reactor component and physical phenomena modeling requirements based on their (a) importance to acceptable modeling of the scenario and (b) impact on the figures of merit for the calculation (e.g., CHF).

In this review, the NRC staff did not re-review the currently approved evaluation model and therefore, did not re-evaluate the approved accident scenario identification process. Instead, the NRC staff focused only those portions that could be impacted by the inclusion of VIPRE-APS. In general, the NRC staff did not find any aspects of the accident scenario identification process that would be impacted by VIPRE-APS, as that process is associated more with the type of analysis simulated by the evaluation model than any specific computer code used in the evaluation model.

SRP Section 15.0.2, Subsection III.3.c provides four review criteria for the assessment of the accident scenario identification process that are covered in the following subsections.

3.3.1.1.1 Structured Process

> [T]he process used for accident scenario identification [should be] a structured process.
>
> SRP Section 15.0.2, Subsection III.3c

The use of VIPRE-APS in place of VIPRE-W did not impact the accident scenario identification process. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.1.2 Accident Progression

> The description of each accident scenario should provide a complete and accurate description of the accident progression.
>
> SRP Section 15.0.2, Subsection III.3c

The use of VIPRE-APS in place of VIPRE-W did not impact the description of the accident process. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.1.3 Phenomena Identification and Ranking

> [T]he dominant physical phenomena influencing the outcome of the accident [should be] correctly identified and ranked.
>
> SRP Section 15.0.2, Subsection III.3c

The use of VIPRE-APS in place of VIPRE-W did not impact the dominant physical phenomena influencing the outcome of the accident. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.1.4 Initial and Boundary Conditions

> [T]he description of each accident scenario [should provide] complete and accurate description of the plant initial and boundary conditions.
>
> SRP Section 15.0.2, Subsection III.3c

The use of VIPRE-APS in place of VIPRE-W did not impact the plant initial and boundary conditions. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.2 Documentation

The development of an evaluation model for use in reactor safety licensing calculations requires a substantial amount of documentation including (a) the evaluation model, (b) the accident scenario identification process, (c) the code assessment, (d) the uncertainty analysis, (e) a theory manual, (f) a user manual, and (g) the quality assurance program (QAP).

In this review, the NRC staff did not re-review the currently approved evaluation model and therefore, did not re-evaluate the approved documentation. Instead, the NRC staff focused only those portions that could be impacted by the inclusion of VIPRE-APS. Because of this focus, some of the criteria did not apply to this review.

SRP Section 15.0.2, Subsection III.3.a provides the documentation that are covered in the following subsections.

3.3.1.2.1 Necessary Documentation

> [T]he documentation [should be reviewed] to determine if (i) all documentation listed in Section II.1 above has been provided [the evaluation model, the accident scenario identification process, the code assessment, the uncertainty analysis, a theory manual, a user manual, and the quality assurance program], (ii) the evaluation model overview provides an accurate roadmap of the evaluation model documentation, (iii) all documentation is accurate, complete, and consistent and, (iv) all symbols and nomenclature have been defined and consistently used.
>
> SRP Section 15.0.2, Subsection III.3a

The set of necessary documentation includes the VIPRE theory manual (Reference 56), the VIPRE-W TR (Reference 45), and APS's LAR and RAI responses. Based on the review of this material, the NRC staff finds that it comprises the necessary documentation to fully describe the VIPRE-APS computer code. Therefore, the NRC staff finds that this criterion has been satisfied.

3.3.1.2.2 Theory Manual

> [T]he theory manual [should be] a self-contained document that it describes the field equations, closure relationships, numerical solution techniques, and simplifications and approximations (including limitations) inherent in the chosen field equations and numerical methods.
>
> SRP Section 15.0.2, Subsection III.3a

VIPRE-APS is a modification of EPRI's VIPRE-01 and therefore, shares much of the same theory manual. The field equations, closure relationships, numerical solution techniques, and simplifications and approximations have not changed since the original NRC staff review, therefore, the NRC staff did not find it necessary to repeat the review of the theory manual. In

response to V-APS RAI-02 by letter dated October 4, 2019, APS provided the changes between the version of VIPRE-01 as initially approved by the NRC staff (MOD-2.0) and the version of VIPRE used to generate VIPRE-APS (MOD-2.6). After reviewing these changes, the NRC found that the changes were consistent with bug fixes, which did not alter the underlying calculational methodology as presented in the theory manual. Based on a review of the theory manual for VIPRE-01, APS's initial submittal, and the RAI responses provided, the NRC staff finds that this information together provides a complete theory manual for VIPRE-APS. Therefore, the NRC staff has concluded that this criterion has been satisfied.

3.3.1.2.3 Closure Relationships

> [T]he theory manual [should identify] the pedigree or origin of closure relationships used in the code and the limits of applicability for all models in the code.
>
> SRP Section 15.0.2, Subsection III.3a

The closure relationships are identified in the base VIPRE-01 documentation (Reference 56). In the licensee's LAR dated July 6, 2018, APS identified the selection of specific closure relationships that are necessary for performing reactor safety analysis with VIPRE-APS. Because APS identified which closure relationships were being used and those relationships are described in the VIPRE-01 documentation, the NRC staff has concluded that this criterion has been satisfied.

3.3.1.2.4 User Manual

> [T]he user manual [should provide guidance] for selecting or calculating all input parameters and code options.
>
> SRP Section 15.0.2, Subsection III.3a

APS is already approved for performing safety analysis with VIPRE-W. Further, in response to V-APS RAI-03 by letter dated October 4, 2019, APS stated that VIPRE-APS will be used in a manner consistent with the TR describing VIPRE-W (Reference 45) with certain exceptions and provided a detailed list of those exceptions. Those exceptions are addressed in Section 3.3.1.3.1 of this SER. Because APS has been previously approved for performing this analysis with VIPRE-W and has committed to applying the same process with VIPRE-APS with the stated exceptions, the NRC finds that the use of VIPRE-APS in place of VIPRE-W would not impact the process of determining the input parameters and code options, given the similarity between the subchannel codes. However, the NRC staff did not re-review the process for selecting or calculating all input parameters and code options, because that was outside the scope of this review. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

Note, the one exception to the NRC staff's review of the mixed-core methodology, which is discussed in Section 3.3.1.3.1 of this SER, as previous reviews of the mixed-core methodology were not clearly documented.

3.3.1.2.5 Options for Licensing Calculations

> [T]he guidance in the [user] manual [should specify] the required and acceptable code options for the specific licensing calculations.
>
> SRP Section 15.0.2, Subsection III.3a

Except for the mixed-core methodology described in Section 3.3.1.3.1 of this SE, the use of VIPRE-APS in place of VIPRE-W did not impact the options for licensing calculations. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.2.6 Required Input

> [The] required input settings are hardwired into the input processor so that the code stops with an error message if the required input is not provided or if the input is not within an acceptable range of values or that administrative controls (an independent reviewer QA check) are in place that accomplish the same purpose.
>
> SRP Section 15.0.2, Subsection III.3a

The use of VIPRE-APS in place of VIPRE-W did not affect the required input. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.2.7 Accident-Specific Guidelines

> [C]omputer codes that are used for multiple accidents and transients [should] include guidelines that are specific to each transient or accident.
>
> SRP Section 15.0.2, Subsection III.3a

The use of VIPRE-APS in place of VIPRE-W did not impact the accident-specific guidelines. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.3 Evaluation Model Development

Models must be present for all phenomena and components that have been determined to be important or necessary to simulate the accident under consideration. The chosen mathematical models and the numerical solution of those models must be able to predict the important physical phenomena reasonably well from both qualitative and quantitative points of view. The degree of imprecision that is allowed in the models will ultimately be determined by the amount of uncertainty that can be tolerated in the calculation. Models that cause non-physical predictions to the extent that misinterpretation of the calculated results or trends in the results may occur, are not acceptable.

In this review, the NRC staff did not re-review the currently approved evaluation model and therefore, did not re-evaluate many of the aspects of that approved evaluation model. Instead, the NRC staff focused only those portions that could be impacted by the inclusion of VIPRE-APS.

SRP Section 15.0.2, Subsection III.3.b provides the eight review criteria for evaluation models that are covered in the following subsections.

3.3.1.3.1 Previously Reviewed and Accepted Codes and Models

> [It should be determined] whether the mathematical modeling and computer codes used to analyze the transient or accident should have been previously reviewed and accepted.
>
> SRP Section 15.0.2, Subsection III.3b

VIPRE-01 has been previously reviewed and approved by the NRC. Therefore, VIPRE-APS must meet the conditions and limitations placed in VIPRE-01 MOD-1.0 and VIPRE-01 MOD-2.0. Those conditions and limitations are addressed below.

3.3.1.3.1.1 Conditions and limitations of VIPRE MOD-1.0

The conditions and limitations of VIPRE MOD-1.0 are as follows:

Condition 1:

> The application of VIPRE-01 MOD01 is limited to PWR licensing calculations with heat transfer regime up to CHF.

In response to V-APS RAI-04 by letter dated October 4, 2019, APS confirmed that VIPRE-APS will be limited to heat transfer regimes up to CHF. Therefore, the NRC staff has found this condition has been satisfied.

Condition 2:

> Use of a steady-state CHF correlation with VIPRE-01 MOD01 is acceptable for reactor transient analysis provided that the CHF correlation and its DNBR limit have been reviewed and approved by the NRC and that the application is within the range of applicability of the correlation including fuel assembly geometry, spacer grid design, pressure, coolant mass velocity, quality, etc. Use of any CHF correlation which has not been approved will require the submittal of a separate topical report for staff review and approval. The use of a CHF correlation which has been previously approved for application in connection with another thermal hydraulic code other than VIPRE-01 will require an analysis showing that, given the correlation data base, VIPRE-01 gives the same or a conservative safety limit, or a higher DNBR limit must be used, based on the analysis results.

This condition is being addressed in Section 3.3.2 of this SE.

Condition 3:

> Each organization using VIPRE-01 for licensing calculations should submit separate documentation describing how they intend to use VIPRE-01 and providing justification for their specific modeling assumptions, choice of particular two-phase flow models and correlations, heat transfer correlations, CHF correlation and DNBR limit, input values of plant specific data such as turbulent mixing coefficient, slip ratio, grid loss coefficient, etc., including defaults.

This condition is being addressed in Sections 3.3.1.2.3 and 3.3.2 of this SE.

Condition 4:

> If a profile fit subcooled boiling model (such as Levy and EPRI models) which was developed based on steady-state data is used in boiling transients, care should be taken in the time step size used for transient analysis to avoid the Courant number less than 1.

In the licensee's supplemental letter dated October 18, 2018 (Reference 2), APS confirmed that the time step will be selected to ensure the Courant number will be maintained greater than 1 when a subcooled void (i.e., boiling) model is used. Therefore, the NRC staff finds that this condition is satisfied.

Condition 5:

> The VIPRE-01 user should abide by the quality assurance procedures described in Section 2.6 of this report [the SE for VIPRE-01 MOD-01 (Reference 34)].

In response to V-APS RAI-04 (Reference 5), APS confirmed that VIPRE-APS will abide by the quality assurance procedures described in Section 2.6 of the SE for VIPRE-01 MOD-1. Therefore, the NRC staff has found this condition has been satisfied.

The NRC staff determined that APS has demonstrated that its use of VIPRE-APS has met the conditions and limitations imposed by the NRC on the use of VIPRE-01 MOD 1.0.

3.3.1.3.1.2 Conditions and limitations of VIPRE MOD-2.0

The first two conditions and limitations of VIPRE-01 MOD-2.0 (Reference 56) apply only to BWR licensing applications. The remaining conditions and limitations are as follows:

Condition 3:

> Section 2.2 of Volume 5 of the submittal identifies a spectrum of limitations of the code. Each user, [in its documentation for the NRC approval, should certify] that the code is not being used in violation of these limitations.

In response to V-APS RAI-04 by letter dated October 4, 2019, APS stated that VIPRE-APS will not be used in the following situations:

- Specific two-phase flow conditions that are characterized by large relative velocity between the phases or radical changes in flow regime, such as low-flow boil-off, annular flow, stratified two phase flow, or countercurrent flow.

- Phenomena dominated by local pressure such as flow-down transient, boiling incepted at low pressure, or Boiling Water Reactor (BWR) transient flow instability.

- Free-field situations not dominated by wall friction.

- Situations out of the applicable range of the constitutive correlations.

The NRC staff determined that APS's confirmation that they will not use VIPRE-APS in violation of the limitations given in Section 2.2 of Volume 5 of the VIPRE manual met this condition. The NRC staff finds that this condition has been satisfied.

Condition 4:

> By acceptance of this code version, we do not necessarily endorse procedures and uses of this code described in Volume 5 as appropriate for licensing applications. As the code developer stated in Reference 25, the materials were provided by the code developers as their non-binding advice on efficient use of the code.
>
> Each user is advised to note that values of input recommended by the code developers are for best-estimate use only and do not necessarily incorporate the conservatism appropriate for licensing type analysis. Therefore, the user is expected to justify or qualify input selections for licensing applications.

The documentation submitted by APS provides that justification for the use of VIPRE-APS in applications. That justification is discussed throughout this SE. The NRC staff determined that APS's submitted documentation met this condition. The NRC staff finds that this condition has been satisfied.

The NRC staff determined that APS has demonstrated that its use of VIPRE-APS has met the conditions and limitations imposed by the NRC on the use of VIPRE-01 MOD 2.0.

3.3.1.3.1.2.1 TR WCAP-14565-P-A

APS has confirmed that VIPRE-APS will follow the methodology described in TR WCAP-14565-P-A (Reference 45) that was approved for VIPRE-W. However, APS is taking certain exceptions to this approved methodology. Those exceptions are addressed below:

1. APS is using a radial noding scheme consistent with a 16x16 lattice, and not the 14x14 lattice as shown in the topical report.

The NRC staff finds that this exception is not a change in practice, but a change in documentation, as APS has 16x16 fuel and has used VIPRE-W to perform analysis on 16x16 fuel.

2. APS is not modeling conduction through the fuel rod, as VIPRE-APS will not be used to perform post-CHF film boiling analysis.

   The NRC staff finds that this exception is acceptable because the modeling of conduction through the fuel rod is only needed for post-CHF analysis.

3. [[ ]]

   APS has been previously approved to use [[ ]]. The NRC staff finds that this same approval would be applicable to VIPRE-APS.

4. APS calculated the impacts of mixed cores directly with VIPRE-APS.

   The NRC has evaluated the mixed-core methodology used with VIPRE-APS in Section 3.3.5 of this SE.

5. [[ ]]

   [[ ]] The NRC staff finds that this approach is consistent with what was previously approved for VIPRE-W and concludes that the same approach would be acceptable for VIPRE-APS.

6. APS is using specific values of ABETA and Turbulent Momentum Factor for analysis of the Framatome CE 16HTP™ fuel.

   The NRC finds that the values of ABETA and Turbulent Momentum Factor are reasonable and concludes that they are acceptable for use in the analysis of CE 16HTP™ fuel.

7. [[ ]]

   APS has been previously approved to use [[ ]]. Because these

methods are very similar, the NRC staff finds that the previous approval would also apply to VIPRE-APS.

8. APS has provided a list of closure models, which will be used for VIPRE-APS.

   The NRC has evaluated the mixed-core methodology used with VIPRE-APS in Section 3.3.1.3.4 of this SE.

Because VIPRE-APS is an application of VIPRE-W, the same conditions and limitations of VIPRE-W (Reference 45) would apply. Those conditions and limitations are addressed as follows for this implementation:

1. Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal.

APS has provided the selection of CHF correlation and the correlation limit for the use of the BHTP correlation in VIPRE-APS. APS has further committed to using its currently approved process for other fuel-dependent parameters. Therefore, the NRC staff finds that this condition has been satisfied.

2. Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE.

APS is applying VIPRE-APS as a replacement for VIPRE-W in performing in-house safety analysis. APS is already approved for performing this analysis with VIPRE-W. After a review of the documentation submitted by APS, the NRC staff did not identify any changes that would impact the approval status of the inputs used. Therefore, the NRC staff finds that this condition is satisfied.

3. The NRC staff's generic SER for VIPRE [...] set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using the WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification.

APS is not using the WRB-1, WRB-2, or WRB-2M in VIPRE-APS, therefore that portion of the condition does not apply. APS has provided the additional justification for the use of the BHTP correlation into VIPRE-APS in Section 3.3.2 of this SER. Therefore, the NRC staff finds that this condition is satisfied.

4. Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core

> maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC staff's generic review of VIPRE ... did not extend to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained.

APS has stated that they will not apply VIPRE-APS in post CHF calculations. Therefore, the NRC staff finds that this condition is satisfied.

The NRC staff has determined that APS is applying VIPRE-APS in a manner consistent with VIPRE-W and has justified any deviations. The NRC staff finds that this criterion is satisfied.

3.3.1.3.2 Physical Modeling

> [T]he physical modeling described in the theory manual and contained in the mathematical models [should be] adequate to calculate the physical phenomena influencing the accident scenario for which the code is used.
>
> SRP Section 15.0.2, Subsection III.3b

The field equations used in VIPRE-APS have not been modified from those used in VIPRE-01. After its review of VIPRE-01 (Reference 33), the NRC staff determined that the computer code modeled the physical phenomena, which occurs during PWR transient and accidents and that this modeling is acceptable for use in licensing applications. After a review of the documentation submitted by APS, the NRC staff did not find anything that would invalidate the previous determination made by the staff that the physical modeling would be adequate for performing reactor safety analysis. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.3.3 Field Equations

> [T]he field equations of the evaluation model [should be] adequate to describe the set of physical phenomena that occur in the [scenario].
>
> SRP Section 15.0.2, Subsection III.3b

The field equations used in VIPRE-APS have not been modified from those used in VIPRE-01. After its review of VIPRE-01 (Reference 33), the NRC staff determined that the computer code modeled the physical phenomena that occurs during PWR transient and accidents and that this modeling is acceptable for use in licensing applications. The NRC staff also stipulated that each organization desiring to use VIPRE-01 submit justification for the use of the closure relationships, as these were not considered in the initial review. The closure relationships are treated in the next subsection. After a review of the documentation submitted by APS, the NRC staff did not find anything that would invalidate the previous determination made by the staff that

the field equations would be adequate to describe the physical phenomena. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.3.4 Validation of the Closure Relationships

> [T]he range of validity of the closure relationships [should be] specified and [should be] adequate to cover the range of conditions encountered in the accident scenario.
>
> SRP Section 15.0.2, Subsection III.3b

In Table 5-5, "VIPRE-01/BHTP Modeling Options," in Attachment 8 to the LAR dated July 6, 2018, APS provided the listing of the closure relationships for VIPRE-APS. In response to V-APS RAI-05 by letter dated October 4, 2019, APS provided justification for the closure relationships chosen. In general, APS is using closure relationships that were previously used in other approved versions of VIPRE or have been recommended for use with subchannel computer codes. APS is using well known closure relationships, which have been previously approved or commonly used and are making reasonable selections for key parameters in other closure relationships consistent with general subchannel modeling practices. However, the validation of the closure relationships is based on VIPRE-APS's prediction of CHF test data. The details of that assessment are provided in Section 3.3.2 of this SE. Because VIPRE-APS with the BHTP CHF model adequately predicted CHF test data confirming the validity of the chosen closure relationships, the NRC staff has concluded that this criterion has been satisfied.

3.3.1.3.5 Simplifying and Averaging Assumptions

> [T]he simplifying assumptions and assumptions used in the averaging procedure [should be] valid for the accident scenario under consideration.
>
> SRP Section 15.0.2, Subsection III.3b

The simplifying and averaging assumptions used in VIPRE-APS have not been modified from those used in VIPRE-01 (Reference 33). The NRC staff previously determined that these equations and derivations were correct in its approval of VIPRE-01 (Reference 45). After a review of the documentation submitted by APS, the NRC staff did not find anything that would invalidate the previous determination. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.3.6 Level of Detail in the Model

> [T]he level of detail in the model should be equivalent to or greater than the level of detail required to specify the answer to the problem of interest.
>
> SRP Section 15.0.2, Subsection III.3b

The level of detail in VIPRE-APS has not been modified from that level in VIPRE-01 (Reference 33). The NRC staff previously determined that level of detail was appropriate in its approval of VIPRE-01 (Reference 45). After a review of the documentation submitted by APS, the NRC staff did not find anything that would invalidate this previous determination. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.3.7 Equations and Derivations

> [T]he equations and derivations [should be] correct.
>
> SRP Section 15.0.2, Subsection III.3b

The equations and derivations used in VIPRE-APS have not been modified from those used in VIPRE-01 (Reference 33). The NRC staff previously determined that these equations and derivations were correct in its approval of VIPRE-01 (Reference 45). After a review of the documentation submitted by APS, the NRC staff did not find anything that would invalidate this previous determination. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.3.8 Similarity and Scaling

> [T]he similarity criteria and scaling rationales [should be] based on the important phenomena and processes identified by the accident scenario identification process and appropriate scaling analyses. [S]caling analyses [should be] conducted to ensure that the data and the models will be applicable to the full-scale analysis of the plant transient.
>
> SRP Section 15.0.2, Subsection III.3b

The implementation of VIPRE-APS does not introduce new physical phenomena, which differs from the current phenomena already considered in the analysis with VIPRE-W. After a review of the documentation submitted by APS, the NRC staff did not find anything that would necessitate any additional scaling analysis to justify the use of VIPRE-APS. The NRC staff concluded that this criterion has been satisfied.

3.3.1.4 Code Assessment

The code assessment provides a complete assessment of all code models against applicable experimental data and/or exact solutions to demonstrate that the code is adequate for analyzing the chosen scenario.

In this review, the NRC staff did not re-review the currently approved evaluation model and therefore, did not re-evaluate some of those aspects of code assessment related to the current evaluation model. Instead, the NRC staff focused only those portions, which were related directly to the assessment of VIPRE-APS and if changing to VIPRE-APS would impact the given review criterion.

SRP Section 15.02, Subsection III.3.d provides topics that are covered in the following subsections:

3.3.1.4.1 Single Version of the Evaluation Model

> [A]ll assessment cases [should be] performed with a single version of the evaluation model.
>
> SRP Section 15.0.2, Subsection III.3d

In response to V-APS RAI-06 by letter dated October 4, 2019, APS confirmed that all assessments of VIPRE-APS were performed with the same evaluation model (i.e., same computer code and same modeling options). Because the assessments were performed with the same evaluation model, the NRC staff has concluded that this criterion has been satisfied.

3.3.1.4.2 Validation of the Evaluation Model

> Integral test assessments must properly validate the predictions of the evaluation model for the full size plant accident scenarios. This validation should cover all of the important code models and the full range of conditions encountered in the accident scenarios.
>
> SRP Section 15.0.2, Subsection III.3d

The physical models of VIPRE-APS have not been modified from those of VIPRE-01 (Reference 56). The NRC staff previously determined that the models were acceptable for modeling the physical phenomena of PWR transient and accident analysis in its approval of VIPRE-01. After a review of the documentation submitted by APS, the NRC staff did not find anything that would invalidate this previous determination. The NRC staff determined that because the important physical phenomena have not changed, further integral testing was not necessary.

As discussed below, further CHF testing was necessary to confirm VIPRE-APS ability to perform accident analysis regarding CE 16HTP™ fuel, but this analysis was required for validation that the code could predict the fuel performance and not for validation on any one specific accident scenario. Because no new integral test data was needed, the NRC has determined that this criterion is addressed in the subsection on the application of BHTP in VIPRE-APS given in Section 3.3.2 of this SE.

3.3.1.4.3 Range of Assessment

> [A]ll code closure relationships based in part on experimental data or more detailed calculations [should be] assessed over the full range of conditions encountered in the accident scenario by means of comparison to separate effects test data.
>
> SRP Section 15.0.2, Subsection III.3d

The closure equations chosen by APS are generally accepted for use in PWR transient and accident licensing analysis. Based on the expected known ranges of the transients and accidents, the mechanisms, which prevent the relationships from being used outside their appropriate ranges (Section 3.3.2.2.6 of this SE), as well as the review of the closure relationships (Section 3.3.1.3.4 of this SE), the NRC staff has determined that there has been adequate assessment of the closure relationships. The NRC staff has concluded that this criterion has been satisfied.

3.3.1.4.4 Numerical Solution

> [T]he numerical solution [should] conserve all important quantities.
>
> SRP Section 15.0.2, Subsection III.3d

The numerical solution scheme used in VIPRE-APS has not been modified from that used in VIPRE-01 (Reference 56). After a review of the documentation submitted by APS, the NRC staff did not identify any changes that would impact the numerical solution scheme in the application of VIPRE-APS as a replacement for VIPRE-W. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.4.5 Code Tuning

> [A]ll code options that are to be used in the accident simulation [should be] appropriate and should not be used merely for code tuning.
>
> SRP Section 15.0.2, Subsection III.3d

APS is applying VIPRE-APS as a replacement for VIPRE-W in performing in-house safety analysis. APS is already approved for performing this analysis with VIPRE-W. After a review of the documentation submitted by APS, the NRC staff did not identify any changes that would result in "code tuning" in the application of VIPRE-APS as a replacement for VIPRE-W. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.4.6 Compensating Errors

> The reviewers should ensure that the documentation contains comparisons of all important experimental measurements with the code predictions in order to expose possible cases of compensating errors.
>
> SRP Section 15.0.2, Subsection III.3d

APS is applying VIPRE-APS as a replacement for VIPRE-W in performing in-house safety analysis. APS is already approved for performing this analysis with VIPRE-W. After a review of the documentation submitted by APS, the NRC staff did not identify any changes that would

result in additional opportunity for compensating errors in the application of VIPRE-APS as a replacement for VIPRE-W. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.4.7 Sensitivity Studies

> [A]ssessments [should be] performed where applicable [specific test cases for LOCA to meet the requirements of Appendix K to 10 CFR Part 50 and TMI [Three Mile Island] action items for PWR small-break LOCA].
>
> SRP Section 15.0.2, Subsection III.3d
>
> Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.
>
> Appendix K to 10 CFR Part 50
>
> A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.
>
> *TMI [Three Mile Island] action items for PWR*

In response to V-APS RAI-07 by letter dated October 4, 2019, APS provided the results of an axial noding sensitivity study, which demonstrated that its response was relatively insensitive to node sizes around the chosen axial node size of approximately 2 inches or smaller. EPRI performed radial noding sensitivity studies as described in Volume 4 of the VIPRE-01 topical report (Reference 33), and recommended that at least one full row of subchannels surround the hot channel. In response to V-APS RAI-03 and V-APS RAI-07 by letter dated October 4, 2019, APS confirmed that it will use the same 16x16 lattice geometry and radial noding scheme as it already uses for CE 16 NGF (Reference 68), and that it may use even more detailed models to perform certain analyses such as for transition cores. The APS models are more detailed than those used in the EPRI sensitivity studies, and more detailed than a 2-loop PWR 14x14 lattice model in the VIPRE-W topical report WCAP-14565-P-A (Reference 45) that the NRC previously reviewed. Because APS has provided an axial sensitivity study, which confirms that the results of VIPRE-APS are converged, the NRC staff has concluded that this criterion has been satisfied.

3.3.1.4.8 Assessment Data

> [P]ublished literature [should be referred to] for sources of assessment data for specific phenomena, accident scenarios, and plant types.
>
> SRP Section 15.0.2, Subsection III.3d

The physical and code model interactions of VIPRE-APS have not been modified from those of VIPRE-01 (Reference 56). After a review of the documentation submitted by APS, the NRC staff determined that the only assessment data, which was needed was the CHF model. This assessment was performed in Section 3.3.2 of this SE. Therefore, the NRC staff has concluded that this criterion has been addressed elsewhere.

3.3.1.5 Uncertainty Analysis

Uncertainty analyses are performed to confirm that the combined code and application uncertainty is less than the design margin for the safety parameter of interest when the code is used in a licensing calculation. Examples of safety parameters are peak cladding temperature, cladding oxidation thickness, DNBR, and CPR.

In this review, the NRC staff did not re-review the currently approved evaluation model and therefore, did not re-evaluate the approved method to perform uncertainty analysis. Instead, the NRC staff focused only on those portions that could be impacted by the inclusion of VIPRE-APS. In general, the NRC staff did not find any aspects of the uncertainty quantification process that would be impacted using VIPRE-APS because that process is associated more with the evaluation model than any specific computer code used in the evaluation model.

SRP Section 15.0.2, Subsection III.3.e contains three review criteria for evaluation models, which are addressed in the following subsections:

3.3.1.5.1 Important Sources of Uncertainty

> [T]he accident scenario identification process [should be] used in identifying the important sources of uncertainty.
>
> [S]ources of calculation uncertainties [should be] addressed, including uncertainties in plant model input parameters for plant operating conditions (e.g., accident initial conditions, set points, and boundary conditions).
>
> To address these uncertainties, demonstrate that "the combined code and application uncertainty should be less than the design margin for the safety parameter of interest in the calculation."
>
> SRP Section 15.0.2, Subsection III.3e

APS is applying VIPRE-APS as a replacement for VIPRE-W in performing in-house safety analysis. APS is already approved for performing this analysis with VIPRE-W. After a review of the documentation submitted by APS, the NRC staff did not identify any changes to the important sources of uncertainty in the application of VIPRE-APS as a replacement for VIPRE-W. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.5.2 Experimental Uncertainty

> [T]he uncertainties in the experimental data base [should be] addressed.
>
> [D]ata sets and correlations with experimental uncertainties that are too large when compared to the requirements for evaluation model assessment should not be used.
>
> SRP Section 15.0.2, Subsection III.3e

APS is applying VIPRE-APS as a replacement for VIPRE-W in performing in-house safety analysis. APS is already approved for performing this analysis with VIPRE-W. After a review of the documentation submitted by APS, the NRC staff did not identify any changes that would create any additional experimental uncertainty in the application of VIPRE-APS as a replacement for VIPRE-W. The only experimental uncertainty would be from the CHF testing, which is accounted for in the CHF model's uncertainty. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.5.3 Calculated and Predicted Results

> For separate effects tests and integral effects tests, ... the differences between calculated results and experimental data for important phenomena [should be] quantified for bias and deviation.
>
> SRP Section 15.0.2, Subsection III.3e

After a review of the documentation submitted by APS, the NRC staff determined that the only test data which was needed was the CHF model. This assessment was performed in Section 3.3.2 of this SE. Therefore, the NRC staff has concluded that this criterion has been addressed elsewhere.

3.3.1.6 Quality Assurance Program

The quality assurance program covers the procedures for design control, document control, software configuration control and testing, and error identification and corrective actions used in the development and maintenance of the evaluation model. The program also ensures adequate training of personnel involved with code development and maintenance, as well as those who perform the analyses.

In this review, the NRC staff did not re-review the currently approved evaluation model and therefore, did not re-evaluate the approved QAP. Instead, the NRC staff focused only those portions which could be impacted by the inclusion of VIPRE-APS. The NRC staff did not find any aspects of the QAP that would be impacted using VIPRE-APS, as that program is associated more with the evaluation model than any specific computer code used in the evaluation model.

SRP Section 15.0.2, Subsection III.3.f contains three review criteria for evaluation models that are discussed in the following subsections:

3.3.1.6.1 Appendix B QAP

> [T]he evaluation model [should be] maintained under a quality assurance program that meets the requirements of Appendix B to 10 CFR Part 50.
>
> SRP Section 15.0.2, Subsection III.3f

APS is applying VIPRE-APS as a replacement for VIPRE-W in performing in-house safety analysis. APS is already approved for performing this analysis with VIPRE-W. After a review of the documentation submitted by APS, the NRC staff did not identify any changes that would impact the QAP due to the application of VIPRE-APS as a replacement for VIPRE-W. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.6.2 Quality Assurance Documentation

> [T]he quality assurance program documentation [should include] procedures that address all these areas [design control, document control, software configuration control and testing, and corrective actions].
>
> SRP Section 15.0.2, Subsection III.3f

APS is applying VIPRE-APS as a replacement for VIPRE-W in performing in-house safety analysis. APS is already approved for performing this analysis with VIPRE-W. After a review of the documentation submitted by APS, the NRC staff did not identify any changes that would impact the quality assurance documentation due to the application of VIPRE-APS as a replacement for VIPRE-W except for those changes that described the difference between the application of those two methods. As stated above, that difference has been thoroughly described in the LAR dated July 6, 2018, and the RAI response dated October 4, 2019. Given this documentation, the NRC staff has concluded that this criterion has been satisfied.

3.3.1.6.3 Independent Peer Review

> [I]ndependent peer reviews [should be] performed at key steps in the evaluation model development process.
>
> SRP Section 15.0.2, Subsection III.3f

APS is applying VIPRE-APS as a replacement for VIPRE-W in performing in-house safety analysis. APS is already approved for performing this analysis with VIPRE-W. After a review of the documentation submitted by APS, the NRC staff did not identify any changes that would impact the impendent peer review due to the application of VIPRE-APS as a replacement for

VIPRE-W. Therefore, the NRC staff has concluded that this criterion does not apply in this review.

3.3.1.7 Conclusion on the use of VIPRE-APS

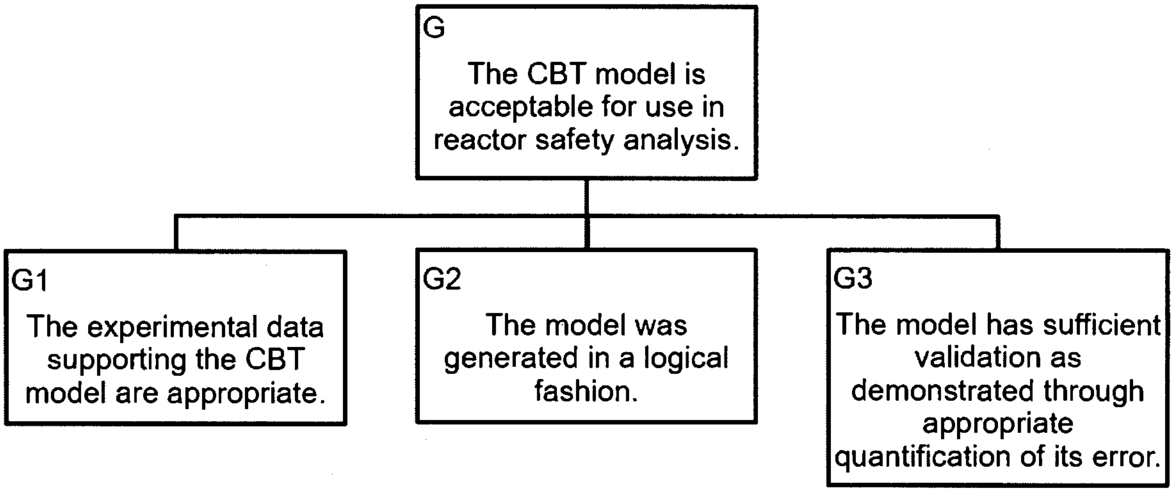
As stated by APS, VIPRE-APS is being applied in a manner consistent with that of the approved VIPRE-W. The NRC staff reviewed the evaluation model for the application of VIPRE-APS and made the following conclusions. The NRC staff concluded that the accident scenario identification process was not impacted by using VIPRE-APS as a replacement for VIPRE-W. The NRC staff concluded that the complete set of documentation describing VIPRE-APS was provided. The NRC staff concluded that the process of developing the evaluation model was adequate. The NRC staff concluded that the code assessment was adequate or was not impacted by the use of VIPRE-APS as a replacement for VIPRE-W. The NRC staff concluded that the uncertainty analysis was not impacted by the use of VIPRE-APS as a replacement for VIPRE-W. The NRC staff concluded that the quality assurance program was not impacted by the use of VIPRE-APS as a replacement for VIPRE-W. In summary, based on the above assessment, the NRC staff finds that the use of VIPRE-APS is an acceptable alternative to VIPRE-W for performing reactor safety analysis at Palo Verde, Units 1, 2, and 3.

3.3.2 Application of BHTP in a Different Subchannel Computer Code

The NRC staff has reviewed numerous implementations of CHF models such as the BHTP model. To perform this evaluation, the NRC staff used a framework similar to the framework used in the NRC staff's SE of the similar CHF and critical power models including the ORFEO-CHF model (Reference 57), the ACE/ATRIUM-11 CPR model (Reference 58), and the NuScale Power CHF model (Reference 59). More details about the framework applied in this review can be found in NUREG/KM-0013, "Credibility Assessment Framework for Critical Boiling Transition Models, A generic safety case to determine the credibility of critical heat flux and critical power models" (Reference 60).

For CHF models, the main goal is: "The critical boiling transition (CBT) is acceptable for use in reactor safety analyses." Based on the NRC staff's experience reviewing these models, a study of previous SEs and SERs, and multiple discussions with various industry experts, this goal is decomposed into the three subgoals given in Figure 1 below.

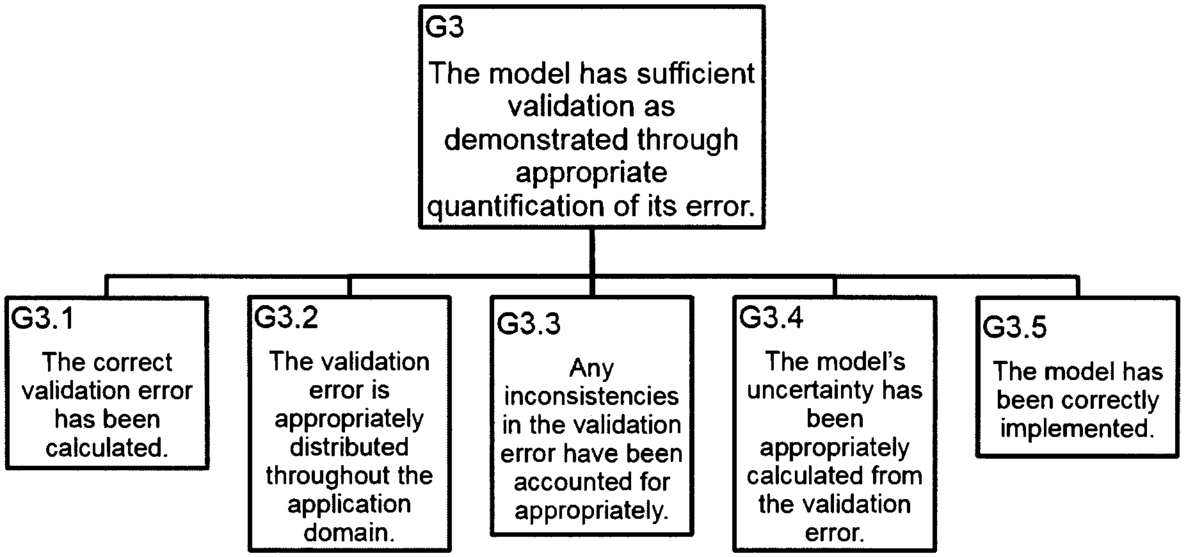
Figure 1: Decomposing G - Main Goal



The BHTP CHF model has already been approved, and therefore, the NRC staff has previously considered these three goals to have been met. Therefore, the implementation of the BHTP into VIPRE-W and VIPRE-APS would not impact the NRC staff's findings on G1 and G2, as those are independent of the computer code in which the model is applied. However, it would impact G3 as the model's error must be generated using a subchannel such as VIPRE. Therefore, the NRC staff will focus its review on ensuring that the validation of the BHTP CHF model does not change when applying the model with VIPRE-W and VIPRE-APS compared to the model's initial application with LYNXT (Reference 32).

Validation is the accumulation of evidence, which is used to assess the claim that a model can predict a real physical quantity (Reference 61). Thus, validation is a never-ending process as more evidence can always be obtained to bolster this claim. However, at some point, when the accumulation of evidence is considered sufficient to make the judgment that the model can be trusted for its given purpose, the model is said to be validated. Demonstrating that the model validation is appropriate is accomplished by using the five subgoals given in Figure 2 below.

Figure 2: Decomposing G3 – Model Validation



3.3.2.1 Validation Error

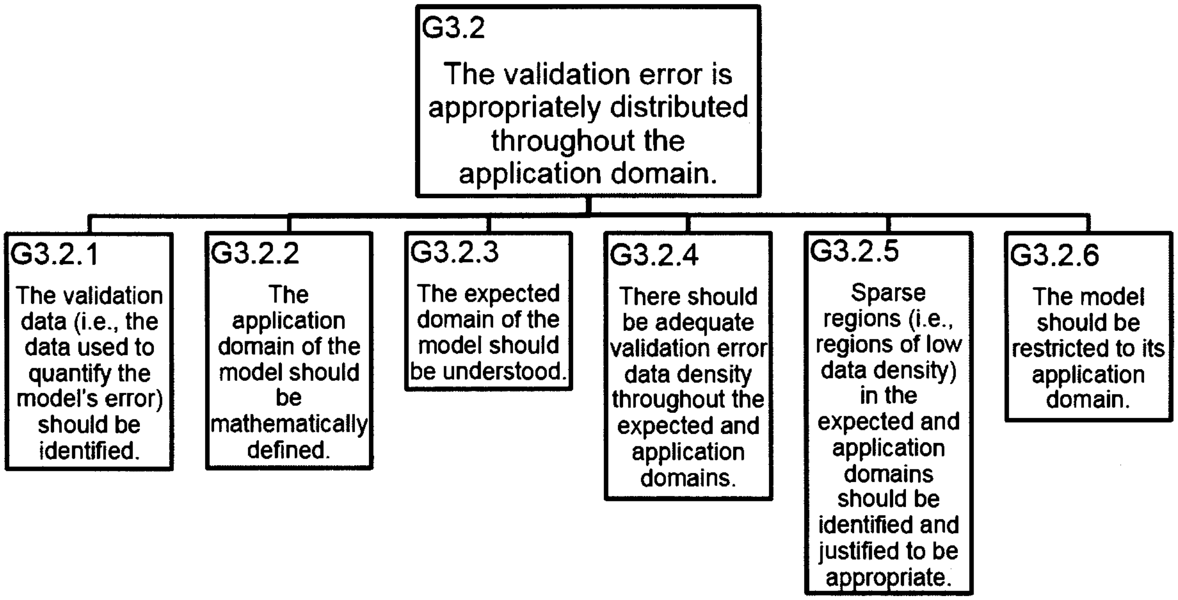
> The correct validation error has been calculated.
>
> G3.1, Review Framework for CBT Models

The validation error is obtained from a ratio of the measured CHF value and the predicted CHF value. However, there are method ways in which the measured and predicted CHF values could be determined. It is important that the same method (e.g., subchannel of lowest DNBR, subchannel where CHF was measured) was used in both the initial approval of BHTP and the applications of BHTP in VIPRE-W and VIPRE-APS. In response to BHTP RAI-01 by letter dated October 4, 2019 (Reference 5), APS stated that the test conditions are input into both versions of VIPRE and is input into the location of minimum DNBR. The heat flux in this subchannel was considered the "measured" value of CHF and the BHTP's prediction at this subchannel was considered the "predicted" value. APS further noted that this method is consistent with the method used in the initial development of the BHTP. Because the method of calculating the validation error in both VIPRE-W and VIPRE-APS is consistent with the method of calculating validation error in the initial approval of BHTP (Reference 32), the NRC staff has concluded that this goal has been met.

3.3.2.2 Data Distribution

The second subgoal in demonstrating that the model's validation is appropriate is to demonstrate that the data is appropriately distributed throughout the application domain. This is typically demonstrated using the six subgoals as given in Figure 3 below.

Figure 3: Decomposing G3.2 – Data Distribution



No further decompositions of the subgoals were deemed useful. Therefore, the evidence demonstrating that the following goals were met are provided below.

In general, the application of an approved CHF model into a new computer code should not impact the distribution of the validation error in the application domain, because that distribution is primarily determined by the experimental data used for validation. Therefore, instead of this section being focused on ensuring that the validation error is appropriately distributed, it is focused on ensuring that the application of the approved model into the new code did not change how the validation error is distributed.

3.3.2.2.1 Validation Data

> The validation data (i.e., the data used to quantify the model's error) should be identified.
>
> G3.2.1, Review Framework for CBT Models

In response to BHTP RAI-02 by letter dated October 4, 2019, APS confirmed that the same validation data used in the initial approval of BHTP was used in qualifying BHTP with VIPRE-W and VIPRE-APS. Because APS has identified the validation data, the NRC staff has concluded that this goal has been met.

3.3.2.2.2 Application Domain

> The application domain of the model should be mathematically defined.
>
> G3.2.2, Review Framework for CBT Models

In Table 5-1, "BHTP CHF Correlation Parameter Ranges," to the LAR dated July 6, 2018 (Reference 1), APS identified the application domain of the BHTP CHF model that will be used in both VIPRE-W and VIPRE-APS. However, this domain differs from the domain initially approved for BHTP. In the LAR, APS refers to Revision 1 of BAW-10241(P)(A), which expanded the domain of BHTP, but the NRC staff specified in its SE that the expansion was only applicable to the LYNXT computer code. In the supplemental response to BHTP RAI-03 by letter dated November 26, 2019 (Reference 6), APS provided the validation, which demonstrates that BHTP conservatively predicts CHF in the increased domain in both VIPRE-W and VIPRE-APS. Because of the validation provided, the NRC staff finds that the domain expansion is also acceptable in both VIPRE-W and VIPRE-APS. Because APS has identified this domain, the NRC staff has concluded that this goal has been met.

3.3.2.2.3 Expected Domain

> The expected domain of the model should be understood.
>
> G3.2.3, Review Framework for CBT Models

This review is focusing on the implementation of the BHTP CHF model into VIPRE-W and VIPRE-APS. Because the same validation data was used in the initial approval of the BHTP model as used here to demonstrate appropriate validation in VIPRE-W and VIPRE-APS, and because the model is being used over the same application domain, the NRC staff has determined that the expected domain would not be impacted by this implementation. Therefore, the NRC staff has concluded that this goal does not apply in this review.

3.3.2.2.4 Data Density

> There should be adequate validation error data density throughout the expected and application domains.
>
> G3.2.4, Review Framework for CBT Models

This review is focusing on the implementation of the BHTP CHF model into VIPRE-W and VIPRE-APS. Because the same validation data was used in the initial approval of the BHTP model as used here to demonstrate appropriate validation in VIPRE-W and VIPRE-APS, and the model is being used over the same application domain, the NRC staff has determined that the density of the data in the application or expected domain would not be impacted by this implementation. Therefore, the NRC staff finds that this goal does not apply in this review.

3.3.2.2.5 Sparse Regions

> Sparse regions (i.e., regions of low data density) in the expected and application domains should be identified and justified to be appropriate.
>
> G3.2.5, Review Framework for CBT Models

This review is focusing on the implementation of the BHTP CHF model into VIPRE-W and VIPRE-APS. Because the same validation data was used in the initial approval of the BHTP model as used here to demonstrate appropriate validation in VIPRE-W and VIPRE-APS, and the model is being used over the same application domain, the NRC staff has determined that the justification of use in any sparse regions would not be impacted by this implementation. Therefore, the NRC staff has concluded that this goal does not apply in this review.

3.3.2.2.6 Restricted Domain

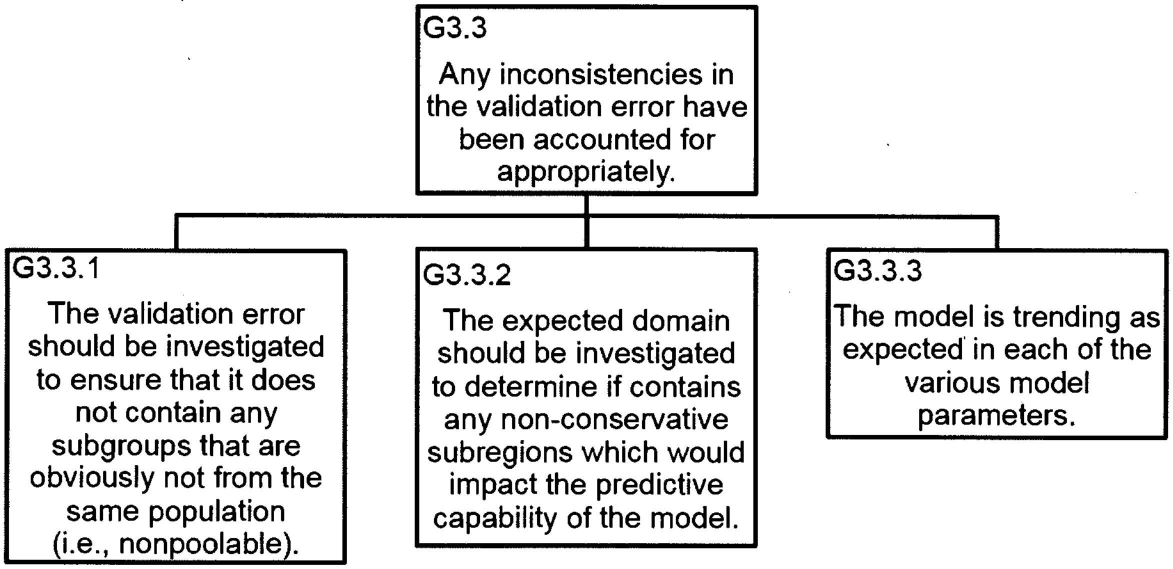
> The model should be restricted to its application domain.
>
> G3.2.6, Review Framework for CBT Models

In response to BHTP RAI-04 by letter dated October 4, 2019, APS confirmed that the BHTP correlation would be controlled through both automatic controls in the computer code or administrative controls for those parameters that are not automatically controlled. Because APS has identified how the models were restricted to its application domains, the NRC staff has concluded that this goal has been met.

3.3.2.3 Inconsistencies in the Model's Error

The third subgoal in demonstrating that the model's validation was appropriate is to demonstrate that the model error is consistent over the application domain. This is typically demonstrated using the three subgoals as given in Figure 4 below.

Figure 4: Decomposing G3.3 – Inconsistencies in the Model's Error



No further decompositions of the subgoals were deemed useful. Therefore, the evidence demonstrating the following goals that were met are provided below.

3.3.2.3.1 Poolability

> The validation error should be investigated to determine if it contains any sub-groups which are obviously not from the same population (i.e., not poolable).
>
> G3.3.1, Review Framework for CBT Models

In response to BHTP RAI-05 by letter dated October 4, 2019, APS provided an analysis of the various subgroups in the BHTP validation data base. APS examined the main subgroups of
[[

]]

Because APS has identified the main subgroups and demonstrated that those subgroups were poolable, the NRC staff has concluded that this goal has been met.

3.3.2.3.2 Non-Conservative Subregions

The expected domain should be investigated to determine if contains any non-conservative subregions which would impact the predictive capability of the model. G3.3.2, Review Framework for CBT Models

The potential for non-conservative analysis was previously reviewed by the NRC staff during its review of BHTP CHF model, and therefore, no further review is required.

In response to BHTP RAI-06, APS provided an analysis, which demonstrated that the predictions of LYNXT and VIPRE were very similar and that both methods approximately predicted non-conservative CHF data points at the same input properties. Further, while the NRC staff was aware of the potential for a slight non-conservative subregion in the unit cell testing, the staff found that the magnitude of the potential non-conservative behavior of that region was very small and would be more than accounted for in other conservatism in the development and assessment of the BHTP model. Therefore, the NRC staff has concluded that this goal has been met.

3.3.2.3.3 Model Trends

The model is trending as expected in each of the various model parameters. G3.3.3, Review Framework for CBT Models

In the licensee's LAR dated July 6, 2018, APS provided a single comparison of measured to predicted value for the BHTP model. In response to BHTP RAI-07 by letter dated October 4, 2019, APS provided plots of the predicted over measured ratios from VIPRE-W and VIPRE-APS versus key parameters (pressure, mass flux, local quality, inlet enthalpy, and shape factor). These plots demonstrated that the BHTP model in each subchannel code do not have any error trends with the key parameters.

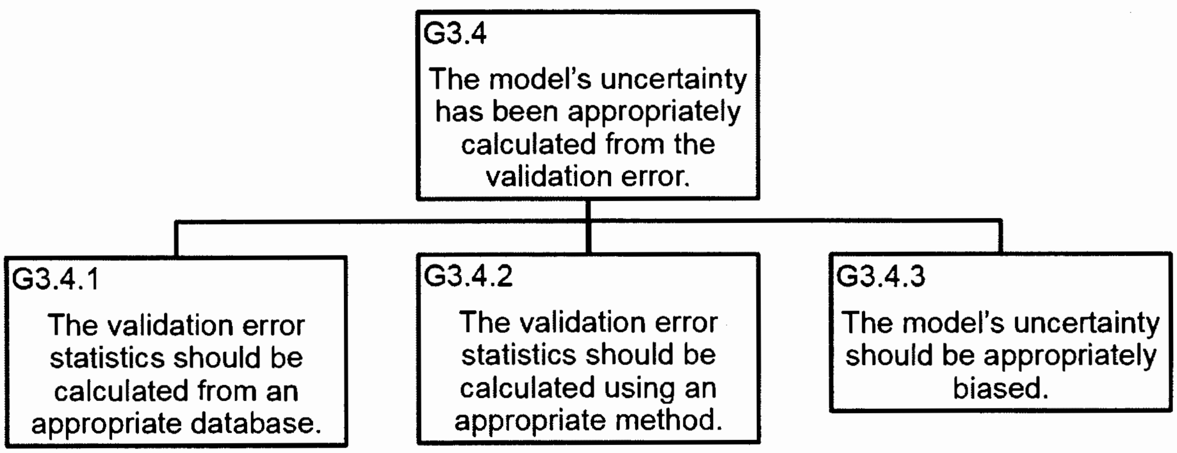
Additionally, in response to BHTP RAI-08 by letter dated October 4, 2019, APS provided a comparison between the predictions of BHTP in VIPRE-W and VIPRE-APS. It should be noted that this comparison is different from that provided in the LAR, as explained in an APS condition report. The new comparison demonstrates that both BHTP used in both VIPRE-W and VIPRE-APS result in the same predictions of CHF, within a very small variance at the fourth significant digit.

Because VIPRE-W and VIPRE-APS result in the same output for the same input and because the error of that output is independent of each key parameter, the NRC staff finds that the model is trending as expected. The NRC staff has concluded that this goal has been met.

3.3.2.4 Quantified Model Error

The fourth subgoal in demonstrating that the model's validation was appropriate is to demonstrate that the model error has been appropriately quantified over the application domain. This is typically demonstrated using the three subgoals as given in Figure 5 below.

Figure 5: Decomposing G3.4 – Quantified Model's Error



No further decompositions of the subgoals were deemed useful. Therefore, the evidence demonstrating the following goals were met as provided below.

3.3.2.4.1 Error Database

> The validation error statistics should be calculated from an appropriate database.
>
> G3.4.1, Review Framework for CBT Models

This review is focusing on the implementation of the BHTP CHF model into VIPRE-W and VIPRE-APS. The same validation data was used in the initial approval of the BHTP model as used here to demonstrate appropriate validation in VIPRE-W and VIPRE-APS. The validation error was based on the predictions of each of these computer codes compared to the measured

data. This comparison also included the extended range of BHTP as discussed in the supplemental response to BHTP RAI-03 dated November 26, 2019. The BHTP CHF model conservatively predicted data in this extended range. Further, that data was not used in generating the statistical DNB limit of the CHF model as ignoring the data for the generation of that limit resulted in a more conservative limit. Because APS provided the validation, which demonstrates that BHTP conservatively predicts CHF in both VIPRE-W and VIPRE-APS, and that comparison confirmed the use of the initially approved DNBR limit for BHTP, the NRC staff has concluded that this goal has been met.

3.3.2.4.2 Statistical Method

The validation error statistics should be calculated using an appropriate method.
G3.4.2, Review Framework for CBT Models

In the licensee's LAR, APS stated that they would use a previously approved **[[** **]]** method for determining the 95/95. Because the statistical method has been previously approved, the NRC staff has concluded that this goal has been met.

3.3.2.4.3 Appropriate Bias for Model Uncertainty

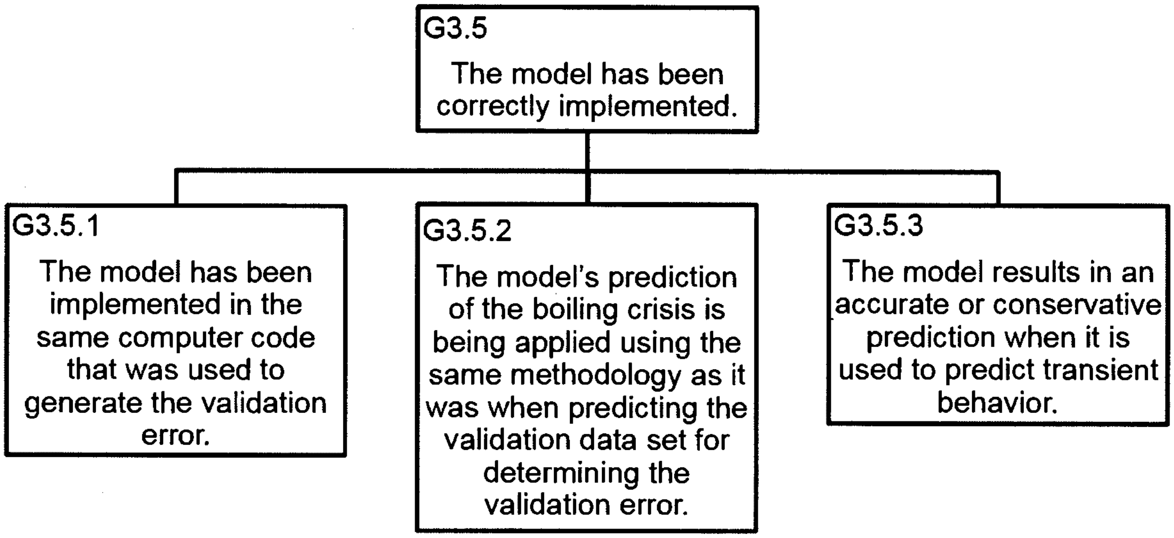
The model's error should be appropriately biased in generating the model uncertainty.
G3.4.3, Review Framework for CBT Models

Based on the data provided in the LAR and the RAI responses, the NRC staff did not find any additional bias needed in generating the model uncertainty. Because the statistical method used to determine the DNBR limit had been previously approved for BHTP and that same method was applied in this LAR, the NRC staff has concluded that this goal has been met.

3.3.2.5 Model Implementation

The fifth subgoal in demonstrating that the model's validation was appropriate is to demonstrate that the model will be implemented in a manner consistent with its validation. This is typically demonstrated using the three subgoals as shown in Figure 6 below.

Figure 6: Decomposing G3.5 – Model Implementation



No further decompositions of the subgoals were deemed useful. Therefore, the evidence demonstrating that the following goals were met are provided below.

3.3.2.5.1 Same Computer Code

> The model has been implemented in the same computer code which was used to generate the validation error.
>
> G3.5.1, Review Framework for CBT Models

The purpose of this part of the review was to re-validate the BHTP CHF model in VIPRE-W or VIPRE-APS. For VIPRE-APS, the BHTP model is being implemented in the same computer code that was also used to perform the error quantification given in this section. Therefore, the NRC staff has determined that this goal has been satisfied for VIPRE-APS.

For VIPRE-W, the BHTP model is being implemented in the VIPRE-W computer code, but with different model options (i.e., closure models) than were in the initial approval of VIPRE-W. Therefore, in response to BHTP RAI-09 by letter dated October 4, 2019, APS provided clarification that when BHTP was used in VIPRE-W, the modeling options used were those given in the LAR. While APS is not using the same computer code, it has re-validated the modeling options for BHTP in VIPRE-W (provided in response to BHTP RAI-07) and has demonstrated that BHTP used in VIPRE-APS and VIPRE-W results in the same predictions for the same input. The NRC staff finds that this re-validation and demonstration of results satisfies the intent of this goal. Therefore, the NRC staff has concluded that this goal has been satisfied for VIPRE-W.

3.3.2.5.2 Same Evaluation Framework

> The model's prediction of the critical boiling transition is being applied using the same methodology as it was when predicting the validation error set for determining the model's error.
>
> G3.5.2, Review Framework for CBT Models

The purpose of this part of the review was to revalidate the BHTP CHF model and how it is applied in the evaluation framework of VIPRE-W or VIPRE-APS. In response to BHTP RAI-10 in a letter dated October 4, 2019 (Reference 5), APS confirmed that future uses of the BHTP CHF correlation in either VIPRE-W or VIPRE-APS will be used in a manner consistent with that used when performing the validation analysis presented above. Therefore, the NRC staff has concluded that this goal has been satisfied.

3.3.2.5.3 Transient Prediction

> The model results in an accurate or conservative prediction when it is used to predict transient behavior.
>
> G3.5.3, Review Framework for CBT Models

This review is focusing on the implementation of the BHTP CHF model into VIPRE-W and VIPRE-APS. Because the model is being used in a similar manner as approved in its initial approval, and because the model is being used in a similar manner as other CHF models, the NRC staff has determined that model's use in transient predictions would not be impacted by this implementation. Therefore, the NRC staff has concluded that this goal does not apply in this review.

3.3.2.6 Conclusion on the Implementation of BHTP in VIPRE-W and VIPRE-APS

Based on the above information, the NRC staff concludes that the validation of BHTP for use in VIPRE-W and VIPRE-APS has been demonstrated through the quantification of its error when compared with experimental data. Therefore, the NRC staff finds that the use of BHTP with VIPRE-W and VIPRE-APS for predicting the behavior of CE 16HTP™ fuel is acceptable.

It should be noted that towards the end of this review, Framatome discovered an issue in the assessment of Test 51 in the LYNXT database (Reference 7). This issue had a small impact on the comparison between LYNXT, VIPRE-W and VIPRE-APS predictions of the CHF data from Test 51. However, correction of the issue would slightly improve the comparison between VIPRE-W and VIPRE-APS and the LYNXT results. APS has entered this issue into its corrective action program. The NRC staff finds that this issue did not change any of the limits, which resulted from the VIPRE analysis. APS and Framatome are addressing this issue appropriately. Most importantly, the NRC staff finds that this issue did not impact any of the NRC staff's conclusions given above.

3.3.3 Modeling of HTP™ Fuel Below the First HTP™ Grid Spacer

In Section 5.3.1, "CHF Correlations for Use Below First HTP™ Grid," in Attachment 8 to the LAR dated July 6, 2019 (Reference 1), APS stated that there was no need to use a specific CHF model below the first HTP™ grid. However, in response to CHF RAI-01 of the letter dated October 4, 2019 (Reference 5), APS amended its LAR and stated that they would use the CE-1 CHF correlation for DNB analysis in the region below the first HTP™ grid spacer for CE 16 HTP™ fuel. APS performed its own analysis, which demonstrated that minimum axial peak power location would occur above the first HTP™ grid and therefore, the minimum DNBR location would also be above the first HTP™ grid. Further, APS forced the location below the first HTP™ grid to result in the minimum DNBR and had to force an extreme and unrealistic power shape to do so.

Further, Framatome confirmed that **[[**

]].

CHF is extremely unlikely to occur below the first HTP™ grid. Should analysis in that region be required, the CE-1 correlation would result in a conservative prediction of CHF. Therefore, the NRC staff finds that the use of the CE-1 correlation below the first HTP™ grid is acceptable.

3.3.4 Use of WLOP CHF model

In the LAR, APS stated that the WLOP CHF model (Reference 48) could be used to predict the CE 16 HTP™ fuel provided that the fuel fell within the application domain provided in Table 5-3 in Attachment 8 to the LAR. However, the APS responses to CHF RAI-02 and CHF RAI-03 dated October 4, 2019 (Reference 5), stated that APS would use the Macbeth CHF correlation for CE 16 HTP™ fuel and not the WLOP CHF correlation. Because APS is no longer requesting permission to use WLOP to perform analysis for CE 16 HTP™ fuel, the NRC staff has not reviewed that request.

3.3.5 Mixed Core Methodology

Section 4.1.2, "Mixed Cores Containing Two Fuel Types," of TR WCAP-15306-NP-A (Reference 45) references multiple documents describing the approved mixed-core methodology approved for VIPRE-W. In the LAR, APS described its mixed-core approach as being different from that of VIPRE-W. In response to MIX RAI-01 to the letter dated October 4, 2019, APS provided additional details on its mixed-core method. Instead of relying on penalties determined for different transition patterns (as in VIPRE-W), APS will model the cycle-specific mixed-core directly. APS applies a **[[**

[1] VIPRE refers to VIPRE-W and VIPRE-APS.

]]

Since APS's mix-core methodology is an extension of its currently approved process, which applies a [[ ]] VIPRE process, and because [[

]], the NRC staff finds that the mixed-core methodology proposed by APS is appropriate for calculating traditional mixed cores. Note, as detailed in MIX RAI-01, as well as other RAIs, the NRC staff did not review the mixed-core methodology's application to mixed fresh batches. The NRC staff would consider such an application of this methodology beyond the scope of what was approved, and further NRC review and approval would be required.

3.3.6 Application of COLSS/CPCS to Generate Setpoints for CE 16 HTP™ Fuel

Section 11, "COLSS/CPCS Setpoints Analysis," in Attachment 8 of the LAR dated July 6, 2018, describes the COLSS/CPCS setpoints process for implementing CE 16HTP™ fuel. The NRC staff had concerns with the COLSS/CPCS setpoints process, which was briefly mentioned in the LAR. Further detail was provided in response to NRC Question 7 of the letter dated October 18, 2018 (Reference 2). In the LAR, the licensee states, in part:

> Therefore, the MSCU [Modified Statistical Combination of Uncertainties] methodology as described in CEN- 356(V)-P-A, Revision 01-P-A, "Modified Statistical Combination of Uncertainties" [Reference 62] is applicable to setpoint analysis of the CE 16HTP™ fuel.

The licensee also states, in part:

> An available alternative for implementation of Framatome CE 16 HTP™ fuel is to use the MSCU process steps as augmented by WCAP-16500-P-A, Supplement 1, Revision 1, "Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)" [Reference 63].

In the supplement to the LAR dated October 18, 2018, the licensee further states, in part:

> APS application of the COLSS/CPCS setpoints methodology to Framatome fuel is as follows:
>
> - For full cores with Framatome fuel: Since Framatome fuel does not have multiple CHF correlations along the axial length of the fuel assembly (except for the bottom portion of the fuel assembly where DNB is not significant) the original setpoints methodology can be used. Use of the augmented process as described for NGF fuel may also be used.
>
> - For mixed cores with CE 16 STD and Framatome fuel: Since both CE 16 STD and Framatome fuel do not have multiple CHF correlations along the axial length of the fuel assembly (except for the bottom portion of the fuel assembly where DNB is not significant) the original setpoints methodology can be used. Use of the augmented process as described for NGF fuel may also be used if desired.

The NRC staff concerns are that (1) neither the original MSCU process nor the augmented MSCU process is directly applicable to CE 16 HTP™ cores and (2) it has not yet been demonstrated that applying either of these methods will produce acceptable results.

During the second regulatory audit, the NRC staff performed a detailed review of the APS calculation documenting the demonstration setpoints analyses. The audit concluded that:

1. The original MSCU methodology is not applicable to CE 16 HTP™ fuel and will likely produce non-conservative setpoints.

2. Strict application of the modified MSCU methodology may produce non-conservative setpoints. These methods are sensitive to axial shape index breakpoints, which are fuel design-specific.

3. Application of the modified MSCU methodology to mixed-core configurations results in a significant loss of DNB thermal margin (relative to a full core of either STD, NGF, or HTP™) due to worse case statistics and/or application of limiting ranges and adjustments.

To address the NRC staff concerns identified during the audit, SNPB RAI-6 requested further justification for the use of the CE 16NGF augmented eight-step process for cores containing CE 16 HTP™ fuel assemblies. In response to SNPB RAI-6 by letter dated October 4, 2019, APS described a revision to the augmented eight-step process to capture the unique characteristics of the CE 16 HTP™ fuel. APS also stated that the original MSCU process would not be used. Further information regarding the COLSS/CPCS setpoints process is provided in response to SNPB RAI-3, -4, and -5 by letter dated October 4, 2019. Therefore, the revisions to the augmented eight-step process addressed the NRC staff's concerns regarding applicability to CE 16 HTP™ fuel.

During the second regulatory audit, the NRC staff reviewed preliminary calculations using the new setpoints methods. Final, quality assured calculations were reviewed during the third

regulatory audit. These calculations demonstrate that the new setpoint methods produce conservative BERR1 penalty factors for the CE 16 HTP™ fuel assembly. Based upon the audit and review of the APS demonstration analysis, the NRC staff found that new setpoint methods produce acceptable results for CE 16 HTP™ fuel. Therefore, the NRC staff finds the COLSS/CPCS setpoints process described in SNPB RAI-6 acceptable.

3.3.7 Thermal-Hydraulic Compatibility

In Section 5.7, "Thermal-Hydraulic Compatibility," in Attachment 8 to the LAR, APS discussed seven areas that were used to demonstrate the thermal-hydraulic compatibility of the CE 16 HTP™ design with CE 16 STD and CE 16NGF fuel. This assessment included consideration of the core pressure drop, bypass flow, crossflow velocity, RCS flow rate, control element assembly (CEA) drop time, fuel rod bow, and guide tube heating.

In the LAR, APS stated that the pressure drop due to a full core of CE 16 HTP™ would be greater than a full core of CE 16 STD but less than a full core of CE 16NGF. In response to TH RAI-01 by letter dated October 4, 2019, APS provided a description of the three different fuel types, as well as its loss coefficients, and a schematic showing the locations of each grid spacer. Because the fuel assemblies used by APS had grid spacers at similar locations and the loss coefficients of those grid spacers was also similar, the NRC finds that the fuel assemblies are thermal-hydraulically similar.

Since APS confirmed that any full or partial cores containing CE 16 HTP™ would be bounded by the current analysis from the perspective of a pressure drop, the NRC staff finds that HTP™ fuel will not adversely impact the core pressure drop analysis.

Since APS confirmed that the bypass value of 3 percent used in the design analysis remains bounding, the NRC staff finds that HTP™ fuel will not adversely impact bypass flow.

Since APS is considering the crossflow velocities caused by CE 16 HTP™ fuel in its mechanical analysis, the NRC staff finds the treatment of crossflow flow acceptable.

In the LAR, APS stated that the change in RCS loop flow will fall between the loop flow for CE 16 STD and CE 16NGF. Since APS confirmed that any change in loop flow would be bounded by the current analysis, the NRC staff finds that HTP™ fuel will not adversely impact the RCS flow rate.

In the LAR, APS stated that the CEA drop time would not be impacted due to the similarity of CE 16 HTP™ fuel with CE 16 STD and CE 16NGF. Further, in response to TH RAI-02 by letter dated October 4, 2019, APS confirmed that CEA drop times are tested following each reload. Since APS is testing the CEA drop times following each reload, the NRC staff finds that HTP™ fuel will not adversely impact control rod scram time.

Since APS is using an approved method to evaluate the impact of rod bow on CE 16 HTP™ fuel, the NRC staff finds the treatment of rod bow acceptable.

In response to TH RAI-03 by letter dated October 4, 2019, APS provided details on an analysis that provided an upper limit to the control rod LHGR needed to preclude boiling in the guide tubes, and then demonstrated that its own control rods would remain below that limit.

3.3.8 Application of other CHF Models into VIPRE-W, VIPRE-APS, and CE Thermal Online Program (CETOP-D)

In the licensee's LAR dated July 6, 2018, APS requested the ability to implement other approved CHF correlations into its subchannel codes. In the licensee's response to CHF RAI-04, the licensee withdrew this request by letter dated November 26, 2019. The NRC staff acknowledged this withdrawal and did not consider the application of other CHF models into subchannel codes as part of this review.

The NRC staff has reviewed the implementation of the BHTP model into VIPRE-APS and VIPRE-W and found both implementations acceptable. The NRC staff has also audited that the setpoint methodology produces conservative penalty factors for CE 16 HTP™ fuel. Since APS has demonstrated that the correction factors in CETOP-D are appropriately calculated compared to VIPRE and has demonstrated that VIPRE analysis is realistic compared to the experimental data, the NRC staff finds the use of the BHTP model in CETOP-D acceptable.

3.4 Non-LOCA Transient Analysis

To quantify the effect of introducing the Framatome CE 16HTP™ fuel design into the Palo Verde safety analysis licensing basis, all Palo Verde UFSAR Chapter 15 (Reference 17) Non-LOCA transient analyses were evaluated.

In total, the Palo Verde UFSAR Chapter 15 transients consist of thirty-five non-LOCA analyses and two additional considerations from the UFSAR Chapter 15 appendices.[2] It has been previously found that four of these analyses are not applicable to System 80 PWRs such as Palo Verde, and therefore require no further consideration. These events are the "Steam Pressure Regulator Failure" (UFSAR Section 15.2.5), "Flow Controller Malfunction Causing a Flow Coastdown" (UFSAR Section 15.3.2), "Flow Controller Malfunction Causing an Increase in BWR Core Flow" (UFSAR Section 15.4.5) and "Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)" (UFSAR Section 15.6.4). Two of the Chapter 15 events (i.e., LOCA and Radiological Material Release), are addressed in separate sections of this SE.

The licensee determined that ten of the non-LOCA analyses are bounded by another related Chapter 15 event, and therefore, these ten analyses do not need to be evaluated alongside the bounding event. The bounded analyses are summarized in the table below along with the event(s) that bound them.

[2] The "additional considerations" referred to are associated with Palo Verde UFSAR Appendix 15D, "Analysis Methods for Loss of Primary Coolant Flow," and Appendix 15E, "Limiting Infrequent Event."

Table 2: Bounded Events from UFSAR Chapter 15

UFSAR Section	Event	Bounding Chapter 15 Event(s)
15.1.1	Decrease in Feedwater Temperature	This event is bounded by the Increase in Main Steam Flow event (UFSAR Section 15.1.3), the Inadvertent Opening of a Steam Generator Atmospheric Dump Valve event (UFSAR Section 15.1.4) and the loss of flow from a SAFDL event (UFSAR Section 15.E).
15.1.2	Increase in Main Feedwater Flow	This event is bounded by the Increase in Main Steam Flow event (UFSAR Section 15.1.3), the Inadvertent Opening of a Steam Generator Atmospheric Dump Valve event (UFSAR Section 15.1.4) and the loss of flow from a SAFDL event (UFSAR Section 15.E).
15.2.1	Loss of External Load	This event is bounded by the Loss of Condenser Vacuum event (UFSAR Section 15.2.3). This event with a loss of offsite power results in an event similar to the Total Loss of Reactor Coolant Flow (UFSAR Section 15.3.1).
15.2.2	Turbine Trip	This event is bounded by the Loss of Condenser Vacuum event (UFSAR Section 15.2.3). This event with a loss of offsite power results in an event similar to the Total Loss of Reactor Coolant Flow event (UFSAR Section 15.3.1).
15.2.4	Main Steam Isolation Valve Closure	This event is bounded by the Loss of Condenser Vacuum event (UFSAR Section 15.2.3). This event with a Loss of Offsite Power results in an event similar to the Total Loss of Reactor Coolant Flow event (UFSAR Section 15.3.1).
15.2.6	Loss of Non-Emergency AC [Alternating Current] Power to the Station Auxiliaries	This event is bounded by the Loss of Condenser Vacuum event (UFSAR Section 15.2.3) and the Total Loss of Reactor Coolant Flow event (UFSAR Section 15.3.1).
15.2.7	Loss of Normal Feedwater Flow	This event is bounded by the Loss of Condenser Vacuum event (UFSAR Section 15.2.3). This event with a loss of offsite power results in an event similar to the Total Loss of Reactor Coolant Flow event (UFSAR Section 15.3.1).
15.3.3	Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power	This event is bounded by single Reactor Coolant Pump Shaft Break with Loss of Offsite Power (UFSAR Section 15.3.4)

UFSAR Section	Event	Bounding Chapter 15 Event(s)
15.5.2	CVCS [Chemical and Volume Control System] Malfunction-Pressurizer Level Control System Malfunction with Loss of Offsite Power	This event is bounded by the Total Loss of Reactor Coolant Flow event (UFSAR Section 15.3.1).
15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	This event is bounded by the Postulated Radioactive Releases Due to Liquid-Containing Tank Failures (UFSAR Section 15.7.3)

3.4.1 Evaluated UFSAR Chapter 15 Non-LOCA Events

The NRC staff's review of the evaluated Palo Verde UFSAR Chapter 15 non-LOCA transients focuses on ensuring that approved methodologies are implemented appropriately and that results (overall transient system response, DNBR, LHGR, and predicted fuel failures) are acceptable. The following subsections address additional specific considerations for select Chapter 15 transients.

The licensee stated that the only changes required to account for use of CE 16HTP™ fuel were to DNB propagation, use of statistical convolution to determine fuel failure, and CEA ejection. The licensee stated that no other modifications to the currently approved methodology, as discussed in the UFSAR Chapter 15 event sections, were required for evaluating non-LOCA transients for CE 16HTP™ fuel.

3.4.1.1 LHGR Acceptance Criteria

For Chapter 15 events in Palo Verde UFSAR Sections 15.1.3, 15.1.4, 15.1.5, 15.1.6, and 15.2.3, the licensee states, in part, in its LAR dated July 6, 2018, that "The transient Linear Heat Rate (LHR) will not exceed 21.0 kW/ft [kilowatts/foot]. Therefore, the fuel centerline melt temperature will not be exceeded." However, Table 2-3, "CFM Rod Local LHGR Limits," in Attachment 10 to the LAR dated July 6, 2018 (Reference 1), indicates that 21.0 kW/ft will not preclude centerline melt for [[

]]. The NRC staff requested additional information to demonstrate that fuel centerline melt temperature will not be exceeded for Chapter 15 events in UFSAR Sections 15.1.3, 15.1.4, 15.1.5, 15.1.6, and 15.2.3. In the licensee's response to SRXB RAI-1 by letter dated October 4, 2019 (Reference 5), APS explained that a radial fall-off credit will be employed to ensure that the "effective" power-to-centerline melt for the gadolinia fuel rods and the high burnup UO_2 fuel rods is also above 21 kW/ft.

The licensee further states that since the power at fuel centerline melt decreases as rod average burnup increases and gadolinia content increases, the core will be designed to ensure the peaking factors in the gadolinia fuel rod and the high burnup UO_2 fuel rod are proportionately lowered to ensure that the "effective" power-to-centerline melt limit is not less than the local power density trip setpoint of 21 kW/ft. As the gadolinia loading is increased, the Uranium-235 enrichments will decrease such that the performance of the gadolinia bearing fuel rod will be similar to a UO_2 fuel rod.

For events in Palo Verde UFSAR Sections 15.4.2 and 15.4.3, the licensee states, in part, in the LAR, that, "The maximum LHGR will remain below the value that causes peak centerline melt temperature (TS 2.1.1.2 limit)." However, the TS 2.1.1.2 limit is provided in terms of peak fuel centerline temperature. Further, there are separate limits for Westinghouse supplied fuel and for Framatome supplied fuel, which vary as a function of burnup. Table 2-3 in Attachment 10 of the LAR (Reference 1) indicates that **[[ ]]**. The NRC staff requested additional information to demonstrate how LHGR values are controlled to ensure that peak centerline melt is precluded and that variations in fuel type and **[[ ]]** are accounted for in the calculations for UFSAR Sections 15.4.2 and 15.4.3. In the licensee's response to SRXB RAI-2 by letter dated October 4, 2019, it is explained that the radial fall-off credit will be used to ensure that the "effective" power-to-centerline melt for the gadolinia fuel rods and the high burnup UO_2 fuel rods is also above 21 kW/ft. The licensee goes on further to explain,

> To determine the minimum power-to-centerline melt, each specific fuel rod type resident in the Palo Verde units has been analyzed with the appropriate NRC approved fuel performance code. The COPERNIC code was employed to predict the linear heat rates at which the onset of fuel centerline melting occurs for the Framatome CE 16HTP™ fuel rods. FATES3B calculations have been performed for Westinghouse CE 16NGF UO_2, CE 16NGF ZrB_2 [zirconium diboride], CE 16 STD UO_2, and CE 16 STD erbia fuel rods.

The NRC staff concludes that the LHGR has been adequately addressed and the licensee's approach is acceptable.

3.4.1.2 DNB Propagation

Section 6.2, "DNB Propagation," in Attachment 8 to the LAR dated July 6, 2018, describes the DNB propagation methods employed for M5® cladding. Based upon a comparison of EDGAR M5® cladding ballooning and burst tests to INTEG predictions based on Zircaloy-4 properties, APS concludes that the existing methods are valid for M5® cladding.

To capture APS's decision regarding the requested approval for DNB propagation, the NRC staff issued SRXB RAI-3, requesting further explanation and justification. During the third regulatory audit, the NRC and APS staff discussed the initial response to SRXB RAI-3 by letter dated October 4, 2019. The NRC staff had concerns that the response suggested a new bases for a proposed 5.0 second allowable time-in-DNB for M5® cladding. Based upon discussions during the second regulatory audit, the NRC staff was expected that the existing bases for the 4.5 seconds allowable time-in-DNB would be adopted for M5® cladding, with appropriate justification. After discussions at the third audit, it was agreed that the response to SRXB RAI-3 would be supplemented to provide the justification for applying the existing screening criteria (4.5 seconds in DNB) to M5® cladding.

The supplemental response to SRXB RAI-3 by letter dated November 26, 2019, provides further justification, including EDGAR ballooning strain rate data for M5® cladding. Based upon the material presented in this RAI response, the NRC staff finds the 4.5 second allowable time-in-DNB screening criteria acceptable for CE 16HTP™ fuel assemblies with M5® cladding.

3.4.1.3 Fuel Failure Prediction Using DNB Statistical Convolution

Section 6.3 of the LAR dated July 6, 2018, describes the application of the approved DNB statistical convolution methods to the CE 16HTP™ fuel assemblies. During the second regulatory audit, the NRC and APS staff discussed the process for assessing fuel rod failures during postulated accidents with mixed-core configurations. It was not clear to the NRC staff that the proposed method would capture fuel previously labeled as non-limiting. For example, the core TH analyses may define the NGF fuel bundles as non-limiting in a core containing a majority of lower exposure HTP™ fuel bundles. As a result of this designation, DNB probability distribution functions would not be created for the NGF bundle. The NRC staff concerns were that under accident conditions, any fuel bundle has the potential to experience fuel rod failure. As such, predicted fuel rod failures, which ignore populations of fuel bundles, may be non-conservative.

To address NRC staff concerns regarding the mixed-core application of DNB statistical convolution, SRXB RAI-5 requested further discussion and justification. In response, APS stated in its letter dated November 26, 2019, that fuel failure calculations are performed in a branched fashion for assembly types identified as potentially limiting in the core TH screening process. For design-basis accidents, which experience a more global transient response where non-limiting assemblies are unlikely to experience fuel damage, this approach is reasonable and appropriate. For the local power excursion associated with a control rod ejection, damage to low power fuel rods may occur. As a result, DNB statistical convolution would address all fuel types. Based on the information presented in Section 6.3, "Fuel Failure Prediction Using DNB Statistical Convolution," in Attachment 8 to the LAR, as supplemented in response to SRXB RAI-5, the NRC staff finds the DNB statistical convolution methods acceptable for application to CE 16HTP™ fuel and transition cycles.

3.4.1.4 Cladding Related Models in the Non-LOCA Transient Evaluation Models

The LAR presented information to demonstrate that the impact of M5® cladding thermal conductivity and cladding specific heat were appropriately considered in the non-LOCA transient analysis system response computer codes CENTS and HERMITE. The NRC staff determined that the information demonstrated that **[[**

]] for non-LOCA transients other than the CEA ejection transient. The HERMITE code requires input of the fuel-to-clad gap coefficient of conductance. The NRC staff issued an RAI to determine if the COPERNIC code, approved for the calculation of fuel-to-clad gap coefficient of conductance for CE 16HTP™ fuel was used to provide the fuel-to-clad gap coefficient of conductance input for the HERMITE code when modeling CE 16HTP™ fuel. In response to SRXB RAI-4 by letter dated October 4, 2019, the licensee confirms that the specific inputs to either the HERMITE or CENTS code for CE 16HTP™ fuel are generated by the Framatome fuel performance code COPERNIC. In addition, COPERNIC includes any effects of the gadolinium.

The licensee further explains that **[[**

]]

The NRC staff concludes that the cladding related models are adequately addressed and are acceptable for use in non-LOCA transient evaluation models.

3.4.1.5 Fuel Design Specific Input Parameters in UFSAR Chapter 15 Analyses

Analysis of Chapter 15 events in Palo Verde UFSAR Sections 15.4.8 and 15.7.4 and UFSAR Appendix 15.E require values of gap conductance (Hgap) and fuel temperature coefficient. The NRC staff reviewed the analyses for these three transients during a second regulatory audit (Reference 12) to confirm that values of these fuel specific properties were provided by the COPERNIC code, which has been approved to accurately predict fuel parameters for Framatome CE 16HTP™ fuel.

3.4.1.6 CEA Ejection Analysis

The [[ ]] data are user input to the core heat conduction model of the STRIKIN-II code. Zircaloy-4 is the original cladding material used in Westinghouse fuels, so Zircaloy-4 material properties are the default data for the STRIKIN-II computer code. [[

]]
However, for the CEA transient, in which the cladding temperature may enter the phase change range, the temperature-dependent values of heat capacity for M5® must be modeled via user input to the STRIKIN-II code. [[
]]

Fuel performance data for input into STRIKIN-II was provided by the COPERNIC code instead of the FATES3B code. Benchmark calculations were performed at hot full power at various cladding and burnup points to demonstrate the differences between the FATES3B and COPERNIC codes as well as the impact of thermal conductivity degradation (TCD) modeled in COPERNIC code. Benchmark calculations have showed that (1) the COPERNIC based output data are acceptable for use in STRIKIN-II code; (2) modeling differences between two codes can lead to differences in centerline temperature at low burnups; (3) for burnup greater that 12 GWd/MTU, the two codes predict consistent temperatures with COPERNIC temperature higher due to TCD effects; and (4) the limiting hot rod burnup point for STRIKIN-II analysis represent the corresponding burnup point of the radial falloff curve to fully absorb the TCD effect.

CEA ejection analysis is updated for a full core of Framatome CE 16HTP™ fuel at the Palo Verde core. The analysis was performed using COPERNIC code and VIPRE code with BHTP CHF correlation instead of STRIKIN-II with CE-1 CHF correlation. The analysis calculated maximum radially averaged enthalpy, fuel centerline temperature, and evaluated DNBR using STRIKIN-II and VIPRE codes. Fuel enthalpy evaluations were performed at hot full power and hot zero power as well as for intermediate power levels. The results reported from the CEA analysis (Table 6-3 of the LAR) indicate that maximum radially averaged enthalpy is less than the acceptance criteria specified in Appendix B of SRP Section 4.2 (Reference 19) for cladding failure and core coolability. Results from the CEA analysis show that the peak fuel centerline

temperature is less than the fuel melt temperature for Framatome fuel (which varies as a function of burnup). The analysis shows that the average enthalpy and peak fuel centerline criteria are met for both the resident CE 16NGF and CE 16 STD fuel designs.

3.4.1.7 Fuel Handing Accident

The fuel handling accident (FHA) is presented in Section 6.7 in Attachment 8 of the LAR dated July 6, 2018, to address the applicability of the current analyses of record (AORs) for the CE 16HTP™ fuel design. The FHA may occur in either the fuel building or inside the containment. Relative to the existing FHA dose AORs, the introduction of Framatome CE 16HTP™ fuel could only affect the FHA source term. The FHA source term is dependent on the fuel assembly isotope inventory in the fuel rod gap spaces, the number of damaged fuel rods, and the fuel rod gap release fractions.

Framatome fuel design parameters (e.g., initial uranium mass, burnup, power factors, and operating histories) are essentially equivalent to those for current CE 16 STD fuel. The change from other zirconium-based cladding to M5® cladding is not only a minor contributor to the source term, but since the elemental compositions of the claddings are similar, the impact of the cladding on the source terms is insignificant and will not change the core isotopic distribution assumed in post-accident conditions. Additionally, system model input changes due to Framatome fuel are small and the impacts to overall transient system responses are insignificant. Based on these facts, CE 16HTP™ fuel has been found to introduce no changes that would affect UFSAR Chapter 15 source terms.

The FHA is currently evaluated assuming damage to all 236 fuel rods in the dropped fuel assembly. An evaluation confirms that damage to all 236 fuel rods in the dropped fuel assembly bounds the potential damage to a dropped CE 16HTP™ fuel assembly.

Amendment No. 153 for Palo Verde (Reference 64) approved a deviation from RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (Reference 65) to allow the use of peak assembly average fuel pin pressure is less than 1200 pounds per square inch gauge (psig) in place of maximum fuel rod pressurization is 1200 psig. This approach allows a few fuel rods to exceed the 1200 psig maximum pressurization while still maintaining the conservative iodine decontamination factor value specified by RG 1.25. Fuel rod pressures for Framatome CE 16HTP™ fuel with M5® cladding and with and without gadolinia, based on 72 hours of incore hold time, have been calculated using the approved COPERNIC fuel rod design computer code. The peak assembly average fuel pin pressure for Framatome CE 16HTP™ fuel is less than 1200 psig, and consequently the CE 16HTP™ fuel rod gap release fractions are bounded by the CE 16 STD and CE 16NGF fuel rod gap release fractions.

Therefore, an FHA involving the dropping of a Framatome CE 16HTP™ fuel assembly is no more severe than an FHA involving the dropping a CE 16 STD or CE 16NGF fuel assembly.

3.4.2 Conclusion on UFSAR Chapter 15 Non-LOCA Events

The NRC staff has reviewed the Palo Verde UFSAR Chapter 15 analyses and appendix considerations provided by APS. For each analysis, the applicability to Palo Verde, the impact of the fuel design change, and the potential for bounding with other analyses were evaluated.

The NRC staff has concluded that the licensee has appropriately incorporated new fuel design information. Therefore, the NRC concludes that APS has adequately demonstrated that the Palo Verde UFSAR Chapter 15 events have been appropriately analyzed, and the results are acceptable and support the transition to Framatome CE 16HTP™ fuel at Palo Verde.

3.5 LOCA Analysis

The NRC regulations require that licensees of operating LWRs analyze a spectrum of accidents involving the loss of reactor coolant to assure adequate core cooling under the most limiting set of postulated design-basis conditions. The postulated spectrum of LOCAs ranges from scenarios with leakage rates just exceeding the capacity of normal makeup systems up through those involving rapid coolant loss from the complete severance of the largest pipe in the RCS.

To support the planned loading of Framatome HTP™ fuel at Palo Verde, the licensee has proposed to add Framatome analysis methods for the postulated spectrum of small- and large-break LOCA events to the list of COLR references in Section 5.6.5.b of the TSs. To this end, the licensee's submittal includes two licensing reports summarizing its analyses of the spectrum of postulated LOCA events:

- ANP-3640P, "Palo Verde Units 1, 2, and 3 Small Break LOCA Summary Report" (Reference 66), which addresses the postulated range of SBLOCA scenarios according to prescriptive, conservative requirements in Appendix K to 10 CFR Part 50, and

- ANP-3639P (Reference 29), which addresses the postulated range of LBLOCA scenarios using a realistic analysis methodology with explicit accounting for uncertainties.

The NRC staff's review of the LOCA analyses for Palo Verde focused upon pertinent sections of the licensee's submittals (particularly licensing reports ANP-3640P and ANP-3639P), including relevant responses to RAIs submitted on October 4, 2019 (Reference 5), as supplemented on November 26, 2019 (Reference 6).

The NRC staff further conducted three regulatory audits of licensee calculations and other information supporting the submittals docketed by the licensee.

As discussed previously, the licensee's proposed amendment does not involve a simple transition from the fuel and analytical methods of one vendor to those of another; rather the proposed amendment effectively requests permission to adopt new safety analysis methods for Framatome fuel, while maintaining in place existing methods applicable to Westinghouse fuel. Therefore, the proposed amendment would allow the licensee a continuing capability to operate the three units at Palo Verde with cores composed of applicable fuel assemblies manufactured by both Framatome and Westinghouse.[3]

Since analysis of the LOCA event generally involves proprietary information and models, each vendor's proprietary analysis, methods may typically only be applied to its own fuel designs. To maintain the capability to load fuel manufactured by both Westinghouse and Framatome, the licensee intends to maintain LOCA evaluation models and analytical calculations from both fuel

[3] Note that applicable fuel designs are those for which current safety analysis methods and the proposed methods reviewed in this SE are applicable, and which have received all required regulatory approvals.

vendors following approval of the proposed amendment request. Hence, in addition to reviewing the licensee's implementation of Framatome's LOCA evaluation models, the NRC staff requested additional information concerning, and focused its third regulatory audit upon, updates to the Westinghouse LOCA analyses for NGF and STD fuel assemblies to account for certain potential mixed-core configurations with co-resident Framatome HTP™ fuel.

3.5.1 Acceptability of LOCA Evaluation Models

As discussed above in Section 3.5, the licensee's demonstration that Palo Verde complies with applicable regulatory requirements for the spectrum of postulated LOCA events depends upon proprietary LOCA evaluation models developed by both Framatome and Westinghouse. Because the NRC staff has previously reviewed the implementation of the applicable Westinghouse LOCA evaluation models at Palo Verde,[4] this section of the SE focuses upon the Framatome LOCA evaluation models the licensee is currently proposing to implement.

To support the proposed loading of Framatome HTP™ fuel, the licensee is proposing to implement Framatome-developed evaluation models to demonstrate compliance with the four acceptance criteria from 10 CFR 50.46 that apply to the short-term LOCA analysis, including peak cladding temperature, maximum (local) cladding oxidation, maximum (core-wide) hydrogen generation, and a coolable core geometry. In accordance with 10 CFR 50.46, the licensee analyzed the spectrum of postulated LOCA events encompassing both small- and large-break LOCA event scenarios to verify satisfaction of applicable regulatory requirements.

The Framatome small- and large-break LOCA evaluation models are S-RELAP5-based methodologies that incorporate a kernel of transient fuel rod thermal-mechanical subroutines from the RODEX2 (small-break evaluation model) or COPERNIC (large-break evaluation model) codes.

The NRC staff's generic approval of Framatome's small- and large-break LOCA evaluation models is documented in previous NRC staff SEs, as shown below in Table 3.

Table 3: Previous NRC Staff Reviews of Framatome LOCA Evaluation Models

TR	Description	Date of NRC SE	Reference
EMF-2328P, Revision 0	PWR SBLOCA evaluation model using S-RELAP5	3/15/01	30
EMF-2328P, Supplement 1	PWR SBLOCA evaluation model using S-RELAP5	12/13/16	67
EMF-2103P, Revision 3	Realistic LBLOCA methodology for PWRs using S-RELAP5	6/17/16	28

While the generic LOCA evaluation models proposed by APS to support loading of Framatome HTP™ fuel have been previously found acceptable, the NRC staff reviews licensees' implementation of analytical evaluation models to ensure:

- confirmation of acceptable plant-specific inputs to the evaluation model,

[4] Most recently, in connection with the introduction of Westinghouse NGF fuel.

- confirmation of adherence to the approved evaluation model,
- confirmation that results calculated using the evaluation model satisfy regulatory acceptance criteria and otherwise conform to expectations, and
- verification of acceptable responses to limitations and conditions specified in the NRC staff's approving SEs.

Subsequent sections of this SE describe the NRC staff's review of these areas.

3.5.2 Evaluation Model Implementation

3.5.2.1 Framatome SBLOCA Evaluation Model

As noted above, the Framatome SBLOCA evaluation model described in EMF-2328P is a conservative, prescriptive methodology that conforms to the requirements of Appendix K to 10 CFR Part 50.

The licensee's implementation of the EMF-2328P evaluation model at Palo Verde is described in licensing report ANP-3640P, Revision 0 (Reference 66). In summary, the Framatome SBLOCA analysis for Palo Verde covers a spectrum of cold-leg breaks ranging from 1 to 9.49 inches in diameter.[5] The licensee also performed supporting analyses, including sensitivity analyses concerning the timing of the tripping of the RCPs (for both cold- and hot-leg breaks) and the safety injection water temperature, analysis of breaks on attached safety injection piping, and analysis of two additional plant-specific LOCA scenarios (i.e., inadvertent opening of pressurizer safety/relief valves and rupture of a reactor vessel instrument tube, both of which are discussed in Section 6.3 of the Palo Verde UFSAR). The licensee performed the SBLOCA analysis assuming the single-failure of one emergency diesel generator, which resulted in the inoperability of one high-pressure safety injection pump, one low-pressure safety injection pump, and one motor-driven auxiliary feedwater pump. **[[**

]] Additional description of the analysis and conservative analytical inputs chosen by the licensee are included in Section 3.0 of licensing report ANP-3640P.

The NRC staff's review of licensing report ANP-3640P found that the licensee's implementation of the Framatome SBLOCA evaluation for Palo Verde generally conforms to the approved evaluation model described in TR EMF-2328P. The NRC staff's regulatory audits corroborate this conclusion.

Information obtained in response to several RAIs by letter dated October 4, 2019, was necessary to support the NRC staff's conclusion, as summarized below.

In SNPB RAI-13 and RAI-14, the NRC staff requested that the licensee justify that its analysis models RCP behavior with adequate conservatism across the entire spectrum of SBLOCA events. In particular, the licensee had analyzed two possible scenarios: (1) RCPs trip concurrently with the reactor and (2) RCPs are manually tripped 5 minutes following a

[5] Consideration of ruptures up to 9.49 inches in diameter covers break areas up to 10 percent of the 30-inch cold-leg piping cross-sectional area, in accordance with the approved methodology in EMF-2328P.

pressurizer pressure decrease below the value necessary to ensure adequate pump net positive suction head (NPSH).[6] As described in Section 7 in Attachment 8 (i.e., technical analysis document) to the LAR dated July 6, 2018, Palo Verde has an existing operator action to trip the RCPs in its standard post-trip actions procedure; however, the assumption of a 5-minute time period for operators to complete this action has not previously been credited in safety analyses. The licensee's technical analysis document further indicates that, in simulator testing of five operating crews, the time required for operators to trip the RCPs was well under 1 minute.

While the two RCP trip scenarios analyzed by the licensee appear appropriate to address break sizes at the lower end of the SBLOCA spectrum, the NRC staff questioned their sufficiency for addressing the upper end of the break spectrum (e.g., greater than or equal to (≥) 6-inch break diameter), which the Framatome evaluation model predicts to be limiting for Palo Verde. For instance, at the limiting 9.1-inch break size, the time of peak cladding temperature was 186 seconds. In such a scenario, a 5-minute delay in tripping the RCPs would be tantamount to running the pumps throughout the event, which has long been recognized as a non-limiting condition. Furthermore, considering the simulator testing described above, the NRC staff expects that operators would most likely trip the RCPs with a delay substantially less than 5 minutes. Therefore, in SNPB RAI-13 and RAI-14, the NRC staff questioned whether the licensee's SBLOCA analysis had neglected RCP operating conditions potentially both more realistic and more limiting than those explicitly analyzed.

In response to SNPB RAI-13 and RAI-14, the licensee provided results for (1) relevant RCP trip sensitivity studies that it had previously performed in support of its proposed amendment request and (2) a new sensitivity study that considered a reduced delay period of 30 seconds prior to tripping the RCPs. The licensee considered a delay period of 30 seconds to be realistic based upon operator responses during simulator testing. As shown below in Section 3.5.3.1 of this SE, the results of both the original and new sensitivity studies proved less limiting than the baseline calculations that modeled tripping of the RCPs concurrently with the reactor trip.

The NRC staff's review found the licensee's responses to SNPB RAI-13 and RAI-14 acceptable because the licensee analyzed the range of potentially limiting conditions with respect to RCP behavior during the SBLOCA event and submitted the most limiting results for the purpose of demonstrating compliance with the acceptance criteria in 10 CFR 50.46(b). Considering the large size of the range of limiting breaks for Palo Verde relative to typical PWRs, which results in an accelerated event progression, the licensee's observation in response to SNPB RAI-14 that even short periods of RCP operation (e.g., 30-60 seconds) may have a beneficial effect for the predicted limiting scenario appears reasonable for the analyzed design configuration at Palo Verde.

In SNPB RAI-15, the NRC staff requested that the licensee justify **[[**

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[6] The licensee's LAR dated July 6, 2018, states that a pressurizer pressure of 1471 pounds per square inch at atmosphere (psia) ensures adequate net positive suction head for the RCPs.

In response to SNPB RAI-15 by letter dated October 4, 2019, the licensee stated that [[

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The NRC staff's review found that the licensee's response acceptable [[

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In SNPB RAI-16, the NRC staff requested that the licensee address whether the switchover to containment sump recirculation could occur prior to the quenching of the reactor core for any scenarios in the postulated range of small breaks. As noticed in SNPB RAI-16, emergency coolant supplied from the recirculation sump would generally have a higher temperature than coolant supplied from the refueling water tank, which may lead to the prediction of higher values for the peak cladding temperature and other figures of merit relevant to the LOCA event.

In response to SNPB RAI-16 by letter dated October 4, 2019, the licensee stated that, for the range of break sizes considered in licensing report ANP-3640P, core quench will occur prior to the switchover to containment sump recirculation. The licensee supported this conclusion with a refueling water tank-depletion calculation, which determined that significant margin to tank depletion would remain for the single-failure scenario considered in the baseline analysis (i.e., loss of one emergency diesel generator) that leaves only one of the two installed trains of safety injection and containment spray available.

The NRC staff's review of the licensee's response to SNPB RAI-16 considered whether scenarios involving two available containment spray pumps could deplete the refueling water tank more rapidly than the conditions assumed by the licensee. In particular, a failure of one train of high-pressure safety injection would provide a comparable quantity of safety injection for the small breaks of interest to SNPB RAI-16, while two trains of available containment spray would deplete the refueling water tank at almost double the rate considered by the licensee. The NRC staff's review of the licensee's response did not find sufficient information to conclude that switchover to containment sump recirculation would occur after core quench for all postulated break sizes and event scenarios within the Palo Verde design basis. However, continued review identified additional relevant factors:

- Considering the conservatisms in the licensee's analysis (e.g., assumption of runout spray flows, assumption of spray actuation at time zero), it appears likely that, even with two containment spray trains in operation, the small-break scenario with the longest quench time may quench prior to depletion of the refueling water tank.

- Plant emergency operating procedures contain instructions to secure one of two operating containment spray pumps provided that expected responses consistent with a design-basis SBLOCA event are observed for containment pressure and safety injection flow.

- Containment sump fluid temperatures for SBLOCA scenarios involving two trains of operating containment spray pumps would be reduced relative to the design-basis maximum sump temperature.

- The licensee's analysis shows significant margin to the acceptance criteria in 10 CFR 50.46(b) for the lower range of small breaks that would have the least margin to depletion of the refueling water tank (see Figure 7, below). Although switchover to hotter sump fluid prior to core quench could result in increases to the predicted figures of merit for breaks at the lower end of the size spectrum, the impact is expected to be within the available margin for these scenarios.

Therefore, considering the licensee's response to SNPB RAI-16 and these additional factors, the NRC staff concluded that the potential for switchover to containment sump recirculation prior to core quench has been acceptably addressed for the current Framatome SBLOCA analysis for Palo Verde.

In SNPB RAI-19, the NRC staff requested that the licensee justify a modification Framatome made to the evaluation model used for the Palo Verde SBLOCA analysis that deviates from the approved evaluation model described in EMF-2328P. **[[**

]]. In addition to requesting a description of the issue and its correction, the NRC staff further requested that the licensee estimate the magnitude of the effect and provide justification that the proposed modification would not adversely affect the basis for the validation and qualification of the approved evaluation model.

In its October 4, 2019, response to SNPB RAI-19, the licensee described the identification and correction **[[**

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In its supplemental response to SNPB RAI-19, dated November 26, 2019, the licensee provided additional information concerning the impact of Framatome's modification on the validation and qualification of its approved SBLOCA evaluation model. The supplemental response reviewed the experiments used to validate the approved evaluation model described in EMF-2328 (i.e., both Revision 0 and Supplement 1), assessing any potential impacts from the vendor's modification to the approved evaluation model. **[[**

]] issue does not affect the validation of the approved evaluation model.

The NRC staff's review concluded that adequate evidence was presented by the licensee's responses to SNPB RAI-19 to demonstrate that the continued applicability of the validation of Framatome's SBLOCA evaluation model was not adversely impacted by the modification Framatome implemented **[[**

]].

3.5.2.2 Framatome LBLOCA Evaluation Model

The Framatome LBLOCA evaluation model described in TR EMF-2103P (Reference 27) is a realistic methodology that quantifies and explicitly accounts for relevant uncertainties using statistical upper tolerance limits intended to ensure 95 percent probability coverage at a 95 percent confidence level (i.e., 95/95). The calculational domain of the approved evaluation model includes a range of potential conditions based upon sampling relevant plant parameters, phenomenological uncertainties, and event assumptions (e.g., break size).

The licensee's implementation of the TR EMF-2103P evaluation model at Palo Verde is described in licensing report ANP-3639P, Revision 0 (Reference 29). In summary, the licensee selected a **[[**

]]. The Palo Verde analysis considered the limiting single failure for CE plants identified in TR EMF-2103P (i.e., loss of one train of ECCS), along with conservative assumptions to maximize containment heat removal. In accordance with the approved methodology in Revision 3 to TR EMF-2103P, modeling of fuel swelling, rupture, and relocation is activated.

The NRC staff's review of licensing report ANP-3639P found that the licensee's implementation of the Framatome LBLOCA evaluation for Palo Verde generally conforms to the approved evaluation model described in TR EMF-2103P, Revision 3. The NRC staff's regulatory audits corroborate this conclusion.

Information obtained in response to one RAI was necessary to support the NRC staff's conclusion, as summarized below.

In SNPB RAI-20, the NRC staff requested that the licensee clarify and justify the modeling of non-Framatome fuel in the Framatome LBLOCA evaluation model. While licensing report ANP-3639P discusses hydraulic modeling of non-Framatome fuel, it does not specifically address other aspects of how non-Framatome assemblies are modeled, particularly fuel cladding-pellet thermal-mechanical modeling, both as a function of burnup during the operating cycle (i.e., to support initialization of the LOCA calculation), and during the transient LOCA event. As noted in SNPB RAI-20, Limitation 4.3, in the NRC staff's SE for TR EMF-2103P states that the Framatome LBLOCA evaluation model is applicable to fuel with M5® cladding, and has not been reviewed for application to fuels with other cladding materials (e.g., Westinghouse STD fuel and NGF).

In response to SNPB RAI-20 by letter dated October 4, 2019, the licensee stated that, with respect to the calculation of fuel rod performance, the Framatome LBLOCA calculation for Palo Verde only models Framatome fuel with M5® cladding. Similarly, the licensee stated that the models and properties for fuel rods and cladding that are embedded in the S-RELAP5 code used to perform the analysis in licensing report ANP-3639P are specific to Framatome fuel designs. The licensee stated that the Framatome analysis only accounts for hydraulic impacts of other vendors' fuel assembly designs on the distribution of flow through the reactor core. Therefore, the licensee stated that the reported figures of merit for peak cladding temperature and maximum local oxidation from the Framatome LOCA analysis are applicable only to Framatome fuel (under both full-core and applicable transition reload scenarios).

The NRC staff's review found the licensee's response acceptable because its limited application of the proprietary Framatome LOCA analysis to fuel manufactured by Framatome. Although Framatome's LBLOCA evaluation model would account for the hydraulic impacts of other vendors' fuels in applicable fuel transition reload scenarios, potentially limiting fuel assemblies manufactured by other vendors would need to be evaluated separately.

3.5.2.3 Analysis of Mixed-Core Configurations

Considering the unique nature of the licensee's proposal, which would permit long-term operation at Palo Verde with cores composed of a mixture of certain types of Framatome and Westinghouse fuels, the NRC staff's review paid particular attention to the treatment of mixed-core configurations. The NRC staff's review of the treatment of mixed-core configurations considered the relevant LOCA evaluation models developed by both Framatome and Westinghouse.

In SNPB RAI-17, the NRC staff requested that the licensee address whether each vendor's LOCA analyses consider a bounding core configuration that would address the impacts of potential variations in core composition on the predicted LOCA figures of merit. Furthermore, should existing mixed-core analyses not bound all potential possibilities, the NRC staff requested that the licensee clarify and justify the conditions under which each analysis would need to be reperformed.

In response SNPB RAI-17 by letter dated October 4, 2019, the licensee first described its planned strategy for performing core reloads. The licensee stated its intention to continue performing full core transitions from one type of fuel to another, in the manner that has traditionally characterized the domestic nuclear industry. The licensee stated that existing fuel management strategies for Palo Verde typically result in the core loading for a given fuel cycle involving approximately 100 fresh fuel assemblies, 100 once-burnt assemblies, and

41 twice-burnt assemblies. The licensee stated that the small- and large-break LOCA analyses performed in support of the proposed amendment request are based upon the existing practice of ordering all reload fuel assemblies for a given fuel cycle from a single vendor. The licensee further stated that the effects of mixed cores are accounted for, whether multiple fuel vendors are involved in a fuel transition. As discussed above relative to SNPB RAI-20, while fuel transitions at Palo Verde involving multiple fuel vendors would generally be expected to involve LOCA analyses from multiple vendors, it may be possible to address fuel transitions involving a single vendor with a more limited set of analyses. In either case, bounding core configurations would be assumed with the intention of conservatively accounting for the hydraulic and mechanical effects of different fuel assemblies.

The licensee further stated that, as part of its normal reload process for Palo Verde, each vendor's LOCA analyses would be evaluated to verify applicability. Considering that certain analytical inputs have been biased conservatively to provide an allowance for minor changes in plant configuration, the licensee stated that it is unlikely that anticipated transition core designs would fall outside the bounds of the relevant analytical input assumptions.

The NRC staff's review found the licensee's response acceptable based upon the proposed reload strategy that involves continuing to perform core reloads in accordance with existing practices. As discussed in the licensee's supplementary submittal dated November 26, 2019, the licensee proposed a license condition that will forbid the introduction of mixed fresh batches of reload fuel assemblies. This license condition will constitute a binding requirement that the licensee adhere to the intended full-batch reload strategy discussed in response to SNPB RAI-17.

3.5.3 Calculated Results

3.5.3.1 Framatome SBLOCA Evaluation Model

The results of the Palo Verde SBLOCA analysis are discussed in Section 4.0 of licensing report ANP-3640P. Key analytical results are summarized below in Figure 7 and Table 4.

Figure 7: Peak Cladding Temperature as a Function of Break Diameter for the Palo Verde Small-Break LOCA Analysis

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Table 4: Predicted Figures of Merit for the Palo Verde Small-Break LOCA Analysis

Figure of Merit	Predicted Value	Acceptance Criterion
Peak Cladding Temperature	1620 °F	≤ 2200 °F
Maximum (Local) Cladding Oxidation	2.96%	≤ 17%
Maximum (Core-Wide) Hydrogen Generation	0.006%	≤ 1%

As shown above in Figure 7, the Framatome SBLOCA evaluation model predicted limiting results for Palo Verde in a range of approximately 8.8 to 9.1 inches in break diameter. All of the limiting figures of merit shown in Table 4 are associated with breaks in this size range. The predicted value of maximum local cladding oxidation shown in Table 4 [[

]]. Further description of the results, including output parameter plots for the limiting case, may be found in Section 4.0 of licensing report ANP-3640P.

The Framatome SBLOCA analysis for Palo Verde predicts that tripping the RCPs contemporaneously with the reactor is more limiting than assuming a delayed trip 5 minutes after reaching the loss of NPSH margin trip criterion.[7] However, as discussed above in

[7] Note that the licensee considered both cold- and hot-leg breaks in the delayed RCP trip sensitivity studies discussed in ANP-3640P.

Section 3.5.2.1, in SNPB RAI-13 and RAI-14, the NRC staff questioned whether the licensee had adequately considered other RCP trip scenarios potentially both more realistic and more limiting than the two analyzed cases. In response, the licensee performed an additional sensitivity study with a 30-second trip delay time following satisfaction of the loss of NPSH pump trip criterion.[8] Results for all three RCP trip scenarios analyzed by the licensee are illustrated below in Figure 8.

Figure 8: Reactor Coolant Pump Trip Delay Sensitivity Results for Palo Verde Small-Break LOCA Analysis

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The results of these sensitivity studies demonstrate the importance of RCP behavior in SBLOCA analyses. For example:

- Consistently with the NRC staff's expectation, predicted results for larger SBLOCA scenarios in the potentially limiting range are seen to be significantly non-limiting with an assumed 5-minute delayed RCP trip.

- Comparing predicted results for the 30-second RCP trip delay case to the baseline case (i.e., trip RCPs with the reactor), **[[**

[8] As discussed above in Section 3.5.2.1, the 30-second trip delay is based on operator response times during simulator exercises.

]].

Despite these evident sensitivities, the analytical results calculated by the licensee show no indication that reasonably expected perturbations to the RCP trip time would result in the prediction of figures of merit for Palo Verde that are more limiting than those presented in Table 4 above.[9]

The NRC staff further observed that the peak cladding temperature of 1620 °F predicted by the Framatome SBLOCA evaluation model following introduction of Framatome HTP™ fuel is nearly identical to the value of 1618 °F computed by the Westinghouse/Combustion Engineering evaluation model for current Palo Verde core designs composed of Westinghouse NGF.

As shown above in Table 4, the figures of merit predicted by the Framatome SBLOCA evaluation model for Palo Verde comply with the acceptance criteria in 10 CFR 50.46. The NRC staff's review further found that the analytical results presented in licensing report ANP-3640P demonstrate reasonably expected behavior and are comparable to previous calculations. The NRC staff's regulatory audit of calculations supporting the small-break LOCA analysis for Palo Verde corroborates this finding.

Information obtained in response to one RAI was necessary to support the NRC staff's conclusion, as summarized below.

In SNPB RAI-18, the NRC staff requested that the licensee address several instances where plots of calculated output from the Framatome SBLOCA evaluation model displayed potentially non-physical behavior. In particular, the NRC staff observed (1) unexpected variation in predicted steam generator fluid masses with no apparent change in reported inlet or outlet mass flows, and (2) unexpected, prolonged periods (e.g., spanning 250-300 seconds), during which the hot assembly mixture level was predicted to remain essentially constant at several distinct values, all of which were below the top of active fuel.

In response to SNPB RAI-18 by letter dated October 4, 2019, the licensee indicated that the variations in steam generator mass reflect fluid redistribution between the steam generators as pressure equilibrium is maintained in response to asymmetric changes in heat transfer from the connected RCS loops. Regarding hot assembly mixture level, [[

]]. The licensee's response included a plot of the calculated void fraction at the core exit and in the upper plenum, which showed no evidence of the unexpected behavior observed in the original plot of hot assembly mixture level.

The NRC staff's review found the licensee's response to SNPB RAI-18 acceptable because it provided adequate evidence that two anomalies identified during the staff's review of ANP-3640P are not indicative of non-physical evaluation model predictions.

[9] Note that this conclusion may be associated with the particularly large limiting break size predicted for Palo Verde.

3.5.3.2 Framatome LBLOCA Evaluation Model

The results of the Palo Verde LBLOCA analysis are discussed in Section 3.0 of ANP-3639P (Reference 29). Key analytical results are summarized below in Figure 9 (reproduced from ANP-3639P) and Table 5.

Figure 9: Peak Cladding Temperature Versus Break Size for the Palo Verde Large-Break LOCA Analysis

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Table 5: Predicted Figures of Merit for the Palo Verde Large-Break LOCA Analysis

Figure of Merit	Predicted Value	Acceptance Criterion
Peak Cladding Temperature	1752 °F	≤ 2200 °F
Maximum (Local) Cladding Oxidation	2.37%	≤ 13%[10]
Maximum (Core-Wide) Hydrogen Generation	0.02%	≤ 1%

The limiting results shown in Table 5 represent a simultaneous upper tolerance limit at the 95 percent coverage and 95 percent confidence level. In accordance with the methodology approved in EMF-2103, Revision 3, [[

]].[11]

Section 3.0 of licensing report ANP-3639P contains considerable information concerning the statistical LBLOCA analysis for Palo Verde, including a tabulation of key results and sampled parameters for each of the [[

]] and a summary of the range of parameters over which fuel rod rupture was predicted to occur.

In SNPB RAI-12, the NRC staff requested that the licensee address certain outlying datapoints indicated on two plots from licensing report ANP-3639P that are reproduced below in Figure 10.[12]

[10] As discussed further in the NRC staff's SE for TR EMF-2103P, Revision 3, the limit of 17 percent local cladding oxidation specified in 10 CFR 50.46(b)(2) presumes use of the Baker-Just correlation; 13 percent is the equivalent acceptance criterion for the Cathcart-Pawel correlation used in TR EMF-2103P.

[11] [[

]]

[12] The NRC staff has added red circles to the original plots from licensing report ANP-3639P to highlight outlying datapoints.

Figure 10: Outlying Datapoints in Predicted Results
for the Palo Verde Large-Break LOCA Analysis

[[

]]

In particular, the NRC staff noted that

- the case that sets the 95/95 limit for peak cladding temperature [[ ]],[13] and
- the maximum local oxidation results showed [[ ]].

In response to SNPB RAI-12 by letter dated October 4, 2019, the licensee stated the following:

- The case involving [[

[13] Careful comparison of Figure 10 and Table 5 shows that the case involving [[ ]]

]]. In addition, the licensee observed that steam binding significantly retarded reflooding of the core, thereby extending the heatup duration.

- Regarding the cases with outlying predictions of maximum local oxidation, the licensee's response focused on several fundamental aspects of the metal-water reaction rate, including its exponential dependence upon cladding temperature, as well as its dependence upon exposure time.

In both instances, the licensee's response compared the outlying cases with other cases for which more centrally distributed results were predicted. The licensee's response concluded that a reasonable physical basis exists for the observed predictions of the Framatome large-break LOCA evaluation model that the NRC staff questioned in SNPB RAI-12.

The NRC staff's review of the licensee's response to SNPB RAI-12 **[[**

]] However, the NRC staff recognizes that the LOCA event involves significant non-linear physical phenomena, and the appearance of outlying predictions is, in itself, not unexpected or problematic. The NRC staff's review, as supported by audits of underlying vendor calculations, found no evidence of non-physical behavior or deviation from the approved Framatome LBLOCA evaluation model, which permits sampling of parameters associated with the event definition (e.g., break area) and initial conditions (e.g., LHGR) that may promote variation in the calculated figures of merit. Therefore, the NRC staff found that the licensee's response to SNPB RAI-12 is acceptable.

The NRC staff observed that the peak cladding temperature of 1752 °F predicted by Framatome's realistic LBLOCA evaluation model following introduction of Framatome HTP™ fuel is significantly lower than the corresponding value of 2106 °F computed by the Westinghouse/Combustion Engineering Appendix K evaluation model for current Palo Verde core designs with Westinghouse NGF (Reference 71). A difference in peak cladding temperature of this magnitude is not unexpected, considering conservatisms associated with traditional Appendix-K-conformant evaluation models relative to best-estimate-plus-uncertainty modeling approaches.

As reflected above in Table 5, the results predicted by the Framatome LBLOCA evaluation for Palo Verde comply with the acceptance criteria in 10 CFR 50.46. The NRC staff's review further found that the LBLOCA results for Palo Verde presented in ANP-3639P (1) demonstrate reasonably expected behavior and (2) display reasonably expected differences as compared to the existing LBLOCA analysis. The NRC staff's regulatory audit of calculations supporting the LBLOCA analysis for Palo Verde corroborate these findings.

3.5.4 Conformance to Applicable Limitations and Conditions

3.5.4.1 Framatome SBLOCA Evaluation Model

TR EMF-2328, Revision 0 contains one condition requiring additional validation of the evaluation model prior to its application to breaks larger than 10 percent of the cold leg cross sectional area. The NRC staff considers this condition addressed because the evaluation model has not been generically approved for breaks above this size, and approval for

application to this break size range has not been requested by Palo Verde in its proposed license amendment.

As described in the associated NRC staff SE, the staff's approval of Supplement 1 to TR EMF-2328, Revision 0, imposed modeling requirements in the following areas:

1. Spectrum of Break Sizes
2. Core Bypass Flowpaths in the Reactor Vessel
3. Reactivity Feedback
4. Delayed RCP Trip
5. Maximum Safety Injection Tank/Refueling Water Tank Fluid Temperature
6. Loop Seal Clearing and Crossover Leg Modeling
7. Breaks in Attached Piping
8. Core Nodalization

In licensing report ANP-3640P, the licensee stated that the Palo Verde SBLOCA analysis was performed in accordance with the NRC-staff-approved S-RELAP5 methodology described in TR EMF-2328(P)(A), Revision 0, as modified in accordance with TR EMF-2328(P)(A), Revision 0, Supplement 1. Because the eight modeling requirements listed above are included in TR EMF-2328(P)(A), Revision 0, Supplement 1, there is reasonable assurance that the licensee conforms to these modeling requirements. The NRC staff's audit of supporting calculations corroborates this conclusion.

3.5.4.2 Framatome LBLOCA Evaluation Model

In Table 3-3, "SE Limitations Evaluation," from licensing report ANP-3639P, the licensee addressed each of the 11 limitations and conditions from the NRC staff's SE for TR EMF-2103P, Revision 3. With only one exception that will be discussed in further detail below (i.e., Limitation 4.4), the licensee's discussion of these limitations and conditions directly affirmed that the analysis in licensing report ANP-3639P conforms to the regulatory positions imposed in the NRC staff's SE on TR EMF-2103, Revision 3.

Limitation 4.4 from the NRC staff's SE on TR EMF-2103, Revision 3, discusses conformance to the modeling guidelines contained in Appendix A of TR EMF-2103, Revision 3, and the provision of justification in cases where plant-specific deviations exist. **[[**

]]

The NRC staff reviewed these deviations from the modeling guidelines and found them reasonable. In this regard, the NRC staff notes that plant-specific variation in geometry is expected, and that Palo Verde in particular has known geometric differences relative to CE reactor designs of earlier vintage. The specification of certain practices as modeling guidelines is intended to provide flexibility for justifiable deviations where necessary to simulate unique plant-specific configurations accurately. Therefore, the NRC staff concludes that the licensee has acceptably addressed Limitation 4.4 from the NRC staff's SE for TR EMF-2103, Revision 3.

Based upon the review described above, the NRC staff finds that the licensee acceptably addressed applicable limitations and conditions associated with the Framatome LBLOCA evaluation model.

3.5.5 Westinghouse Small- and Large-Break LOCA Analyses

The NRC staff previously reviewed the Westinghouse LOCA analyses for Palo Verde in conjunction with a proposed license amendment to support loading Westinghouse NGF (Reference 68). In the present review, the NRC staff focused upon the impacts of introducing Framatome HTP™ fuel on the predicted figures of merit for Westinghouse fuel assemblies.

In SNPB RAI-28, the NRC staff requested that the licensee (1) describe the Westinghouse small- and large-break LOCA analyses for mixed-core conditions involving Framatome HTP™ fuel and (2) provide the results of the Westinghouse LOCA analysis and confirm that the acceptance criteria specified in 10 CFR 50.46(b) remain satisfied.

In response to this RAI and SNPB RAI-17 by letter dated October 4, 2019, the licensee described the following four transition core scenarios and analyzed them using the Westinghouse small- and large-break LOCA evaluation models:

- a transition from a full core of Westinghouse STD fuel to a full core of Framatome HTP™ fuel (except for the hot assembly),

- a transition from a full core of Westinghouse NGF to a full core of Framatome HTP™ fuel (except for the hot assembly),

- a transition from a mixed-core of Westinghouse NGF and STD fuel to a full core of Framatome HTP™ (except for the hot assembly), and

- a transition from a full core of Framatome HTP™ (except for the hot assembly) to a full core of Westinghouse NGF.

The licensee performed explicit, quantitative analyses of these transition scenarios with the approved Westinghouse LBLOCA methodology currently in use at Palo Verde (i.e., the 1999 EM, References 72, 73, 74, 75, and 76). The analyses performed by the licensee found that the calculated figures of merit were not significantly affected by transition core phenomena and that the current AORs remain bounding.

Because the approved Westinghouse SBLOCA methodology currently in use at Palo Verde (i.e., the S2M, References 77, 78, and 79) is a **[[ ]]**, the licensee dispositioned the impact of transition core effects using qualitative insights, concluding that the current AORs likewise remain bounding.

The NRC staff's review of the licensee's response to SNPB RAI-28 found that the licensee has reasonably accounted for mixed-core effects associated with the introduction of Framatome HTP™ fuel on the Westinghouse small- and large-break LOCA analyses. Explicit analysis using the current Westinghouse LBLOCA evaluation model for Palo Verde provides confidence that mixed-core impacts will not adversely affect the LBLOCA AOR. Considering that the physical behavior during a SBLOCA is governed largely by gravitational forces, in contrast to the

inertially dominated LBLOCA event, the NRC staff considers it reasonable to expect that similar conclusions hold for the SBLOCA analysis. The licensee's analytical results further appear reasonable considering the relative similarity of the fuel assembly designs involved in the analyzed mixed-core configurations and specific fuel assembly design data included in the licensee's submittal.[14] The NRC staff's audit of the Westinghouse LOCA calculations described in the responses to SNPB RAI-17 and -28 further supports this conclusion.

3.5.6 Long-Term Core Cooling

Section 7 in Attachment 8 (i.e., the technical analysis document) of the LAR dated July 6, 2018 (Reference 1), contains the licensee's assessment of long-term core cooling for Palo Verde, which is intended to demonstrate compliance with 10 CFR 50.46(b)(5). The licensee's long-term core cooling assessment covers three main areas:

- boric acid precipitation in the reactor vessel,
- long-term decay heat removal strategy, and
- post-LOCA debris blockage in the reactor vessel.

3.5.6.1 Boric Acid Precipitation and Decay Heat Removal

Regarding the first two items, boric acid precipitation and the long-term decay heat removal strategy, the licensee determined that the previous analyses performed for Westinghouse NGF continue to apply following introduction of Framatome HTP™ fuel.

With respect to boric acid precipitation, the licensee stated that the core fluid volume with Framatome HTP™ fuel will be equivalent to that with Westinghouse STD fuel. As such, the licensee concluded that the available boric acid mixing volume, and hence the calculated precipitation times, would also be equivalent. In Table 8-13 in Attachment 8 to the LAR, to implement Westinghouse NGF (Reference 80), the licensee listed significant parameters associated with its boric acid precipitation analysis. As confirmed in the licensee's response to SNPB RAI-27, these parameters either remain valid or will be verified to be valid on a cycle-specific basis (e.g., maximum boron concentration).

With respect to long-term decay heat removal, the licensee stated that the important input parameters in this analysis remain applicable following the introduction of Framatome HTP™ fuel. The licensee listed the important input parameters in Table 8-15 in Attachment 8 to its request to implement Westinghouse NGF (Reference 80). The NRC staff noticed that Table 8-15 includes parameters associated with reactor power and decay heat, RCS temperature and cooldown rate, steam generator heat transfer, stored condensate, and the shutdown cooling system. The parameters the licensee identified as important to the long-term decay heat analysis are not specifically associated with fuel assembly design.

Based on the evidence described above, the NRC staff agrees with the licensee's determination that the existing analyses for boric acid precipitation and long-term decay heat removal continue to apply following the introduction of Framatome HTP™ fuel. The NRC staff's evaluation of the licensee's existing analyses of boric acid precipitation and long-term decay heat removal are

[14] For instance, the licensee's submittal identifies that the flow resistance of the Framatome HTP™ fuel is between that of Westinghouse STD fuel and NGF; the mixed-core impacts of these two Westinghouse fuel designs have already been accounted for in the current analyses of record (Reference 68).

described in the SE for the implementation of Westinghouse NGF at Palo Verde (Reference 68). While the licensee's assessment that these analyses apply to Framatome HTP™ fuel is somewhat simplified, the NRC staff's finds it physically reasonable that the boric acid precipitation and decay heat removal analyses would be primarily influenced by system design attributes and would not be significantly affected by modest changes in fuel design. Although other factors not included in Tables 8-13 and 8-15 may influence vessel water levels and available mixing volumes (e.g., fuel assembly flow resistance), these factors also are sufficiently similar between the Framatome HTP™ fuel and Westinghouse NGF and STD fuel, considering analytical conservatisms and margins. As noted above, the licensee's response to SNPB RAI-27 confirms that the key parameters in the analyses of boric acid precipitation and long-term decay heat removal continue to apply to Palo Verde.

3.5.6.2 Post-LOCA Debris Blockage in the Reactor Vessel

Regarding the potential for post-LOCA debris blockage in the reactor vessel, in Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on emergency Recirculation During Design Basis Accident at Pressurized-Water Reactors," dated September 13, 2004 (Reference 81), the NRC staff requested that PWR licensees address recognized impacts of post-LOCA debris on the ECCS in long-term cooling mode.[15] Although Palo Verde has implemented modifications, including a significantly increased ECCS recirculation strainer area, the licensee acknowledged that it has not fully resolved all concerns with post-LOCA debris identified in GL 2004-02. The licensee stated that it continues to work toward resolving concerns with post-LOCA debris in accordance with the policy outlined by the Commission in its Staff Requirements Memorandum associated with SECY-12-0093, "Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance" (Reference 82).

The NRC staff found that the licensee's approach to resolving concerns with post-LOCA debris blockage in the reactor vessel to be generally consistent with the policy established by the Commission. However, in SNPB RAI-11, the NRC staff requested further information to assure that introduction of Framatome HTP™ fuel would not significantly exacerbate the potential for post-LOCA debris blockage in the reactor vessel during the interim period prior to complete resolution of the issues identified in GL 2004-02.

In response to SNPB RAI-11 by letter dated October 4, 2019, the licensee discussed its rationale for concluding that the loading of Framatome HTP™ fuel would not pose additional concerns with post-LOCA debris blockage in the reactor vessel. The licensee's response acknowledged that the only downstream flow restriction with a clearance smaller than the maximum hole size of the Palo Verde ECCS recirculation strainers would be the fuel filters on the Framatome HTP™ assemblies that would be loaded following acceptance of the present amendment request. However, the licensee noted that the approved TR WCAP-16793 methodology for evaluating downstream effects in the reactor vessel derived an acceptable fiber limit of 15 grams per fuel assembly; in comparison, the licensee calculated the current Palo Verde fiber loading to be 13.8 grams per fuel assembly.

Based upon the information provided in response to SNPB RAI-11, the NRC staff's review found that the loading of Framatome HTP™ fuel would not be expected to exacerbate the potential for

[15] Although not directly relevant to the present review associated with the loading of Framatome HTP™ fuel and implementation of associated analysis methods, GL 2004-02 also encompasses post-LOCA debris impacts on containment heat removal.

post-LOCA debris blockage in the reactor vessel. The NRC staff's conclusion is based upon the limited quantity of fibrous debris expected at Palo Verde following a LOCA event, as well as conservatisms inherent in the TR WCAP-16793 methodology, some of which are discussed in the licensee's RAI response.

3.5.7 LOCA Analysis Conclusion

The NRC staff reviewed the information in the licensee's submittals pertaining to the analysis of the spectrum of postulated LOCA events for Palo Verde, including TRs ANP-3639P and ANP-3640P, as well as relevant responses to RAIs. The NRC staff's review was further supported by three regulatory audits, which were used to confirm information in docketed submittals from the licensee. Based upon its review, as documented above, the NRC staff has concluded that:

(1) the licensee has proposed to implement the Framatome small- and large-break LOCA evaluation models described in EMF-2328P and EMF-2103P in an acceptable manner,

(2) the licensee has acceptably accounted for the introduction of Framatome HTP™ fuel in its small- and large-break LOCA analyses for Westinghouse fuel, and

(3) compliance with the applicable regulatory requirements, namely 10 CFR 50.46, Appendix K to 10 CFR Part 50, and GDC 35, has been acceptably demonstrated.

3.6 Containment Response

Introduction of CE 16HTP™ at Palo Verde has a minimal impact on the containment response. APS completed mass and energy (M&E) evaluations for LBLOCA, main steam line break (MSLB), and containment sub-compartment line breaks. The changes introduced by the adoption of CE 16HTP™ do not result in a challenge to the current bounding AOR. Ultimately, these minor changes are insignificant with respect to nominal operating conditions and represent inconsequential changes to system operating and design limits. Based on the licensee's statement that these differences are indiscernible relative to the plant nominal operating point, the NRC staff agreed that the associated effect is inconsequential. Though these changes are minor, M&E release AORs have still been examined to ensure continued applicability at Palo Verde.

M&E Release for LOCA

There are three major phases examined for the M&E analysis during a LBLOCA: blowdown, reflood/post-reflood, and long term boiloff. An evaluation of HTP™'s impact on LOCA M&E release AORs was performed.

For the short term (blowdown phase) M&E release, there is not sufficient time for the reactor core and the primary and secondary sides to interact to significantly affect the M&E releases. Additionally, the parameters, which impact the M&E release analysis, have not changed (maximum RCS pressure and temperature) with the HTP™ transition.

Similarly, the reflood/post-reflood and long term boiloff evaluations also remain bounded by the AOR. The fuel performance parameters for CE 16HTP™ fuel are bounded by those assumed in the AOR. In response to SRXB RAI-7 by letter dated October 4, 2019, the licensee described the evaluated fuel parameter of interest to determine the impact in the reflood/post-reflood

phase. The evaluated parameters included: core average LHR, pellet and cladding geometry, centerline temperature, decay heat, and metal/water reaction. For the core average LHR, the AOR blowdown value is larger and results in more energy being transferred to the coolant, which is conservative for containment M&E releases. The pellet and cladding geometry, the pellet outside diameter, cladding inside diameter, and cladding outside diameter are the same between CE 16 STD and the Framatome CE 16HTP™ fuel and any small differences in the flow area will be insignificant since the flow area has already been included in the core pressure drop that was evaluated to have a negligible impact. The CE 16HTP™ fuel centerline temperature is slightly higher at 3014 °F than the AOR value of 3000 °F; however, the location of the maximum fuel centerline temperature is different between the CE 16HTP™ and CE 16x16. The maximum centerline temperatures appearing at a different axial node for the two analyses can be attributed to the COPERNIC adjustment of the axial power shape. Based on the different computer code use of axial power shape and the modeling and calculation differences between the two analyses, the difference of 14 °F is negligible. The AOR assumed an initial core power level of 102 percent and remains bounding for the CE 16HTP™ fuel decay heat. Finally, for the metal/water reaction, the M&E release analysis biases the reactor core parameters to extract as much energy from the fuel and components as possible in order to generate steam. This results in lower fuel clad temperatures such that the metal-water reaction is negligible.

The changes due to pressure drop, average LHGR, and geometries, etc., were considered and determined to be either bounded by the AOR or of negligible impact. Therefore, the NRC staff concludes that the LOCA M&E release AORs remain applicable.

M&E Release for MSLB Accidents

M&E releases for the MSLB accidents are calculated at 102 percent, 75 percent, and 0 percent power levels. The core stored energy is the parameter of interest when evaluating if a new fuel impacts the AOR for the MSLB, which calculates the containment response. For the transition to CE 16HTP™, the core stored energy was evaluated to be bounded by the AOR stored energy. The NRC staff determined that the AORs remain applicable because the evaluations were completed at equal or higher stored energy.

M&E Release for Containment Sub-compartment Line Breaks

The sub-compartment line break transients are of short duration and the M&E releases are not significantly affected. There is not sufficient time for the reactor core and the primary and secondary sides to interact. The analysis is impacted by the RCS pressure and temperature. In the transition, the RCS pressure and temperature have not changed. Therefore, the AOR continues to remain applicable.

The NRC staff concludes that the M&E AORs remain bounding or are not impacted by the transition to CE 16HTP™ fuel.

3.7 Proposed Technical Specification Changes

The licensee proposed revisions to several TSs to support loading Framatome HTP™ fuel:

- The licensee proposed to add a new fuel centerline melt SL for Framatome HTP™ fuel into SL 2.1.1.2. In its RAI responses by letters dated October 4 and November 26, 2019 (References 5 and 6), the licensee proposed a further revision to specify separate fuel

centerline melt SLs for Westinghouse fuel with erbium and Westinghouse fuel with zirconium diboride or no burnable poison.

- The licensee proposed to revise the description of fuel assemblies in the reactor core in TS 4.2.1 by deleting existent names of specific cladding materials (e.g., Zircaloy, ZIRLO), and replacing them with the generic term "zirconium-alloy clad." The licensee further proposed revisions to TS 4.2.1 intended to clarify requirements applicable to lead test assemblies.

- The licensee proposed to add analytical methodologies necessary to calculate core operating limits for Framatome HTP™ fuel into TS 5.6.5.b.

The NRC staff's review of the proposed changes, including revisions to the original submittal proposed by the licensee in response to RAIs from the NRC staff, is documented below.

3.7.1 Fuel Centerline Melt Safety Limits (SL 2.1.1.2)

Currently, SL 2.1.1.2 contains a single fuel centerline melt limit. The licensee proposed a revision to SL 2.1.1.2 that would create separate fuel centerline melt limits for Westinghouse- and Framatome-supplied fuel designs.

While agreeing with the intent of the proposed revision in SNPB RAI-21, the NRC staff questioned whether the wording proposed by the licensee (i.e., "Westinghouse supplied fuel" and "Framatome supplied fuel") could lend itself to overly broad and technically unsupportable interpretations. For instance, the specific centerline melt limits proposed by the licensee may not apply to all fuel designs supplied by each vendor.

The licensee's response to SNPB RAI-21 acknowledged the concern and, following discussions during the audit held on June 17-20, 2019, further proposed adding to SL 2.1.1.2 a third fuel centerline melt limit. The licensee found the addition of a third limit appropriate because the adjustment from CENPD-382-P-A for burnable poison specified in existing SL 2.1.1.2 applies strictly to the erbia burnable absorber used in Westinghouse STD fuel, and not the zirconium diboride burnable absorber typically used in Westinghouse NGF. Hence, the licensee's revised proposed version of SL 2.1.1.2 in the supplement to the LAR dated November 26, 2019, as reproduced below, contains three separate fuel centerline melt limits, each of which would be applied to the appropriate fuel design:

2.1.1.2 In MODES 1 and 2,

2.1.1.2.1 The peak fuel centerline temperature for Westinghouse supplied fuel using erbium as a burnable poison shall be maintained < 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-382-P-A).

2.1.1.2.2 The peak fuel centerline temperature for Westinghouse supplied fuel using zirconium-diboride as a burnable poison, or not using a

burnable poison integral to the fuel pellet, shall be maintained < 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup).

2.1.1.2.3 The peak fuel centerline temperature for Framatome supplied fuel using gadolinium as a burnable poison, or not using a burnable poison integral to the fuel pellet, shall be maintained < 4901°F (decreasing by 13.7°F per 10,000 MWD/MTU for burnup).

The licensee's response to SNPB RAI-21 further acknowledged that the proposed SLs are appropriate for application to the Westinghouse STD fuel and NGF designs in current use at Palo Verde, as well as the Framatome HTP™ fuel design that would be loaded, contingent upon acceptance of the proposed license amendment. Should additional fuel designs be licensed for use at Palo Verde in the future, any existent SLs must be evaluated at that time, in accordance with regulatory requirements, to determine their continued applicability. If, at such time, existent limits are found inadequate to protect the additional fuel design(s) from fuel centerline melt, acceptable limits must be devised for the new fuel design(s) to assure compliance with 10 CFR 50.36(c)(1).

The NRC staff finds that the licensee's response to SNPB RAI-21 is acceptable because (1) it would apply technically justifiable fuel centerline melt limits to each fuel type that is currently planned to be loaded at Palo Verde, (2) it explicitly acknowledges that the proposed fuel centerline melt limits are not necessarily applicable to all current or future fuel designs manufactured by Westinghouse or Framatome, and (3) it explicitly acknowledges that, prior to future adoption of new fuel designs, APS must either verify the continued acceptability of existent fuel centerline melt limits or propose new acceptable limits.

Likewise, the NRC staff finds the licensee's proposed revision to SL 2.1.1.2 acceptable because, in accordance with 10 CFR 50.36(c)(1), it specifies appropriate SLs to protect against fuel centerline melt for each type of fuel the licensee plans to load at Palo Verde following the adoption of the proposed license amendment.

3.7.2 Fuel Assemblies in the Reactor Core (TS 4.2.1)

In its supplemental RAI response dated November 26, 2019, the licensee proposed a revised version of TS 4.2.1 as follows:

The reactor shall contain 241 fuel assemblies.

a. Each assembly shall consist of a matrix of fuel rods with an NRC approved cladding material with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. Each unit-

specific COLR shall contain an identification of the fuel types and cladding material in the reactor, and the associated COLR methodologies.

b. A limited number of lead test assemblies not meeting 4.2.1.a may be placed in nonlimiting core regions. Each unit-specific COLR shall contain an identification of any lead test assemblies in the reactor.

As discussed below, the proposed revision to TS 4.2.1 would primarily affect two areas:

- allowable fuel rod cladding materials
- lead test assemblies

3.7.2.1 Allowable Fuel Rod Cladding Materials

Currently, TS 4.2.1 specifies that fuel assemblies used at Palo Verde shall consist of fuel rods clad with Zircaloy, ZIRLO, or Optimized ZIRLO. The licensee's proposed revision to TS 4.2.1, as described in its supplemental RAI response dated November 26, 2019, involves deletion and replacement of the names of these three specific cladding materials with a statement that fuel rods shall use "an NRC approved cladding material."[16]

In SNPB RAI-22, the NRC staff questioned whether the removal of specific cladding material designations from the TSs would be consistent with the requirement in 10 CFR 50.36(c)(4) that "[d]esign features to be included [in the Design Features section of the TSs] are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety" and are not covered under 10 CFR 50.36(c)(1)–(3).

Ultimately, the NRC staff's review of the proposed wording of TS 4.2.1.a in the licensee's supplemental RAI response focused upon the requirement for NRC approval of the fuel cladding materials used at Palo Verde. The requirement to use only cladding materials reviewed and approved by the NRC staff for the intended application would reasonably assure that a change from one type of approved cladding to another approved cladding would not trigger the criterion cited in 10 CFR 50.36(c)(4) (i.e., have a significant effect on safety).

Furthermore, as discussed further below, the licensee proposed a revision to TS 5.6.5 to require that the COLR submitted to the NRC staff on a cycle-specific basis identify the specific cladding materials in use at each unit.

The NRC staff finds that the licensee's proposed changes to TS 4.2.1 regarding the allowable fuel rod cladding materials is acceptable because the proposed changes would satisfy applicable regulatory requirements specified in 10 CFR 50.36(c)(4) by ensuring that only NRC-approved cladding materials may be used at Palo Verde. Furthermore, the NRC staff would remain apprised of the specific cladding materials in use at each unit at Palo Verde via the COLR submitted for each fuel cycle.

[16] Note that the licensee had originally proposed in its LAR dated July 6, 2018, to replace the specific cladding material names cited in TS 4.2.1 with the umbrella term "zirconium-alloy clad."

3.7.2.2 Lead Test Assemblies

As proposed in its supplemental RAI response dated November 26, 2019, the licensee's revision to TS 4.2.1 involves the following changes concerning lead test assemblies:

- Move the existing discussion of lead test assemblies into a new paragraph (i.e., TS 4.2.1.b) and add wording to emphasize that lead test assemblies need not satisfy the description in proposed TS 4.2.1.a. The licensee indicated that the new paragraph break and new wording would clarify that the description of fuel assemblies in proposed TS 4.2.1.a is intended for batch reload assemblies and does not apply to lead test assemblies.

- Delete existing sentence stating that "Other cladding material may be used with an approved exemption." The licensee justified deletion of this sentence by arguing that (1) it imposes no requirement beyond those already in effect per 10 CFR 50.46, and (2) its removal would be consistent with the language used in the Standard Technical Specifications for CE Plants in NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants" (Reference 83).

- Add a new sentence stating that "Each unit-specific COLR shall contain an identification of any lead test assemblies in the reactor." As discussed further below, the licensee proposed a revision to TS 5.6.5 to require that the COLR identify any lead test assemblies in use at each unit on a cycle-specific basis.

The proposed wording of TS 4.2.1.b in the supplemental RAI response dated November 26, 2019, incorporates changes relative to the originally proposed version in the LAR that were intended to address NRC staff concerns expressed in SNPB RAI-23 regarding the potential for inconsistent interpretations as to the applicability of text in the fuel assembly description (new TS 4.2.1.a) to lead test assemblies (new TS 4.2.1.b). The revised wording included in the licensee's supplemental RAI response addresses the NRC staff's concern by clearly delineating TS requirements for reload fuel assemblies and lead test assemblies.

The NRC staff also agrees with the proposed deletion of the existing sentence indicating that exemptions are required for lead test assemblies clad with materials not encompassed by TS 4.2.1. The NRC staff's regulatory position concerning lead test assemblies is expressed in letters to the industry dated June 24, 2019 (Reference 84). As noted in these letters, the use of lead test assemblies with different cladding materials or other properties does not necessarily require an exemption. Consistent with the regulatory path identified for lead test assemblies in the June 24, 2019, letters, licensees should evaluate lead test assembly campaigns on a case-by-case basis and take appropriate actions.

Finally, the NRC staff agrees with the licensee's proposed addition of a requirement to list in the COLR any lead test assemblies loaded into the core for each unit on a cycle-specific basis. This requirement, which also involves a corresponding change to TS 5.6.5, would keep the NRC staff apprised of any lead test assemblies in use at Palo Verde.

3.7.3 Core Operating Limits Report (TS 5.6.5)

As discussed below, the supplemental RAI response dated November 26, 2019 includes two proposed changes to TS 5.6.5.

3.7.3.1 Inclusion of Certain Information Concerning Fuel Assemblies into the COLR

The licensee proposed to modify TS 5.6.5.a to specify the following additional required content in the cycle-specific COLR for each unit concerning the fuel assemblies loaded into the core as follows:

> 13. Fuel types and cladding material in the reactor for Specification 4.2.1.a and 4.2.1.b, and the associated COLR methodologies for Specification 4.2.1.a.

The licensee proposed this change to TS 5.6.5.a to keep the NRC staff apprised of the reload fuel assemblies, cladding materials, lead test assemblies, and fuel-specific COLR methods in use at each unit at Palo Verde on a cycle-specific basis. As discussed above, the licensee proposed changes to TS 4.2.1 that would permit greater regulatory flexibility regarding fuel assembly design. Part of the NRC staff's basis for granting this increased flexibility is the corresponding changes to TS 5.6.5.a that ensure the NRC staff will be appropriately apprised of reload fuel assembly designs, cladding materials, lead test assemblies, and fuel-specific COLR methods in use at Palo Verde.

In particular, the licensee's proposal to identify in the COLR which fuel type(s) each COLR methodology is used to analyze arose in response to an NRC staff RAI. Recognizing that the licensee's proposal requests permission for long-term, simultaneous use of certain fuel assemblies designed by both Westinghouse and Framatome, in SNPB RAI-24, the NRC staff requested additional information concerning the necessity of clearly defining the fuel-specific applicability of each COLR reference. In particular, the NRC staff requested that the licensee clarify whether each COLR reference applies to certain Westinghouse fuel designs, Framatome HTP™ fuel, or certain fuel designs from both vendors. The NRC staff further requested justification for cases where COLR methodologies developed by one fuel vendor would be used to generate core operating limits for fuel supplied by a different vendor.

In its response to SNPB RAI-24 dated October 4, 2019 (Reference 5), the licensee stated that its proposed addition to TS 5.6.5.a would identify which COLR methods apply to which fuel type. The licensee stated that, at the present time, the only instances where COLR methods from one vendor would be applied to fuel assemblies manufactured by a different vendor are as described in its proposed license amendment. Should future licensing basis changes be implemented in this area, the licensee affirmed that they would be made in accordance with applicable regulatory requirements (e.g., 10 CFR 50.59, "Changes, tests, and experiments"; 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit").

The NRC staff's review finds that the licensee's response to SNPB RAI-24 is acceptable because it proposed clearly to delineate the fuel-specific applicability of the COLR methods listed in TS 5.6.5.b. In particular, absent the specific exceptions described in the licensee's LAR that have been found acceptable in this SE, the licensee would apply vendor-specific methodologies as intended when determining operating limits for fuel assemblies in the reactor core. The licensee's identification in the COLR of which fuel types each COLR method is

applied to would keep the NRC staff apprised of the licensee's implementation of approved analysis methods on a fuel-specific basis.

Based upon the evaluation above, the NRC staff finds that the licensee's proposed change to TS 5.6.5.a is acceptable. The additional COLR content proposed for inclusion in TS 5.6.5.a would permit appropriate NRC staff monitoring of the licensee's core reload process in light of the additional regulatory flexibility conferred by changes to other TSs included in this proposed amendment (i.e., TS 4.2.1 and TS 5.6.5.b).

3.7.3.2 New Analytical Methods for Determining Core Operating Limits (TS 5.6.5.b)

The licensee proposed to modify TS 5.6.5.b by listing additional references to approved TRs containing methodologies for determining core operating limits for Framatome HTP™ fuel. Specifically, the licensee proposed to add the following TR methods listed below in Table 6 to TS 5.6.5.b.

Table 6: Proposed Additions to TS 5.6.5.b, COLR References[17]

Topical Report	Title	Application
EMF-2103P-A	Realistic LBLOCA Methodology for PWRs	TS 3.2.1, Linear Heat Rate
EMF-2328(P)(A)	PWR SBLOCA Evaluation Model, S-RELAP5 Based	TS 3.2.1, Linear Heat Rate
BAW-10231P-A	COPERNIC Fuel Rod Design Computer Code	TS 3.2.1, Linear Heat Rate
BAW-10241(P)(A)	BHTP DNB Correlation Applied with LYNXT	TS 3.2.4, Departure from Nucleate Boiling Ratio
EPRI-NP-2511-CCM-A	VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores	TS 3.2.4, Departure from Nucleate Boiling Ratio

Each of these TR methods has been previously reviewed and approved by the NRC staff on a generic basis. Furthermore, as documented in the foregoing SE (see in particular Sections 3.3 and 3.5), the NRC staff's review has found that the calculational methods described in the TRs listed in Table 6 are acceptable for implementation at Palo Verde. Therefore, the NRC staff finds that the addition of the approved TR references listed in Table 6 to Palo Verde TS 5.6.5.b are acceptable.

3.8 Human Performance

3.8.1 Description of Operator Action(s) and Assessed Safety Significance

In the LAR dated July 6, 2018 (Reference 1), APS described the proposed changes to the TSs for Palo Verde, which would be required to support implementation of Framatome CE 16HTP™ fuel design. The licensee stated that although the TS changes do not involve any plant modifications that could affect system reliability, component performance, or the possibility

[17] Note that Palo Verde's TS 5.6.5.b omits the dates and revision numbers for approved topical reports in accordance with a previously approved TS change consistent with Technical Specifications Task Force (TSTF) Traveler TSTF-363, "Revise Topical Report references in ITS [improved Technical Specifications] 5.6.5, COLR" (Reference 85). The licensee maintains dates and revision numbers for each COLR reference in the COLR.

of operator error, there is a new time requirement for an existing operator action. In Section 7.2, "New Time Requirement for LOCA Mitigation Operator Action," in Attachment 10 to the LAR, the licensee identified the affected operator action as the manual action to stop all RCPs when pressurizer pressure drops below the RCP NPSH limits, during a SBLOCA.

In accordance with the generic risk categories established in Appendix A to NUREG-1764 (Reference 25), the manual operator action of tripping RCPs reviewed herein is considered "risk-important," because upon a loss of cooling to the RCP seals, it is important for the operators to quickly trip the pumps in order to prevent an RCP seal LOCA. Therefore, the human action (HA) that is the subject of this review was identified as a Level I safety significance HA. Further, the NRC staff performed a qualitative assessment and considered several specific factors associated with the proposed change. The NRC staff noted that the proposed change does not introduce any new HAs, does not involve a change in automation for the operator action associated with tripping of RCPs, does not involve a change to the human-system interface, and does not involve a change in operator staffing levels or operator qualifications. The NRC staff considered the above-mentioned factors, as part of the qualitative assessment, and determined that a reduction of the level of human factors engineering (HFE) review is appropriate, because there are no changes to the HA, other than the new time requirement for completing the action. Therefore, the NRC staff performed a Level II HFE review.

3.8.2 Operating Experience Review

In the licensee's letter dated May 17, 2019 (Reference 4), the licensee described the operating experience review that was performed by APS, as it relates to the proposed change establishing a new time requirement to stop all RCPs within 5 minutes following loss of subcooling margin. The licensee stated that a delayed RCP trip study was performed for Palo Verde in accordance with the methodology described in the Framatome TR EMF-2328(P)(A), Revision 0, Supplement 1(P)(A), Revision 0 (Reference 30), as approved in the NRC SE Report dated February 1, 2017 (Reference 86). The licensee further stated that the time critical action (TCA) requirement to trip the RCPs after a loss of subcooled margin was previously implemented at three CE designed plants. The APS review of internal and external operating experience did not identify any human performance issues associated with neither the proposed operator action, nor associated procedures and training. In addition, the licensee stated that a pilot testing conducted at Palo Verde concluded that, on average, control room operators would trip the RCPs within 30 seconds following loss of subcooled margin, which is well within the proposed TCA time limit of 5 minutes.

Based on the fact that the licensee considered applicable internal and external operating experience, the NRC staff finds that the licensee's treatment of the operating history review HFE program element is acceptable.

3.8.3 Procedure Development

APS stated in Section 7.2 in Attachment 10 to LAR dated July 6, 2018, that the existing procedure 40EP-9EO01, "Standard Post Trip Actions," provides those operator actions, including immediate actions, which must be accomplished following an automatic or manually initiated reactor trip and the diagnostic actions necessary to determine a preliminary diagnosis of the events. The requirement to stop all RCPs when pressurizer pressure drops below the RCP NPSH is unchanged from the existing Step 5.4 of this procedure. Therefore,

no changes to procedure 40EP-9EO01 are required. In the licensee's letter dated May 17, 2019, the licensee further clarified that the tripping of RCPs is also required by procedure 40EP-9EO03, "Loss of Coolant Accident," which specifies actions to mitigate the efforts of LOCA, which would be entered following preliminary event diagnosis pursuant to procedure 40EP-9EO01. Other emergency operating procedures that may be used during RCS depressurization events also include provisions for tripping of the RCPs.

The licensee further stated that the requirement to stop all RCPs within 5 minutes following pressurizer pressure dropping below the RCP NPSH limits during a SBLOCA event will be added to procedure 40DP-9ZZ04, "Time Critical Action (TCA) Program," which provides a means to document periodic validation of credited action items, and a means to ensure that changes to the plant or to procedures or protocols do not invalidate credited action items.

In Attachment 1 to the LAR dated July 6, 2018, the licensee provided the following Regulatory Commitment:

> APS commits to incorporating into the Time Critical Action Program the requirement to stop all Reactor Coolant Pumps within 5 minutes following pressurizer pressure dropping below the RCP NPSH limits during a Small Break LOCA event. Scheduled completion date: upon implementation of Framatome HTP™ Fuel.

Based on the fact that there are no changes needed to the operating procedures, and the new time requirement for stopping all RCPs within 5 minutes following pressurizer pressure dropping below the RCP NPSH limits during a SBLOCA event will be added to the TCA program as described in the abovementioned Regulatory Commitment, the NRC staff finds the treatment of the procedure development HFE program element is acceptable.

3.8.4 Training Program Development

APS stated in Section 7.2 in Attachment 8 to the LAR dated July 6, 2018, that since the procedure step to stop all RCPs when pressurizer pressure drops below the RCP NPSH limits already exists in procedure 40EP-9EO01 (without a specific time limit), the operators are already trained on and familiar with the procedure step. In the licensee's letter dated May 17, 2019, the licensee further clarified that the operators receive periodic training on procedure 40EP-9EO01 and other procedures that include provisions for tripping of the RCPs following a loss of subcooled margin and are therefore familiar with the task demand.

The licensee further stated that the new requirement to stop all RCPs within 5 minutes following the pressurizer pressure dropping below the RCP NPSH limits will be added to the TCA program described in procedure 40DP-9ZZ04.

In the licensee's letter dated May 17, 2019, the licensee provided additional information regarding the process of updating the operator training program, to clarify how the commitment to enter the new time requirement into the TCA program results in the necessary changes to the operator training program being implemented. The licensee stated that procedure 40DP-9ZZ04 requires that impact reviews be issued to potentially affected workgroups when the TCA program is revised to add a new TCA. One of these impact reviews will be assigned to the Palo Verde Operations Training Organization, which will prepare a needs analysis in accordance with procedure 15DP-0TR08, "Systematic Approach to Training (SAT)." The needs analysis

identifies the required training and determines whether any changes to affected training programs are needed due to the addition of the TCA. APS stated that they do not anticipate that this evaluation will result in a new task or knowledge requirement, as the direction to trip RCPs following a loss of subcooled margin already exists in the Palo Verde standard post-trip actions procedure 40EP-9EO01, as well as other emergency operating procedures, including procedure 40EP-9EO03.

The licensee further stated that procedure 40DP-9ZZ04 requires that TCAs be revalidated at least every 5 years. This ensures that plant operators retain the knowledge and abilities to respond to TCAs within their time requirements. Periodic revalidation may also reveal whether subsequent changes in plant design or procedures have caused a change in plant operator performance, which may necessitate further modifications to operator training.

The NRC staff finds that the licensee's treatment of the training program development HFE program element is acceptable.

3.8.5 Human Factors Verification and Validation

The licensee stated in Section 7.2 in Attachment 8 to its letter dated July 6, 2018, that five operating crews were evaluated on the new time limit, and the average time of completion of the HA was "well under one minute."

In the licensee's letter dated May 17, 2019, the licensee provided additional information regarding the planned and completed activities conducted by APS to provide reasonable assurance that the HA can be accomplished within the newly established time limit. The licensee stated that prior to submitting the LAR dated July 6, 2018, APS performed a pilot testing, which utilized five different crews of licensed operators and involved a SBLOCA scenario that required tripping the RCPs. The pilot testing indicated that, on average, test participants would trip the RCPs approximately 30 seconds following loss of subcooled margin. No human engineering discrepancies (HEDs) were identified during pilot testing. APS concluded that the results of pilot testing provide some assurance that sufficient margin exists to allow for more variation in actual performance than in validation test performance.

The licensee further stated that following NRC approval of the LAR, APS will perform verification and validation (V&V) of RCP trip timing as a TCA in accordance with the requirements of Palo Verde procedure 40DP-9ZZ04. The validation team personnel, who are normally members of the operations training organization, will have independence from the personnel responsible for the actual design. The validation team may also include other personnel, such as site procedures standards personnel who have responsibility for procedure maintenance and revision. The licensee further clarified that the V&V test objectives will be established to ensure that the integrated system adequately supports plant personnel in safely operating the plant (i.e., that the RCP trip timing as a TCA is met as specified in the licensing basis). The validation testbed for V&V of the RCP trip timing TCA will be the Palo Verde control room dynamic simulators, which are normally used for training of licensed operators. The Palo Verde control room simulators represent the actual control rooms with high physical fidelity including the presentation of alarms, displays, controls, operator aids, and procedures. The licensee stated that the information, controls, and response of the testbed during V&V is expected to accurately match that, which would be present during an actual SBLOCA. Validation of the new RCP trip timing TCA will be performed by at least three different crews, to provide reasonable assurance that the TCA can be completed within the required time. The licensee stated that, crews that

participate in V&V will be selected without bias (for example, crews identified as being more experienced).

The licensee further stated that the process for data analysis and HED identification during V&V will be as described in Palo Verde procedure 40DP-9ZZ04. Specifically, validation of a TCA is considered adequate if the TCA is completed in a shorter duration than 80 percent of the TCA required time. If the TCA is completed within 80 percent to 100 percent of the required time, then additional validation of the TCA will be performed with other participants, and/or an engineering review will be performed to determine methods of reducing the actual performance time. Any identified HEDs will be addressed by remediations, retesting, performance of additional validations using other participants, and/or evaluating for degrading trends in TCA completion time, depending on the nature of the HED.

Based on the fact that the tripping of the RCPs following a loss of subcooling is an existing requirement in the Palo Verde emergency operating procedures, that licensed operators are trained on the use of these procedures, that the pilot testing concluded that TCA performance would be well within the five-minute time limit, and the licensee's plans to perform additional V&V activities following the approval of the proposed LAR by the NRC, the NRC staff finds that the licensee's treatment of the Human Factors V&V HFE program element is acceptable.

3.8.6 Human Performance Monitoring

In Attachment 1 to the LAR dated July 6, 2018, APS provided a regulatory commitment to incorporate into the TCA program the requirement to stop all RCPs within 5 minutes following pressurizer pressure dropping below the RCP NPSH limits during a SBLOCA event. The licensee further stated in Section 7.2 in Attachment 8 to the LAR, that the addition of the new time requirement to the TCA program provides a means to document periodic validation of credited action items and serves as a means to ensure that changes to the plant or to procedures or protocols do not invalidate credited action items.

The NRC staff evaluated the statements by the licensee and finds that the administrative protection against inadvertent change and the periodic validation of items documented in the TCA program is an acceptable strategy for ensuring that the conclusions drawn from the integrated system validation remain valid with time, and that no significant safety degradation occurs because of any changes made in the plant. Therefore, the NRC staff finds that the licensee's treatment of the human performance monitoring HFE program element is acceptable.

4.0 LICENSE CONDITIONS

As discussed in the succeeding sections of this SE, in order to address issues identified during the NRC staff's review, the licensee proposed the following license conditions to support the granting of the license amendment:

- revision of the wording of an existing license condition concerning application of the Westinghouse/Combustion Engineering FATES3B fuel performance code, and

- addition of a new license condition to forbid operation with mixed batches of fresh fuel in the reactor core.

4.1 Revision to License Condition Concerning FATES3B Fuel Performance Code

The NRC staff's acceptance of Amendment No. 205, permitting implementation of Westinghouse NGF at Palo Verde, Units 1, 2, and 3, was conditioned upon the licensee applying an appropriate radial falloff curve penalty to address potentially nonconservative predictions of stored energy and other parameters by the FATES3B fuel performance code as a function of burnup for Westinghouse NGF assemblies (Reference 68).

In its review of the present LAR, the NRC staff observed that the licensee had proposed generic wording (e.g., Westinghouse-supplied fuel) in lieu of naming specific fuel designs or characteristics (e.g., cladding materials) in several of the proposed TS changes described above. In SNPB RAI-26, the NRC staff requested that the licensee address whether similarly encompassing language would be appropriate for the existing license condition imposed in Amendment No. 205, which currently refers specifically to Westinghouse NGF.

In its supplemental RAI response dated November 26, 2019, the licensee responded to SNPB RAI-26 by proposing to revise the existing license condition from Amendment No. 205 to explicitly indicate that the radial falloff curve penalty is applicable to both NGF and any future Westinghouse fuel designs analyzed with the FATES3B code. The revised license condition is reproduced below, with bold text indicating the proposed revision:

> APS shall apply a radial power fall off (RFO) curve penalty, equivalent to the fuel centerline temperature reduction in Section 4 of Attachment 8 to the Palo Verde license amendment request dated July 1, 2016, to accommodate the anticipated impacts of thermal conductivity degradation (TCD) on the predictions of FATES3B at high burnup for Westinghouse Next Generation Fuel **or to future Westinghouse-supplied fuel designs introduced at PVNGS to which the FATES3B fuel performance code would be applied.**
>
> To ensure the adequacy of this RFO curve penalty, as part of its normal reload process for each cycle that analysis using FATES3B is credited, APS shall verify that the FATES3B analysis is conservative with respect to an applicable confirmatory analysis using an acceptable fuel performance methodology that explicitly accounts for the effects of TCD. The verification shall confirm satisfaction of the following conditions:
>
> i. The maximum fuel rod stored energy in the confirmatory analysis is bounded by the maximum fuel rod stored energy calculated in the FATES3B and STRIKIN-II analyses with the RFO curve penalty applied.
>
> ii. All fuel performance design criteria are met under the confirmatory analysis.
>
> If either of the above conditions cannot be satisfied initially, APS shall adjust the RFO curve penalty or other core design parameters such that both conditions are met.

The NRC staff found the licensee's proposed revision to the license condition from Amendment No. 205 acceptable because it clarifies that an RFO curve penalty, verified using an acceptable

fuel performance methodology that explicitly accounts for fuel TCD, must be applied when FATES3B is used to analyze not only Westinghouse NGF, but also future Westinghouse fuel designs. The revised license condition would ensure that the licensing basis changes requested in the proposed amendment, which among other things may offer increased regulatory flexibility with respect to licensing new fuel designs, would be appropriately balanced by a binding requirement to continue accounting for the impacts of TCD in a technically defensible manner.

4.2 New License Condition Concerning Mixed Batches of Fresh Fuel

The NRC staff's review identified a question concerning whether approval of the proposed license amendment would permit operation with mixed batches of fresh fuel. This question was based upon the observation that the analytical methodologies used by the licensee for reactor core design and safety analysis have generally been developed based upon implicit assumptions that:

- all fresh fuel is of a single design and
- the party performing the analysis has access to all relevant information concerning the fresh fuel design that is expected to set most or all core operating limits.

Regarding the first point, incorporation of different fuel designs into a reactor core generally results in increased analytical uncertainties. For instance, design differences associated with features such as spacer grids, flow mixers, fuel rod dimensions, etc., can result in diversion of flow from fuel assemblies of one design to those of another. Furthermore, analytical methodologies developed by a given fuel vendor may lack approved and validated models for directly simulating important physical behaviors necessary to initialize or simulate certain events in the plant safety analysis (e.g., departure from nucleate boiling propagation, fuel cladding strain, fuel pellet-cladding interactions) for fuel assemblies designed by a different vendor. Consequently, engineering assumptions or other approximate modeling approaches with increased uncertainty are typically employed.

The potential for increased analytical uncertainties, as exemplified above, has been accounted for in existing mixed-core methodologies that were developed specifically to address transitions from a legacy fuel design to a single, new type of reload assembly.[18] Reliance upon the premise that all fresh fuel is of a consistent design allows existing mixed-core methods considerable simplification relative to the general case where the limiting fuel design is unknown. Hence, the conservative direction for various engineering assumptions becomes clearer, and the use of conservative approximations for legacy fuel can generally be offset by its increased margins to operating limits.

Regarding the second point above (i.e., access to all relevant information concerning the fresh fuel design), while APS directly performs many fuel-related analyses for Palo Verde, other calculations, including structural evaluations and LOCA analyses, are typically performed by the fuel vendor. When a reactor core is designed with mixed batches of fresh fuel from different vendors, in lieu of modeling approaches based upon (typically vendor-proprietary) data associated with a specific fuel design and its associated behavior, outside vendors may be

[18] In the NRC staff's experience, domestic reactor licensees have historically operated with all fresh assemblies in a given reload batch typically being of the same design.

required to use approximate engineering assumptions and methods that involve increased uncertainty for fresh fuel assemblies manufactured by another vendor. Furthermore, precise determination of the magnitude of the uncertainty associated with the application of these approximate methods can be intractable. While limited use of such approximations may be appropriate for less-limiting legacy fuel designs in typical fuel transitions, the NRC staff has not reviewed or approved such approximate approaches for unrestricted long-term application with mixed batches of fresh fuel.

Two examples of TRs referenced by Palo Verde that discuss mixed-core configurations are EMF-2103P-A, Revision 3 (see TR EMF-2103R3Q1P, discussion of RAI 14 (Reference 27)), and CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model" (see Section 3.2 of Reference 75)). In both cases, these TRs discuss mixed-core applications in terms of transitioning from a legacy assembly manufactured by one vendor to a new fresh assembly design manufactured by a different vendor. The continual loading of mixed batches of fresh fuel provided by multiple vendors does not appear to be addressed by, and hence appears to be beyond the scope of, these topical report methods.

In light of these concerns, the NRC staff requested in SNPB RAI-25 that APS provide justification if its analytical methods for reactor core design and safety analysis that are listed in TS 5.6.5.b would be applied to core configurations involving mixed batches of fresh fuel. If the analytical methods would be applied to mixed batches of fresh fuel, the NRC staff requested that the licensee address the potential for increased uncertainty in the predicted results and provide evidence that the methods have been validated for this application. If the analytical methods would not be applied to mixed batches of fresh fuel, the NRC staff requested that the licensee cite an existing regulatory requirement or propose a binding requirement that would prohibit such applications.

In its supplemental RAI response dated November 26, 2019, the licensee responded by restating that (1) its normal reload process will continue to be based upon full batch reloads from a single fuel vendor and (2) when fuel transitions are implemented, they will continue to involve a shift from a full core of one fuel type to a full core of another fuel type. The licensee further stated that it is no longer seeking permission for operation with mixed batches of fresh reload fuel assemblies in the reactor core at any unit at Palo Verde.

To support approval of the proposed license amendment, the licensee further proposed the following license condition to serve as a binding restriction to forbid operation with mixed batches of fresh fuel without prior NRC staff approval:

> Prior to use of fresh fuel from multiple fuel vendors in a single reload batch, APS will obtain NRC approval of the methodology used to perform the associated reload safety analyses. Lead Test assemblies per Technical Specification (TS) 4.2.1.b are not considered mixed fresh fuel.

The NRC staff found that the proposed license condition would constitute a binding requirement forbidding application of the existing analysis methods used to determine operating limits at Palo Verde to reactor cores comprised of mixed batches of fresh fuel. Such a restriction, in conjunction with the granting of the present license amendment, would prevent application of the existing analysis methods listed in TS 5.6.5.b of the Palo Verde TS under conditions for which they have not previously been reviewed and approved. Therefore, the NRC staff finds the licensee's adoption of the proposed license condition is acceptable.

5.0 SUMMARY

The NRC staff has reviewed the LAR and its supplements to evaluate the acceptability of the Palo Verde transition to Framatome CE 16HTP™ fuel design with Framatome safety analyses and core design methodologies. Based on its review, the NRC staff has determined that the licensee provided an adequate technical basis to support the proposed LAR. Specifically, the NRC staff has determined that the licensee has demonstrated that (1) the fuel assembly and fuel rod analyses satisfy the NRC-approved design criteria for normal and faulted conditions, (2) the licensee complies with the staff limitations and conditions imposed for application of topical reports used in the execution of the LAR, (3) Framatome codes and methods are applicable for Palo Verde, (4) the safety analysis results submitted to the NRC staff demonstrate compliance with applicable regulatory requirements, and (5) the proposed TS changes and license conditions are acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arizona State official was notified of the proposed issuance of the amendment on February 15, 2020. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to 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 January 8, 2019 (84 FR 90), 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.

8.0 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.

9.0 REFERENCES

1. Lacal, M. L., Arizona Public Service Company, letter to U.S. Nuclear Regulatory Commission, "Palo Verde Nuclear Generating Station Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529, and 50-530, "License Amendment Request and Exemption Request to Support the Implementation of Framatome High Thermal Performance Fuel," dated July 6, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18187A417).

2. Lacal, M. L., Arizona Public Service Company, letter to U.S. Nuclear Regulatory Commission, "Palo Verde Nuclear Generating Station Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529, and 50-530, "Supplemental Information Regarding License Amendment Request and Exemption Request to Support the Implementation of Framatome High Thermal Performance Fuel," dated October 18, 2018 (ADAMS Accession No. ML18296A466).

3. Lacal, M. L., Arizona Public Service Company, letter to U.S. Nuclear Regulatory Commission, "Palo Verde Nuclear Generating Station Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529, and 50-530, Renewed Operating License Nos. NPF-41, NPF-51, NPF-74, "Audit Presentation Slides Regarding License Amendment Request and Exemption Request to Support the Implementation of Framatome CE 16HTP™ Fuel," dated March 1, 2019 (ADAMS Accession No. ML19060A298).

4. Lacal, M. L., Arizona Public Service Company, letter to U.S. Nuclear Regulatory Commission, "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529, and 50-530, Renewed Operating License Nos. NPF-41, NPF-51, NPF-74, "Response to NRC Staff Request for Additional Information from Reactor Assessment and Human Performance Branch Regarding License Amendment and Exemption Requests Related to the Implementation of Framatome High Thermal Performance Fuel," dated May 17, 2019 (ADAMS Accession No. ML19137A118).

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Date: March 4, 2020

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 - ISSUANCE OF AMENDMENT NOS. 212, 212, AND 212 TO REVISE TECHNICAL SPECIFICATIONS TO SUPPORT THE IMPLEMENTATION OF FRAMATOME HIGH THERMAL PERFORMANCE FUEL (EPID L-2018-LLA-0194) DATED MARCH 4, 2020

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ADAMS Accession Nos.:
Proprietary ML20031C968
Non-proprietary ML20031C947 ***by memo** ****by e-mail**

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DATE	02/10/2020	02/10/2020	01/17/2020	01/17/2020
OFFICE	NRR/DSS/STSB/BC *	NRR/DIRS (DRO)/IRAB (IOLB)/BC*	OGC (NLO)**	NRR/DORL/LPL4/BC
NAME	VCusumano	AMasters (CCowdrey)	DRoth	JDixon-Herrity
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