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

January 23, 2018

Mr. Robert S. Bement
Executive Vice President Nuclear/
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 AMENDMENTS TO REVISE TECHNICAL SPECIFICATIONS TO SUPPORT THE IMPLEMENTATION OF NEXT GENERATION FUEL (CAC NOS. MF8076, MF8077, AND MF8078; EPID L-2016-LLA-0005)

Dear Mr. Bement:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 205 to Renewed Facility Operating License No. NPF-41, Amendment No. 205 to Renewed Facility Operating License No. NPF-51, and Amendment No. 205 to Renewed Facility Operating License No. NPF-74 for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated July 1, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16188A336), as supplemented by letters dated June 2, 2017 and December 15, 2017 (ADAMS Accession Nos. ML17153A373 and ML17349A990, respectively).

The amendments revise the TSs to support the implementation of next generation fuel (NGF). In addition to the amendments and in accordance with the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.12, "Specific exemptions," the licensee requested an exemption from certain requirements of 10 CFR 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 Optimized ZIRLO™ as a fuel rod cladding material. In summary, the proposed change will allow for the implementation of NGF including the use of Optimized ZIRLO™ fuel rod cladding material. The NGF assemblies contain advanced features to enhance fuel reliability, thermal performance, and fuel cycle economics. The NRC staff's review of the exemption has been completed and is documented in a separate safety evaluation (ADAMS Accession No. ML17319A214).

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

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

Sincerely,

MOBarnin for

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. 205 to NPF-41
2. Amendment No. 205 to NPF-51
3. Amendment No. 205 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. 205
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 dated July 1, 2016, as supplemented by letters dated June 2, 2017 and December 15, 2017, 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. 205, 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) Additional Conditions

   The Additional Conditions contained in Appendix D, as revised through Amendment No. 205, 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

Robert J. Pascarelli, 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: January 23, 2018

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. 205
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 dated July 1, 2016, as supplemented by letters dated June 2, 2017 and December 15, 2017, 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. 205, 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) Additional Conditions

   The Additional Conditions contained in Appendix D, as revised through Amendment No. 205, 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

Robert J. Pascarelli, 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: January 23, 2018

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. 205
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 dated July 1, 2016, as supplemented by letters dated June 2, 2017 and December 15, 2017, 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. 205, 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) Additional Conditions

   The Additional Conditions contained in Appendix D, as revised through Amendment No. 205, 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

Robert J. Pascarelli, 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: January 23, 2018

ATTACHMENT TO LICENSE AMENDMENT NOS. 205, 205, AND 205

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 Licenses Nos. NPF-41, NPF-51, and NPF-74, Appendix A – Technical Specifications, and Appendix D – Additional Conditions, 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
4.0-1	4.0-1
5.6-7	5.6-7

Appendix D – Additional Conditions

REMOVE	INSERT
--	- 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. 205, 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. 205, 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. 205, 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

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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) Supplement No. 1 to NUREG-0737 Requirements

Deleted

(9) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 205, 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. 205, 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. 205, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Additional Conditions.

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. Each assembly shall consist of a matrix of Zircaloy or ZIRLO or Optimized ZIRLO fuel rods 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. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. Other cladding material may be used with an approved exemption.

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

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]

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.

(continued)

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. 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.

ENCLOSURE 5

SAFETY EVALUATION RELATED TO NEXT GENERATION FUEL

LICENSE AMENDMENT REQUEST

ARIZONA PUBLIC SERVICE COMPANY

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

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

(NON-PROPRIETARY)

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. 205, 205, AND 205 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

1.0 INTRODUCTION

By letter dated July 1, 2016 (Reference 1), as supplemented by letters dated June 2, 2017 (Reference 2) and December 15, 2017 (Reference 48), Arizona Public Service Company (APS, the licensee) submitted a license amendment request (LAR) for Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, requesting an approval to revise the Technical Specifications (TSs) to support the implementation of next generation fuel (NGF). The proposed changes would revise TS 4.2.1, "Reactor Core, Fuel Assemblies" and TS 5.6.5.b, "Core Operating Limits Report (COLR)" to allow the use of Combustion Engineering (CE) 16×16 NGF clad with the Optimized ZIRLO™ material in PVNGS, Units 1, 2, and 3. The proposed change to TS 4.2.1 would add the phrase "or Optimized ZIRLO™" to support the use of this fuel rod cladding material at PVNGS. The proposed change to TS 5.6.5.b would add five documents to PVNGS's list of previously reviewed and approved analytical methods for determining core operating limits. These two proposed TS changes will be described and assessed subsequently in this safety evaluation (SE).

Approval of this LAR would allow the use of CE 16×16 NGF clad with Optimized ZIRLO™, [1] at PVNGS, Units 1, 2, and 3. Therefore, the U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the results of the applicable safety analyses (i.e., mechanical design, core design, fuel performance, thermal hydraulics, corrosion, and loss-of-coolant accident (LOCA) and non-LOCA analyses) and prototype testing that demonstrate that NGF can be safely operated in PVNGS for both full-core implementation and transitional cycles containing mixed cores of NGF and the currently used "Value Added Fuel," referred to herein as STD.

In addition to the LAR and in accordance with the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.12, "Specific exemptions," the licensee is requesting an exemption from certain requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems [ECCS] for light-water nuclear power reactors," and 10 CFR Part 50,

[1] "Optimized ZIRLO™", Low Tin ZIRLO™, and "ZIRLO™" are trademarks or registered trademarks of Westinghouse Electric Company, LLC (Westinghouse).

Appendix K, "ECCS Evaluation Models," to allow the use of Optimized ZIRLO™ as a fuel rod cladding material. The cladding material exemption supports the fuel transition to CE 16x16 NGF and has been documented separately and concurrently (Reference 3). In summary, the proposed change will allow for the implementation of NGF including the use of Optimized ZIRLO™ fuel rod cladding material. The NGF assemblies contain advanced features to enhance fuel reliability, thermal performance, and fuel cycle economics.

The supplemental letters dated June 2 and December 15, 2017, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 4, 2016 (81 FR 68469).

2.0 REGULATORY EVALUATION

As specified in 10 CFR 50.36, "Technical specifications," TS must be included by applicants for a license authorizing operation of a production or utilization facility. In particular, 10 CFR 50.36(c) requires that TS include (1) safety limits, 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. The proposed change to TS 4.2.1 is within the design features category (i.e., fuel assemblies within the reactor core), and the proposed change to TS 5.6.5.b is within the administrative controls category.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Section 4.2, "Fuel System Design" (Reference 4), provides regulatory guidance for the review of fuel rod cladding materials and fuel design. In addition, the SRP provides guidance for compliance with the applicable General Design Criteria (GDC) in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." According to the SRP Section 4.2, the fuel system safety review provides assurance that:

- fuel is not damaged as a result of normal operation and anticipated operational occurrences (AOOs),
- fuel damage is never so severe as to prevent control rod insertion when it is required,
- the number of fuel rod failures is not underestimated for postulated accidents, and
- coolability of the fuel is always maintained.

Appendix A to 10 CFR Part 50, GDC 10, "Reactor Design," establishes specified acceptable fuel design limits (SAFDLs) that should not be exceeded during any condition of normal operation, including the effects of AOOs. SAFDLs are established to ensure that the fuel is not damaged; that is, the fuel rods do not fail and the fuel system dimensions remain within operational tolerances.

Appendix A to 10 CFR Part 50, GDC 27, "Combined reactivity control systems capability," relates to the reactivity control system being designed with appropriate margin and, in conjunction with the ECCS, being capable of controlling reactivity and cooling the core under post-accident conditions.

Appendix A to 10 CFR Part 50, GDC 35, "Emergency Core Cooling," relates to providing an ECCS to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) the clad metal-water reaction is limited to negligible amounts. Specific limits on fuel and clad damage under LOCA conditions are contained in 10 CFR 50.46, as discussed further below in Section 3.5.8 of this SE as well as the NRC staff's SE (Reference 3) on APS's exemption request that was reviewed concurrently.

3.0 TECHNICAL EVALUATION

3.1 Introduction

The PVNGS cores each contain 241 fuel assemblies with 76 full-strength (i.e., containing boron carbide and possibly silver-indium-cadmium absorbers) and 13 part-strength (i.e., containing Inconel 625 slugs) control element assemblies (CEAs). Each current STD fuel assembly consists of a 16×16 square matrix of zircaloy- or ZIRLO™-clad fuel rods with an initial fuel composition of natural or slightly enriched uranium dioxide (UO_2). Each STD fuel assembly consists of 236 fuel rods, four outer guide tubes, one center instrument tube, Inconel top and bottom grids, nine zircaloy mid-grids, and upper and lower end fittings.

The subject of this LAR is the transition from STD fuel to CE 16×16 NGF, herein referred to as simply "NGF." NGF assemblies consist of a 16×16 square matrix of Optimized ZIRLO™-clad fuel rods with an initial fuel composition of natural or slightly enriched UO_2. Each NGF assembly consists of 236 fuel rods, four outer guide tubes, one center instrument tube, Inconel top and bottom grids, nine low tin ZIRLO™ side-supported vaned mid-grids, an additional two low tin ZIRLO™ intermediate flow mixing (IFM) grids, and upper and lower end fittings.

Some notable differences between the two fuel designs are:

- Use of Optimized ZIRLO™ fuel cladding material with a zirconium diboride (ZrB_2) integral fuel burnable absorber (IFBA) coating
- Use of side-supported vaned mid-grids
- Addition of two IFM grids
- Slight reduction to fuel rod and pellet diameters
- Increased assembly pressure drop

Specific details of the NGF design can be found in proprietary topical report (TR) WCAP-16500-P-A (Reference 5) and related supplements, Supplement 1-A (Reference 6) and Supplement 2-A (Reference 7).

3.2 Proposed Changes to TS 4.2.1 and TS 5.6.5.b

Transition to NGF requires modifications to TS 4.2.1 and TS 5.6.5.b. TS 4.2.1 specifies certain parameters of the fuel assembly design, number of assemblies, and implementation at PVNGS, including the fuel rod cladding material used in construction of the assemblies. Since NGF is

clad with Optimized ZIRLO™, wording must be added to TS 4.2.1 to allow its use. Specifically, the proposed change to TS 4.2.1 will add "or Optimized ZIRLO™" to the cladding materials listed in the sentence describing the fuel rods. APS provided marked up and clean, revised TS pages reflecting the proposed change in attachments to the submittal (Reference 1).

The NRC staff has previously approved Optimized ZIRLO™ fuel cladding based upon (1) similarities with standard ZIRLO™, (2) demonstrated material performance, and (3) a commitment to provide irradiated data and validate fuel performance models ahead of burnups achieved in batch applications. Specifically, Addendum 1-A (Reference 8) to CENPD-404-P-A (Reference 9) describes the approval for use of Optimized ZIRLO™ as a cladding material and Addendum 2-A (Reference 10) describes the applicable corrosion model. Applicability of the related addenda to PVNGS has been evaluated by NRC staff in this SE. This TS change is also supported by the related cladding material exemption to 10 CFR 50.46 and 10 CFR Part 50, Appendix K requirements. Based on the acceptance of the addenda for use at PVNGS in this SE, in conjunction with the approval of the cladding material exemption, the NRC staff finds the change to TS 4.2.1 acceptable, as written in the provided marked-up and clean, revised TS pages.

TS 5.6.5.b identifies supporting technical documents by number and title. Implementation of NGF requires reanalysis of many of the supporting Analyses of Record (AORs) that demonstrate the ability for PVNGS to operate safely within federal regulations. Updated evaluations of these analyses have been reviewed as part of this SE. Due to the design changes featured in NGF, a number of new methodologies have been developed. These methodologies, along with a number of supplements (or addenda/revisions), have been reviewed and approved by NRC for NGF application. In order to support the PVNGS, Units 1, 2, and 3 licensing basis, and to be used in updated AORs, the new methodologies must be added to the list contained in TS 5.6.5.b.

The new methodologies being added to the list in TS 5.6.5.b include:

(1) WCAP-16500-P-A, "CE 16×16 Next Generation Fuel Core Reference Report" (Reference 5)

(2) WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor [PWR] Non-LOCA Thermal-Hydraulic Safety Analysis"[2] (Reference 11)

(3) CENPD-387-P-A, "ABB Critical Heat Flux Correlations for PWR Fuel" (Reference 12)

(4) WCAP-16523-P-A, "Westinghouse Correlations WSSV [Westinghouse Side Supported Vane] and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes" (Reference 13)

(5) WCAP-16072-P-A, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs" (Reference 14)

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[2] Note that the Westinghouse version of the VIPRE-01 code described in WCAP-14565-P will frequently be referred to in this SE as "VIPRE-W" (i.e., with the intent of distinguishing the Westinghouse version of VIPRE-01 from different versions of the code in use by other organizations). However, particularly in quotations, the original designation from WCAP-14565-P-A (i.e., "VIPRE-01") will also be used.

The NRC staff notes that PVNGS has adopted Technical Specification Task Force (TSTF) Traveler TSTF-363, "Revise Topical Report References in ITS [Improved TS] 5.6.5, COLR" (Reference 15) in Amendment No. 137 to the PVNGS Operating Licenses. Therefore, TS 5.6.5.b identifies supporting technical documents by number and title only. The COLR more specifically identifies these documents, including addenda, revisions, and supplements. Therefore, a number of amending documents are referenced in support of the fuel transition, but are not specifically added to the TS 5.6.5.b list since the original versions of the documents are already specified by number and title in the list. Rather, these documents are to be added to the COLR list in support of the transition. The NRC staff review will assess the applicability of all supporting documents, regardless of where they are referenced.

Based on the approval for use of these methodologies and their supporting amending documents at PVNGS for NGF, the NRC staff finds the addition of the methodologies to PVNGS TS 5.6.5.b to be acceptable, as specified in the provided marked-up and clean, revised TS pages.

3.3 Analytical Methodologies Supporting NGF Implementation at PVNGS

The NRC staff's review in this section aims to ensure that the methodologies used to update the PVNGS AORs for implementation of NGF, are applicable to PVNGS and are used within the conditions and limitations prescribed in the SE for each approved document. Sections 3.3.1 through 3.3.12, below, provide the NRC staff's analysis and conclusions regarding the applicability and acceptability of each methodology for its intended use at PVNGS, Units 1, 2, and 3.

3.3.1 WCAP-16500-P-A, Revision 0

WCAP-16500-P-A, Revision 0 (Reference 5), describes the NGF mechanical design for the CE Nuclear Steam Supply System (NSSS) and the methods and models used for evaluating its acceptability. Additionally, it describes fuel mechanical and reload design methodologies intended to support fuel design and licensing applications. Since WCAP-16500-P-A has been approved for use at CE NSSS plants such as PVNGS, the NRC staff finds the TR to be applicable to PVNGS, Units 1, 2, and 3.

The NRC staff's SE for WCAP-16500-P-A, Revision 0, contains ten conditions and limitations. Licensees referencing WCAP-16500-P-A are expected to ensure compliance with the ten conditions and limitations. APS has documented compliance with these ten conditions and limitations in its application (Reference 1). Since the NRC staff has concluded that (1) all conditions and limitations have been met, as described in the following subsections, and (2) that the TR is applicable to PVNGS, the NRC staff further concludes that the use of WCAP-16500-P-A methodologies for PVNGS, 1, 2, and 3, is acceptable. For completeness, each condition and limitation is restated below along with the NRC staff's evaluation of APS's response.

3.3.1.1 Condition and Limitation 1

Cycle Specific Design Criteria

Using approved methods, the licensee must ensure that all of the design criteria specified in TR WCAP-16500-P are satisfied on a cycle-specific basis (SE Section 3.3.1).

Safety Evaluation for WCAP-16500-P-A
Condition and Limitation 1

APS has indicated that fuel temperature and rod internal pressure criteria are confirmed on a reload basis as part of the approved reload process. The NRC staff observed that, in particular, APS has identified two design criteria associated with Section 3.3.1 of the NRC staff's SE for WCAP-16500-P, which need to be satisfied on a cycle-specific basis. All other design criteria from WCAP-16500-P-A must also be satisfied for PVNGS; however, the other design criteria were not specifically cited in the above condition and limitation as requiring verification on a cycle-specific basis. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.1.2 Condition and Limitation 2

Fuel Assembly and Component Design

Fuel assembly component design and configuration (e.g., type and distribution of spacer grids and IFM grids) are limited to the five designs described in TR WCAP-16500-P and in response to RAI [Request for Additional Information] 2 (SE Section 3.2).

Safety Evaluation for WCAP-16500-P-A
Condition and Limitation 2

The licensee stated that the NGF grids as described in WCAP-16500-P-A, Revision 0, and WCAP-16500-P-A, Supplement 2 comply with the condition. Therefore, this condition and limitation has been satisfied.

3.3.1.3 Condition and Limitation 3

Peak Rod Average Burnup

The reference fuel assembly design, CE 16×16 NGF, its fuel mechanical design methodology and design criteria, are approved up to a peak rod average burnup of 62 GWd/MTU [gigawatt days per metric ton of uranium]. A fuel burnup limit may exist, however, either explicitly or implicitly, in other portions of a plant's licensing basis. The NRC staff's approval of this topical report allows the CE 16×16 NGF assembly to reach a rod average burnup of 62 GWd/MTU. However, a licensing amendment request, specifically addressing each plant's licensing basis including radiological consequences, is required prior to extending burnup beyond current levels. Further, the NRC staff's SE for Optimized ZIRLO™ (Addendum 1 to TR WCAP-12610-P-A and TR CENPD-404- P-A) specified a 60 MWd/kgU [megawatt-days per kilogram of uranium] burnup limit and this limitation must be revised prior to extending the peak rod average burnup for the NGF design (SE Section 3.4).

Safety Evaluation for WCAP-16500-P-A
Condition and Limitation 3

APS has stated that calculations are performed up to 62 GWd/MTU, but the rod average burnup limit remains at the currently licensed value of 60 GWd/MTU[3] for NGF. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.1.4 Condition and Limitation 4

Growth Prediction Accuracy Demonstration

Licensees shall demonstrate the accuracy of their growth predictions based upon measured data and this validation shall be ahead of the burnups achieved by batch implementation. The growth model validation (e.g., measured versus predicted) should be documented in a letter(s) to the NRC (SE Section 3.2.1).

Safety Evaluation for WCAP-16500-P-A
Condition and Limitation 4

Westinghouse demonstrated the accuracy of growth predictions based upon measured data and provided this growth model validation to the NRC in Westinghouse document LTR-NRC-10-40 (Reference 16). Therefore, the NRC staff has concluded that this condition and limitation has been met.

[3] Note that 1 GWd/MTU = 1 MWd/kgU

3.3.1.5 Condition and Limitation 5

Interim Margin Penalty to Digital Setpoints

To compensate for NRC staff concerns related to the digital setpoints process, an interim margin penalty of 6 percent must be applied to the final addressable constants (e.g., BERR1*1.06, **[[ ]]** calculated following the 1/64 hypercube setpoints process.... Removal of this interim margin penalty will be considered after the digital setpoints methods have been formalized, documented (e.g., Revision to TR WCAP-16500-P), and approved by the NRC (SE Section 3.7).

Safety Evaluation for WCAP-16500-P-A
Condition and Limitation 5

The digital setpoints methods utilized for NGF introduction at PVNGS Units 1, 2 and 3 have been formalized and approved by the NRC in WCAP-16500-P-A, Supplement 1-A, Revision 1 (Reference 6). Therefore, a 6 percent interim margin penalty is not applied to the final addressable constants and the NRC staff has concluded that this condition and limitation is no longer applicable.

3.3.1.6 Condition and Limitation 6

Transition Core Analysis

Licensees are required to demonstrate that during transition cores, DNB [departure from nucleate boiling] margin gains associated with the NGF design offset (1) any impacts of flow starvation due to increased pressure drop and (2) uncertainty associated with predicting local flow characteristics. Further, licensees must detail the analytical methods and results of their transition core LOCA and non-LOCA analyses (SE Sections 3.7 and 3.10).

Safety Evaluation for WCAP-16500-P-A
Condition and Limitation 6

APS has acknowledged that NGF assemblies will likely experience some flow starvation due to flow diversion in the mixing vane grid region in transition cores. It has been demonstrated that thermal margin gains from the WSSV critical heat flux (CHF) correlation offset any flow penalty in that region. The thermal margin gains associated with the NGF design are realized through the CETOP/VIPRE-W benchmarking analysis performed in accordance with the methodology described in WCAP-16500-P-A, Revision 0. The licensee provided the analytical methods used to analyze transition cores in an attachment to its application and are discussed in subsequent sections of this SE. Based on this information and the review of the methods, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.1.7 Condition and Limitation 7

LOCA Analyses

Implementation of CE 16×16 NGF assemblies necessitate re-analysis of the plant-specific LOCA analyses. Licensees are required to submit a license amendment containing the revised LOCA analyses for NRC review. Upon approval, the revised LOCA analyses constitute the analysis-of-record and baseline for which future changes will be measured in accordance with 10 CFR 50.46(a)(3) (SE Sections 3.7).

Safety Evaluation for WCAP-16500-P-A
Condition and Limitation 7

The licensee submitted the results of the revised LOCA analyses supporting the transition to NGF via attachment to the submittal. These analyses are evaluated by the NRC staff below in Section 3.5.8 of this SE. Upon approval, these calculations will become AORs for PVNGS. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.1.8 Condition and Limitation 8

Peak Local Power

Using approved models and methods, Westinghouse will continue to limit peak local power experienced during Condition I and II events to ensure that fuel temperature remains below melting temperature at all burnups. This evaluation may be both plant and cycle-specific (SE Section 3.3.4).

Safety Evaluation for WCAP-16500-P-A
Condition and Limitation 8

APS has stated that calculations have been performed to ensure that fuel melt does not occur below the steady-state and transient peak local heat generation rate. Additionally, the limiting peak local power was evaluated to ensure the fuel melt acceptance criterion is met for transient events. The peak power limits are to be verified to be applicable for the reload cycle per the approved reload process. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied. The NRC staff's review of the analyses for Condition I and II events is contained below in Section 3.5.7 of this SE.

3.3.1.9 Condition and Limitation 9

Update COLR

The NRC staff's approval of TR WCAP-16500-P establishes the licensing basis for batch implementation of the CE 16×16 NGF assembly design. Licensees wishing to implement this fuel design are required to submit a license amendment request, where applicable, updating their Core Operating Limits Report list of methodologies with the "A" version of this TR.

Safety Evaluation for WCAP-16500-P-A
Condition and Limitation 9

Addition of WCAP-16500-P-A to TS 5.6.5.b and, consequently, to the COLR list of methodologies is requested as part of this LAR. Therefore, the NRC staff has concluded that this condition and limitation has been met.

3.3.1.10 Condition and Limitation 10

Appendix A LOCA Model

The NRC staff's review did not include the LOCA model changes [described] in Appendix A of TR WCAP-16500-P. Therefore, a licensee will have to submit a license amendment, if they desire to use the Appendix A LOCA model changes.

Safety Evaluation for WCAP-16500-P-A
Condition and Limitation 10

Changes to the LOCA model outlined in Appendix A were resubmitted to the NRC by Westinghouse under CENPD-132, Supplement 4-P-A, Addendum 1-P (Reference 17) and have been approved for use in LARs as described in WCAP-16523-P-A (Reference 13). Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.2 WCAP-16500-P-A, Supplement 1, Revision 1

WCAP-16500-P-A, Supplement 1, Revision 1 (Reference 6), describes a revised analytical process for calculating Core Operating Limits Supervisory System (COLSS) and Core Protection Calculator System (CPCS) addressable constants and database constants for plant reloads with NGF assemblies. Since WCAP-16500-P-A, Supplement 1, Revision 1, has been approved for use with NGF and describes the setpoints methodology for CE NSSS plants such as PVNGS, the NRC staff finds the TR to be applicable to PVNGS, Units 1, 2, and 3, with NGF.

The NRC staff's SE for WCAP-16500-P-A, Supplement 1, Revision 1, contains no conditions and limitations. Therefore, the NRC staff further concludes that the use of WCAP-16500-P-A, Supplement 1, Revision 1, methodologies for PVNGS, Units 1, 2, and 3, is acceptable.

3.3.3 WCAP-16500-P-A, Supplement 2

WCAP-16500-P-A, Supplement 2 (Reference 7), describes three evolutionary design changes to mid- and IFM- spacer grids that, in combination, improve grid to rod fretting (GTRF) performance and resistance to crud formation while improving fabricability of the grids. Since the changes detailed in WCAP-16500-P-A, Supplement 2, have been approved for the NGF design and do not present any new plant-specific challenges, the NRC staff finds the TR to be applicable to NGF at PVNGS, Units 1, 2, and 3.

The NRC staff's SE for WCAP-16500-P-A, Supplement 2, contains three conditions and limitations. Licensees referencing WCAP-16500-P-A, Supplement 2, are expected to ensure compliance with the three conditions and limitations. The licensee has documented compliance with these three conditions and limitations in its application (Reference 1). Since the NRC staff has concluded that (1) all conditions and limitations have been met, as described in the following subsections, and (2) that the TR is applicable to PVNGS, the NRC staff further concludes that the use of WCAP-16500-P-A, Supplement 2, for NGF at PVNGS, Units 1, 2, and 3, is acceptable. For completeness, each condition and limitation is restated below along with the NRC staff's evaluation of APS's response.

3.3.3.1 Condition and Limitation 1

Combined Grid Changes

CE 16×16 NGF spacer grids **[[**
]] must also apply the Modified Outer Strap (MOS) design change. Intermediate Flow-Mixing (IFM) grids for which the MOS design change is not applicable may **[[** **]]** since the **[[**
]] do not challenge any safety analyses or design criterion.

Safety Evaluation for WCAP-16500-P-A, Supplement 2
Condition and Limitation 1

APS has indicated that the manufacturing of grids complies with this condition and limitation. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.3.2 Condition and Limitation 2

Analysis of Grid Design Changes

Any changes or combinations of changes approved in this safety evaluation shall be analyzed and explicitly accounted for according to approved licensed methodologies prior to implementation.

Safety Evaluation for WCAP-16500-P-A, Supplement 2
Condition and Limitation 2

APS has stated that the combined mechanical design changes described in WCAP-16500-P-A, Supplement 2 have been explicitly accounted for in the analyses performed to support

implementation of NGF. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.3.3 Condition and Limitation 3

Peak Rod Average Burnup

Licensees may not reference the proposed approach to address IN 2012-09 detailed in Supplement 2 to WCAP-16500-P/WCAP-16500-NP submittal as this approach has not been reviewed or approved by the NRC staff.

Safety Evaluation for WCAP-16500-P-A, Supplement 2
Condition and Limitation 3

APS has stated that no credit was taken for the approach to address Information Notice (IN) 2012-09 (Reference 18), which details operating experience involving evaluations of fuel assembly structural response to external loads and associated issues the NRC staff identified during reviews of fuel designs for design certification applications, outlined in WCAP-16500-P, Supplement 2. APS originally proposed a license condition to ensure IN 2012-09 is addressed in the future following the availability of an NRC-approved approach. However, as discussed below in Section 3.6, the NRC staff determined that a license condition is not necessary to address this issue because existing regulatory requirements require a plant shutdown following a seismic event with vibratory ground motion exceeding that of the operating basis earthquake, as per 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." Prior to resuming operation, the licensee is further required to demonstrate that no functional damage has occurred that would result in undue risk to the public health and safety. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.4 WCAP-14565-P-A, Revision 0

WCAP-14565-P-A, Revision 0 (Reference 11) describes Westinghouse VIPRE (VIPRE-W) modeling and qualification for PWR core thermal hydraulic design and non-LOCA safety analysis. Since WCAP-14565-P-A, Revision 0, has been approved by NRC for PWR licensing applications, the NRC staff finds the TR to be applicable to PVNGS, Units 1, 2, and 3.

The NRC staff's SE for WCAP-14565-P-A, Revision 0, contains four conditions and limitations. Licensees referencing WCAP-14565-P-A, Revision 0, are expected to ensure compliance with the four conditions and limitations. The licensee has documented compliance with these four conditions and limitations in its application (Reference 1). Since the NRC staff has concluded that (1) all conditions and limitations have been met, as described in the following subsections, and (2) that the TR is applicable to PVNGS, the NRC staff further concludes that the use of WCAP-14565-P-A, Revision 0, methodologies for PVNGS, Units 1, 2, and 3 is acceptable. For completeness, each condition and limitation is restated below along with the NRC staff's evaluation of APS's response.

3.3.4.1 Condition and Limitation 1

Justification for Fuel-Dependent Parameters

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

Safety Evaluation for WCAP-14565-P-A, Revision 0
Condition and Limitation 1

APS has supplied justification for the use of each CHF correlation and the corresponding DNBR safety limit, the engineered hot channel factors for enthalpy rise, and other fuel-dependent parameters as follows.

The WSSV CHF correlation with a 95/95 correlation DNBR safety limit of 1.12 approved with VIPRE-W in WCAP-16523-P-A was used in the DNBR calculations for the mixing vane grid regions of the NGF design.

The ABB-Non-Vane (ABB-NV) CHF correlation with a 95/95 correlation limit of 1.13 approved with VIPRE-W in WCAP-14565-P-A, Addendum 1-A (Reference 19), and Addendum 2-P-A (Reference 20) was used in the DNBR calculations for the non-mixing vane grid regions for the NGF design, as well as for the STD fuel design, in place of the currently licensed CE-1 CHF correlation.

The Westinghouse Low Pressure (WLOP) CHF correlation can be used as an alternative to the Macbeth CHF correlation for the analysis of both STD and NGF in PVNGS when the primary CHF correlation is not applicable (i.e., during low-pressure conditions). In WCAP-14565-P-A, Addendum 2-P-A, the WLOP 95/95 correlation limit of 1.18 was approved for use with VIPRE-W.

The correlation limits used in the NGF safety analysis DNBR calculations for the loaded fuel types in PVNGS are consistent with the approved values in WCAP-16523-P-A for the WSSV CHF correlation and WCAP-14565-P-A, Addendum 2-P-A, for the ABB-NV and WLOP CHF correlations. The engineered hot channel factors and other fuel-dependent parameters in the PVNGS DNBR calculations for the STD and NGF designs were justified as part of the statistical combination of uncertainties (SCU) DNBR safety limit calculations.

The NRC staff has concluded that each CHF correlation, along with its corresponding DNBR correlation limit, has been previously approved and is applied as intended. Likewise, the engineered hot channel factors for enthalpy rise and other fuel-dependent parameters are determined using an approved approach found to be applicable to PVNGS for DNBR calculations. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.4.2 Condition and Limitation 2

Boundary Condition Justification

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.

Safety Evaluation for WCAP-14565-P-A, Revision 0
Condition and Limitation 2

The core boundary conditions used in the VIPRE-W DNBR calculations are all generated from NRC-approved codes and analysis methodologies. Continued applicability of the core boundary conditions as VIPRE-W input is verified on a cycle-by-cycle basis using the CE NSSS reload methodology. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.4.3 Condition and Limitation 3

Use of Other CHF Correlations

The NRC staff's generic SER [safety evaluation report] 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 [safety] limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR [safety] limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification.

Safety Evaluation for WCAP-14565-P-A, Revision 0
Condition and Limitation 3

As discussed in response to Condition 1, the WSSV CHF correlation with a 95/95 correlation limit of 1.12 was approved in WCAP-16523-P-A for use in VIPRE-W. The ABB-NV CHF correlation with a 95/95 correlation limit of 1.13 and the WLOP CHF correlation with a 95/95 correlation limit of 1.18 were approved in WCAP-14565-P-A, Addendum 2-P-A, for use in VIPRE-W. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.4.4 Condition and Limitation 4

Post CHF Conservatism

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.

Safety Evaluation for WCAP-14565-P-A, Revision 0
Condition and Limitation 4

For the PVNGS NGF safety analysis, the licensee stated that the VIPRE-W code does not model the time-dependent physical changes that may occur within fuel rods at elevated temperatures in the post-CHF region. The NRC staff also observed that no conservative inputs were proposed by the licensee to account for post-CHF effects. Therefore, because the licensee has not used the VIPRE-W code in the post-CHF regime, the NRC staff concludes that this condition and limitation has been met. Furthermore, as specified in the SE on WCAP-14565-P-A, Revision 0, if in the future the licensee uses VIPRE-W in the post-CHF regime, appropriate justification must be submitted to the NRC staff for each such usage to ensure that conservative results are obtained.

3.3.5 WCAP-14565-P-A, Addendum 1-A

WCAP-14565-P-A, Addendum 1-A (Reference 19), installed the ABB-NV and ABB-TV CHF correlations into VIPRE-W. The ABB-NV correlation is used for CE 14×14 and 16×16 fuels with non-mixing vane grids. The ABB-TV correlation is used for 14×14 Turbo fuel with mixing vane grids and is therefore not used in any PVNGS analysis. Since WCAP-14565-P-A, Addendum 1-A, has been approved, and VIPRE-W has been found applicable to PVNGS, the NRC staff finds WCAP-14565-P-A, Addendum 1-A, applicable to PVNGS, Units 1, 2, and 3.

The NRC staff's SE for WCAP-14565-P-A, Addendum 1-A, contains a requirement specified in Section 3.1 of the SE and three conditions and limitations. Licensees referencing WCAP-14565-P-A, Addendum 1-A, are expected to ensure compliance with the requirement and the three conditions and limitations. The licensee has documented compliance with the requirement and these three conditions and limitations in its application (Reference 1). Since the NRC staff has concluded that (1) all requirements and conditions and limitations have been met, as described in the following subsections, and (2) that the TR is applicable to PVNGS, the NRC staff further concludes that the use of WCAP-14565-P-A, Addendum 1-A, methodologies for PVNGS, Units 1, 2, and 3 is acceptable. For completeness, the SE requirement and each condition and limitation is restated below along with the NRC staff's evaluation of APS's response.

3.3.5.1 SE Section 3.1 Requirement

Continued Adherence to CENPD-387-P Conditions

Westinghouse will apply the VIPRE-01 code with the ABB CHF correlations under the following conditions consistent with the requirements in the CENPD-387-P safety evaluation:

1. The 95/95 DNBR limits of the ABB-NV and ABB-TV correlations are not lower than the current NRC-approved limit of 1.13 for the CE-PWR fuels.

2. The ABB-NV and ABB-TV correlations are used with the VIPRE-01 code, in addition to the TORC and CETOP-D codes currently used for CE-PWRs. This addendum demonstrates the VIPRE-01 equivalency to TORC for DNBR calculations.

3. The ABB-NV and ABB-TV correlations are used with the optimized F_c [non-uniform axial] shape factor to account for effects of non-uniform axial power shapes.

4. The current range of applicability for the ABB-NV and ABB-TV correlations as shown in Table 2-1 of the TR remains applicable.

5. The ABB-NV and ABB-TV correlations are used only for the CE-PWR fuel designs with NRC-approved methodology for PWR safety analysis.

6. Technology transfer is accomplished through a process that meets the guidance of [Generic Letter] GL 83-11, Supplement 1.

Safety Evaluation for WCAP-14565-P-A, Addendum 1-A
SE Section 3.1 Requirement

Each item in the numbered list corresponds to a condition and limitation in CENPD-387-P-A (Reference 12), which is discussed further in Section 3.3.7 of this SE. The licensee stated the VIPRE-W modeling used for this analysis meets these requirements. These items have been evaluated and determined to be met for PVNGS and are discussed in more detail below. APS has confirmed that they have been met for ABB-NV correlation as well. Therefore, the NRC staff has concluded that this requirement has been satisfied.

3.3.5.2 Condition and Limitation 1

Consistency with WCAP-14565-P-A

Addendum 1 to the WCAP-14565-P-A VIPRE-01 model must remain consistent with that for the DNB data analysis described in WCAP-14565-P-A VIPRE-01.

Safety Evaluation for WCAP-14565-P-A, Addendum 1-A
Condition and Limitation 1

APS has confirmed that the VIPRE-W[4] model was used as described in WCAP-14565-P-A. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.5.3 Condition and Limitation 2

DNBR Safety Limit

The current 95/95 DNBR limit of 1.13 remains unchanged.

Safety Evaluation for WCAP-14565-P-A, Addendum 1-A
Condition and Limitation 2

APS confirmed that the ABB-NV 95/95 DNBR safety limit of 1.13 remains unchanged for the thermal-hydraulic analysis performed for PVNGS. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.5.4 Condition and Limitation 3

Range of Applicability

DNBR calculations for CE-PWR fuels are within the current applicable range defined in Table 2-1 of the TR.

Safety Evaluation for WCAP-14565-P-A, Addendum 1-A
Condition and Limitation 3

Table 2-1 of the TR (or recreated as Table 2-1 in the licensee's application dated July 1, 2016) contains ranges of applicability for the ABB-NV and ABB-TV correlations. As mentioned above, the ABB-TV correlation is not applicable to NGF and is not used in this application. Additionally, the range of applicability listed for the ABB-NV correlation has been updated and superseded in WCAP-14565-P-A, Addendum 2-A. Therefore, the NRC staff has concluded that this condition and limitation is no longer applicable and can be considered met.

[4] As previously noted, the name "VIPRE-W" has been used to distinguish the Westinghouse version of the VIPRE-01 code described in WCAP-14565-P-A from other organizations' versions of this code. Hence, the quoted conditions and limitations from past NRC staff SEs referring to VIPRE-01 in this context directly apply to VIPRE-W.

3.3.6 WCAP-14565-P-A, Addendum 2-P-A

WCAP-14565-P-A, Addendum 2-P-A (Reference 20), describes a modification to the ABB-NV CHF correlation based on rod bundle data at low-pressure and low-flow conditions. This modified low-pressure ABB-NV correlation is designated as the WLOP correlation in this report. The WLOP correlation predicts DNBR more accurately than either W-3 or Macbeth correlation at low pressure and low flow conditions. Additionally, the WLOP correlation is based on non-vaned grid regions. Therefore, it predicts a conservatively low DNBR for grid regions that contain mixing vanes. Since WCAP-14565-P-A, Addendum 2-P-A, has been approved for use in VIPRE-W and VIPRE-W itself has been found applicable to PVNGS, the NRC staff finds WCAP-14565-P-A, Addendum 2-P-A, to be applicable to PVNGS, Units 1, 2, and 3.

The NRC staff's SE for WCAP-14565-P-A, Addendum 2-P-A, contains four conditions and limitations. Licensees referencing WCAP-14565-P-A, Addendum 2-P-A, are expected to ensure compliance with the four conditions and limitations. The licensee has documented compliance with these four conditions and limitations in its application (Reference 1). Since the NRC staff has concluded that (1) all conditions and limitations have been met, as described in the following subsections, and (2) that the TR is applicable to PVNGS, the NRC staff further concludes that the use of WCAP-14565-P-A, Addendum 2-P-A, methodologies for PVNGS, Units 1, 2, and 3 is acceptable. For completeness, each condition and limitation is restated below along with the NRC staff's evaluation of APS's response.

3.3.6.1 Condition and Limitation 1

Ranges of Applicability

The applicable range[s] of the ABB-NV and WLOP correlations are presented in Table 1 and Table 2, respectively, of this SE.

Safety Evaluation for WCAP-14565-P-A, Addendum 2-P-A
Condition and Limitation 1

Table 1 of the TR SE (or recreated as Table 1 in the licensee's application dated July 1, 2016) lists the updated range of applicability for the ABB-NV correlation, which supersedes the range mentioned in Section 3.3.5.4 of this SE. Similarly, Table 2 of the TR SE (or recreated as Table 2 in the licensee's application dated July 1, 2016) lists the range of applicability for the WLOP correlation. Ranges of applicability included in these tables are for the following parameters: pressure, local mass velocity, local quality, heated length (inlet to CHF location), heated hydraulic diameter ratio, and grid distance.

APS has stated that for the NGF safety analysis, DNB calculations using the ABB-NV and WLOP correlations were confirmed to be in full compliance with the parameter ranges of the CHF correlations as specified in Table 1 and Table 2, respectively. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.6.2 Condition and Limitation 2

Power-Shape Correction Factor

The ABB-NV correlation and the WLOP correlation must use the same F_c factor for power shape correction as used in the primary DNB correlation for a specific fuel design.

Safety Evaluation for WCAP-14565-P-A, Addendum 2-P-A
Condition and Limitation 2

APS has stated that for the NGF safety analysis, DNB calculations with the ABB-NV and WLOP correlations used the same F_c factor for power-shape correction that was applied for the WSSV correlation, which is the primary CHF correlation for the NGF design at PVNGS. The same optimized F_c factor was also applied to the ABB-NV CHF correlation for STD fuel applications. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.6.3 Condition and Limitation 3

Justification for Fuel-Dependent Parameters

Selection of the appropriate DNB correlation, DNBR safety limit, engineering hot channel factors for enthalpy rise, and other fuel-dependent parameters will be justified for each application of each correlation on a plant specific basis.

Safety Evaluation for WCAP-14565-P-A, Addendum 2-P-A
Condition and Limitation 3

This condition and limitation is essentially identical to Condition and Limitation 1 in WCAP-14565, Revision 0, and has been addressed above in Section 3.3.4.1 of this SE. In that section, the NRC staff concluded that this condition and limitation has been satisfied. Therefore, the NRC staff likewise concludes that this condition and limitation continues to be met for WCAP-14565-P-A, Addendum 2-P-A.

3.3.6.4 Condition and Limitation 4

Correlation Implementation

The ABB-NV correlation for Westinghouse PWR applications and the WLOP correlation must be used in conjunction with the Westinghouse version of the VIPRE-01 (VIPRE) code since the correlations were justified and developed based on VIPRE and the associated VIPRE modeling specifications.

Safety Evaluation for WCAP-14565-P-A, Addendum 2-P-A
Condition and Limitation 4

APS has confirmed that the Westinghouse version of the VIPRE-01 code (i.e., VIPRE-W) was implemented for all DNB analyses of the PVNGS fuel types. Therefore, the NRC staff has concluded that this condition and limitation has been met.

3.3.7 CENPD-387-P-A

CENPD-387-P-A (Reference 12), describes the PWR CHF correlations for ABB CE PWR 14×14 and 16×16 fuels. The ABB-NV correlation is for ABB-CE PWR 14×14 and 16×16 fuels with non-mixing vane grids. The ABB-TV correlation is for the 14×14 Turbo fuel with mixing vane grids, and is therefore not being used at PVNGS. Since CENPD-387-P-A has been approved and has been adopted by WCAP-14565-P-A and its related addenda, the NRC staff finds the TR to be applicable to PVNGS with NGF, when used as described in accordance with WCAP-14565-P-A and the related amending methodologies.

The NRC staff's SE for CENPD-387-P-A contains six conditions and limitations. Licensees referencing CENPD-387-P-A are expected to ensure compliance with the six conditions and limitations. The licensee has documented compliance with these six conditions and limitations in its application (Reference 1). Since the NRC staff has concluded that (1) all conditions and limitations have been met, as described in the following subsections, and (2) that the TR is applicable to PVNGS, the NRC staff further concludes that the use of CENPD-387-P-A methodologies for NGF at PVNGS is acceptable, when used as described in accordance with WCAP-14565-P-A and the related amending methodologies. For completeness, each condition and limitation is restated below along with the NRC staff's evaluation of APS's response.

3.3.7.1 Condition and Limitation

DNBR Safety Limit

The ABB-NV and ABB-TV correlations indicate a minimum DNBR limit of 1.13 will provide a 95 percent probability with 95 percent confidence of not experiencing CHF on a rod showing the limiting value.

Safety Evaluation for CENPD-387-P-A
Condition and Limitation 1

As discussed in Section 3.3.5.3 of this SE, APS has confirmed that the approved ABB-NV 95/95 DNBR safety limit of 1.13 remains unchanged. APS has also confirmed that the ABB-TV correlation will not be used at PVNGS in implementing the transition to NGF. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.7.2 Condition and Limitation 2

TORC and CETOP-D Implementation

The ABB-NV and ABB-TV correlations must be used in conjunction with the TORC code since the correlations were developed on the basis of the TORC [code] and the associated TORC input specifications. The correlations may also be used in the CETOP-D code in support of reload design calculations.

Safety Evaluation for CENPD-387-P-A
Condition and Limitation 2

APS has confirmed that the ABB-NV CHF correlation will be used with the TORC and CETOP-D codes in support of reload design calculations. Ultimately, for licensing applications, the CETOP-D code is used in place of the legacy TORC code. Additionally, the use of the ABB-NV CHF correlation in conjunction with VIPRE-W in support of reload design calculations has been addressed in Section 3.3.5 of this SE. Therefore, the NRC staff concludes that this condition and limitation has been satisfied.

3.3.7.3 Condition and Limitation 3

Power-Shape Correction Factor

The ABB-NV and ABB-TV correlations must also be used with the ABB-CE optimized F_c, shape factor to correct for non-uniform axial power shapes.

Safety Evaluation for CENPD-387-P-A
Condition and Limitation 3

APS has stated that for the DNB safety analyses using the ABB-NV CHF correlation, the optimized Tong F_c factor for power-shape correction was applied to DNBR predictions for non-mixing vane grid regions. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.7.4 Condition and Limitation 4

Range of Applicability

This condition and limitation contains a table that specifies the range of applicability for the ABB-NV and ABB-TV correlations.

Safety Evaluation for CENPD-387-P-A
Condition and Limitation 4

The table contained in the TR condition and limitation is similar to those contained in the conditions and limitations for WCAP-14565-P-A and its addenda. Ranges of applicability for ABB-NV (and ABB-TV) are given for pressure, local mass velocity, local quality, heated length (inlet to CHF location), grid spacing, and heater hydraulic diameter ratio.

APS has indicated that the results of DNBR calculations using the ABB-NV CHF correlation were confirmed to comply with the parameter ranges of the CHF correlations as specified in the table. The DNBR calculations using the ABB-NV CHF correlation with the VIPRE-W code were performed in accordance with the limitations and conditions described in the SE to WCAP-14565-P-A, Addendum 2-P-A, which provides updated parameters for the same correlation. Therefore, the NRC staff has concluded that this condition and limitation has been met.

3.3.7.5 Condition and Limitation 5

ABB Correlation Implementation

The ABB-NV and ABB-TV correlation[s] will be implemented in the reload analysis in the exact manner described in Section 7.1 of Topical Report CENPD-387-P, Revision 00-P.

Safety Evaluation for CENPD-387-P-A
Condition and Limitation 5

APS has indicated that the ABB-NV correlation is applied according to Section 7.1 of CENPD-387-P-A for non-mixing vane grid spans for CE 16×16 STD and NGF assemblies. The WSSV and WSSV-T correlations are applied for the mixing vane grid spans of the NGF as described in Sections 6.1 and 6.2 of WCAP-16523-P-A, respectively, instead of the ABB-TV correlation. Therefore, the NRC staff has concluded that this condition and limitation has been met.

3.3.7.6 Condition and Limitation 6

Technology Transfer

Technology transfer will be accomplished only through the process described in Reference 5 [Letter from Ivan Rickard, ABB-CE, to NRC Document Control Desk, dated February 23, 2000], which includes ABB-CE performing an independent benchmarking calculation for comparison to the licensee generated results to verify that the new CHF correlations are properly applied for the first application by the licensee.

Safety Evaluation for CENPD-387-P-A
Condition and Limitation 6

APS has successfully participated in the CE Reload Technology Transfer Program, including independent reload core design and verification calculations, as addressed in the NRC-approved PVNGS Reload Analysis Methodology Report. Therefore, the NRC staff has concluded that this condition and limitation has been met.

3.3.8 WCAP-16523-P-A

WCAP-16523-P-A (Reference 13), describes the development of PWR CHF correlations for designs containing structural mixing vane grids and IFM grids with side-supported vanes. The WSSV and WSSV-T correlations are developed for use with VIPRE-W and TORC, respectively, and are used to model 14×14 and 16×16 fuel designs with side-supported vanes for CE PWRs. Since WCAP-16523-P-A has been approved for use in VIPRE-W and TORC, and these correlations have been developed for PWRs with CE 16×16 fuel such as PVNGS, the NRC staff finds WCAP-16523-P-A to be applicable to PVNGS, Units 1, 2, and 3.

The NRC staff's SE for WCAP-16523-P-A contains four conditions and limitations. Licensees referencing WCAP-16523-P-A are expected to ensure compliance with the four conditions and limitations. The licensee has documented compliance with these four conditions and limitations in its application (Reference 1). Since the NRC staff has concluded that (1) all conditions and limitations have been met, as described in the following subsections, and (2) that the TR is applicable to PVNGS, the NRC staff further concludes that the use of WCAP-16523-P-A methodologies for PVNGS is acceptable. For completeness, each condition and limitation is restated below along with the NRC staff's evaluation of APS's response.

3.3.8.1 Condition and Limitation 1

WSSV Use with VIPRE

The WSSV correlation must be used in conjunction with the VIPRE code since the correlation was developed based on VIPRE and the associated VIPRE input specifications. Other uses of the WSSV correlation should reference this TR and be based on appropriate benchmarking with VIPRE.

Safety Evaluation for WCAP-16523-P-A
Condition and Limitation 1

APS has indicated that the WSSV CHF correlation with a 95/95 correlation limit of 1.12 (approved in WCAP-16523-P-A) was used in the VIPRE-W DNBR calculations for the mixing vane grid regions of the NGF design. The WSSV correlation was installed into the CETOP-D code and was benchmarked against the VIPRE-W code for NGF safety analysis applications. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.8.2 Condition and Limitation 2

WSSV Use with TORC

The WSSV-T correlation must be used in conjunction with the TORC code since the correlation was developed based on TORC and the associated TORC input specifications. The correlations may also be used in the CETOP-D code in support of reload design calculations benchmarked by TORC.

Safety Evaluation for WCAP-16523-P-A
Condition and Limitation 2

APS has specified that the WSSV-T CHF correlation will only be used with the TORC and CETOP-D codes in support of reload design calculations. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.8.3 Condition and Limitation 3

Power-Shape Correction Factor

The WSSV and WSSV-T correlations must also be used with the optimized Tong F_c shape factor for non-mixing vane and side-supported mixing vane grids to correct for non-uniform axial power shapes.

Safety Evaluation for WCAP-16523-P-A
Condition and Limitation 3

APS has confirmed that for the NGF safety analysis, DNB calculations using the WSSV CHF correlation applied the optimized Tong F_c factor for power-shape correction to the mixing vane grid region DNBR predictions. The optimized Tong F_c factor for power-shape correction was also applied in conjunction with the ABB-NV CHF correlation for non-mixing vane grid region DNBR predictions. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.8.4 Condition and Limitation 4

Range of Applicability

The range of applicability for both the WSSV and WSSV-T correlations [is]:

Parameter	Units	Range
Pressure	psia	1,495 to 2,450
Local coolant quality	--	≤ 0.34
Local mass velocity	10^6 lbm/hr-ft^2	0.90 to 3.46
Matrix heated hydraulic diameter, Dhm	inches	0.4635 to 0.5334
Heated hydraulic diameter ratio, Dhm/Dh	--	0.679 to 1.00
Heated length, HL	inches	48* to 150
Grid spacing	inches	10.28 to 18.86
* Set as minimum HL value, applied at all elevations below 48 inches		

Safety Evaluation for WCAP-16523-P-A
Condition and Limitation 4

APS has indicated that the results of DNBR calculations using the WSSV CHF correlation were confirmed to comply with the parameter ranges of the CHF correlations as specified in the table above. APS also stated that any future DNB analyses using the WSSV-T CHF correlation would also confirm compliance with the NRC-approved parameter ranges. Therefore, the NRC staff has concluded that this condition and limitation has been met.

3.3.9 WCAP-16072-P-A

WCAP-16072-P-A (Reference 14), describes the use of ZrB_2 IFBA in CE 14×14 and 16×16 fuel assembly designs. Since WCAP-16072-P-A has been approved for use with CE fuel assembly designs such as NGF, the NRC staff finds WCAP-16072-P-A to be applicable to PVNGS, Units 1, 2, and 3.

The NRC staff's SE for WCAP-16072-P-A contains five conditions and limitations. Licensees referencing WCAP-16072-P-A are expected to ensure compliance with the five conditions and limitations. The licensee has documented compliance with these five conditions and limitations in its application (Reference 1). Since the NRC staff has concluded that (1) all conditions and limitations have been met, as described in the following subsections, and (2) that the TR is applicable to PVNGS, the NRC staff further concludes that the use of WCAP-16072-P-A methodologies for PVNGS is acceptable. For completeness, each condition and limitation is restated below along with the NRC staff's evaluation of APS's response.

3.3.9.1 Condition and Limitation 1

COLR Analytical Methods

A license amendment is required to add this TR to the Core Operating Limits Report analytical methods listed in the licensee's TS.

Safety Evaluation for WCAP-16072-P-A
Condition and Limitation 1

As part of this LAR, WCAP-16072-P-A is to be added to TS 5.6.5.b, the COLR analytical methods list. Therefore, this condition and limitation has been satisfied.

3.3.9.2 Condition and Limitation 2

Axial Cutback Region Power Margins

Plant-specific core design guidelines or cycle-specific calculations shall be used to verify that required power margins in the axial cutback regions are maintained within safety analysis limitations.

Safety Evaluation for WCAP-16072-P-A
Condition and Limitation 2

APS has specified that the required power margins in the axial cutback regions are maintained on a cycle-specific basis per the approved reload process. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.9.3 Condition and Limitation 3

Moderator Temperature Coefficient (MTC)

Plant TS SRs on MTC validate the physics predictions and ensure that plant operations remain within allowable limits. In addition to current SRs, licensees shall confirm that the peak positive HFP [hot, full-power] MTC is within the TS limits at the highest RCS [reactor coolant system] soluble boron concentration predicted during full power operation. The peak positive HFP MTC shall be derived by adjusting the measured MTC at HFP BOC [beginning-of-cycle] conditions to the maximum HFP soluble boron concentration expected during the cycle. In order to ensure a conservative adjustment, a direct measurement of MTC is required at the highest RCS soluble boron concentration predicted during full power operation. This direct measurement is only required for the first application of ZrB_2 IFBA in a CE 14×14 or 16×16 fuel assembly design. During the first cycle implementation, Westinghouse shall provide the staff with a letter containing the following information:

i. Measured HFP BOC MTC (TS SR)

ii. Measured HFP MTC at highest RCS soluble boron concentration

iii. Calculated HFP MTC at highest RCS soluble boron concentration

iv. Demonstrated accuracy of the calculated HFP MTC within current analytical uncertainties.

In addition, plant procedures used to perform MTC surveillances shall be updated, where appropriate, to reflect the calculated peak positive HFP MTC along with ZrB_2 IFBA's distinctive trend in RCS critical boron concentration.

Safety Evaluation for WCAP-16072-P-A
Condition and Limitation 3

This is not the first application of ZrB_2 in a CE 16×16 fuel design. Therefore, this measurement is not required to be performed. APS has indicated that it will update the PVNGS plant procedures used to perform MTC surveillances where appropriate. Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.9.4 Condition and Limitation 4

Hydriding During Condition III or IV Events

Prior to startup following a Condition III [infrequent event] or IV [limiting fault] event, licensees must evaluate clad hydriding to ensure that hydrides have not precipitated in the radial direction (in accordance with Section 3.2 of this SE).

Safety Evaluation for WCAP-16072-P-A
Condition and Limitation 4

APS has indicated that hydriding will be evaluated by the appropriate functional group should a Condition III or IV event occur. Therefore, the NRC staff has concluded that this condition and limitation has been met.

3.3.9.5 Condition and Limitation 5

Fuel Cladding Burst

CEN-372-P-A constraints and limitations with regard to rod internal pressure and DNB propagation must continue to be met. In addition, licensees must ensure that the following two conditions are satisfied:

a. For Condition I (normal), Condition II (moderate frequency), and Condition III (infrequent) events, fuel cladding burst must be precluded for ZrB_2 IFBA fuel rods. Using models and methods approved for CE fuel designs, licensees must demonstrate that the total calculated stress remains below cladding burst stress at the cladding temperatures experienced during any potential Condition II or Condition III event. Within the confines of the plant's licensing basis, licensees must evaluate all Condition II events in combination with any credible, single active failure to ensure that fuel rod burst is precluded.

b. For Condition IV non-LOCA events which predict clad burst, the potential impacts of fuel rod ballooning and bursting need to be specifically addressed with regard to coolable geometry, RCS pressure, and radiological source term.

Safety Evaluation for WCAP-16072-P-A
Condition and Limitation 5

In response to Part a. of the above condition, APS has indicated that rod internal pressure is verified for Condition I, II, and III events as part of the approved reload process. The PVNGS licensing basis allows a limited amount of fuel failure due to DNB for selected Condition III events. For these events, DNB propagation (i.e., ballooning) was evaluated and shown to be precluded by the analyses discussed in Attachments 7 and 8 to the Enclosure, Section 7.1.3 (Reference 1). Therefore, the NRC staff concludes that the condition in Part a. has been met.

In response to Part b. of the above condition, APS has indicated that DNB propagation is addressed by the analyses discussed in Section 7.4.8 of Attachments 7 and 8 of the Enclosure to the licensee's submittal (Reference 1). Burst is addressed by the analyses discussed in Section 2.4 of Attachments 7 and 8 of the same submittal. In its application, the licensee stated that sufficient DNB thermal margin is reserved by TS LCOs to prevent all Updated Final Safety Analysis Report (UFSAR) Chapter 15 design-basis non-LOCA events from violating the DNB SAFDL, with the exception of the Sheared Shaft/Seized Rotor, Pre-Trip Steam Line Break,[5] and CEA Ejection events. In response to NRC RAI 8 (Reference 2), the licensee amended this list further to include the Inadvertent Loading of a Fuel Assembly into the Improper Position event. Each of these events is classified as a Condition III or IV event, for which DNB exceedance is permissible. An evaluation was performed to confirm that DNB propagation acceptance criteria are satisfied for each of these events. Therefore, the NRC staff concludes that the condition in Part b. has been met, and in combination with meeting condition Part a., the condition and limitation has been met as a whole.

3.3.10 CENPD-404-P-A, Addendum 1-A

The NRC staff notes that CENPD-404-P-A is already included in TS 5.6.5.b and the COLR list of analytical methodologies. Its applicability and acceptability for PVNGS has already been approved and, therefore, does not need to be addressed in this SE. However, implementation of NGF requires the use of CENPD-404-P-A, Addendum 1-A, and Addendum 2-A, which are to be added to the PVNGS COLR list of analytical methodologies and do require review in this SE.

CENPD-404-P-A, Addendum 1-A (Reference 8), describes an extension of the regulatory definition of ZIRLO™ to allow for the optimization of ZIRLO™ for enhanced corrosion resistance in more adverse in-reactor primary chemistry environments and at higher fuel duties with higher burnups. Since CENPD-404-P-A, Addendum 1-A, has been previously approved and the related cladding material wording change to TS 4.2.1 and the supporting cladding material exemption for PVNGS is being approved concurrently in a separate SE, the NRC staff finds CENPD-404-P-A, Addendum 1-A, to be applicable to PVNGS, Units 1, 2, and 3.

The NRC staff's SE for CENPD-404-P-A, Addendum 1-A, contains ten conditions and limitations. Licensees referencing CENPD-404-P-A, Addendum 1-A, are expected to ensure compliance with the ten conditions and limitations. The licensee has documented compliance with these ten conditions and limitations in its application (Reference 1). Since the NRC staff has concluded that (1) all conditions and limitations have been met, as described in the following subsections, and (2) that the TR is applicable to PVNGS, the NRC staff further concludes that the use of CENPD-404-P-A, Addendum 1-A, methodologies for PVNGS, Units 1, 2, and 3 is acceptable. For completeness, each condition and limitation is restated below along with the NRC staff's evaluation of APS's response. These evaluations are identical to those made as part of the related cladding material exemption SE.

[5] "Pre-trip" terminology denotes that analysis of the steam line break event may be split into two discrete phases, pre-trip and post-trip.

3.3.10.1 Condition and Limitation 1

Exemption

Until rulemaking to 10 CFR Part 50 addressing Optimized ZIRLO™ has been completed, implementation of Optimized ZIRLO™ fuel clad requires an exemption from 10 CFR 50.46 and 10 CFR Part 50 Appendix K.

Safety Evaluation for CENPD-404-P-A, Addendum 1-A
Condition and Limitation 1

APS concurrently submitted an exemption request for PVNGS, Units 1, 2, and 3, which is addressed in a separate SE (Reference 3). Therefore, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.10.2 Condition and Limitation 2

Burnup Limit

The fuel rod burnup limit for this approval remains at currently established limits: 62 GWd/MTU for Westinghouse fuel designs and 60 GWd/MTU for CE fuel designs.

Safety Evaluation for CENPD-404-P-A, Addendum 1-A
Condition and Limitation 2

Because PVNGS uses a CE fuel design (and not a Westinghouse fuel design) and APS has confirmed that PVNGS will continue to use a 60 GWd/MTU rod burnup, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.10.3 Condition and Limitation 3

Corrosion Limit

The maximum fuel rod waterside corrosion, as predicted by the best-estimate model, will **[[**

]] of hydrides for all locations of the fuel rod.

Safety Evaluation for CENPD-404-P-A, Addendum 1-A
Condition and Limitation 3

This condition and limitation has been superseded by Condition and Limitation 1 of Addendum 2-A to CENPD-404-P-A. Therefore, the NRC staff has concluded that this condition and limitation is no longer applicable.

3.3.10.4 Condition and Limitation 4

Conditions on Approved Methodologies

All the conditions listed in previous NRC SE approvals for methodologies used for standard ZIRLO™ and Zircaloy-4 fuel analysis will continue to be met, except that the use of Optimized ZIRLO™ cladding in addition to standard ZIRLO™ and Zircaloy-4 cladding is now approved.

Safety Evaluation for CENPD-404-P-A, Addendum 1-A
Condition and Limitation 4

Because APS has confirmed that future analysis using Optimized ZIRLO™ will continue to meet all conditions associated with approved methods, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.10.5 Condition and Limitation 5

Application Domain

All methodologies will be used only within the range for which ZIRLO™ and Optimized ZIRLO™ data were acceptable and for which the verifications discussed in Addendum 1 and responses to RAIs were performed.

Safety Evaluation for CENPD-404-P-A, Addendum 1-A
Condition and Limitation 5

Because APS has confirmed that the application of Optimized ZIRLO™ will be consistent with the approach accepted in CENPD-404-P-A, Addendum 1-A, and that confirmation of these conditions is part of the normal reload design process, the NRC staff has concluded that this condition and limitation has been satisfied.

3.3.10.6 Condition and Limitation 6

Lead Test Assemblies (LTA) Data

The licensee is required to ensure that Westinghouse has fulfilled the following commitment: Westinghouse shall provide the NRC staff with a letter(s) containing the following information (Based on the schedule described in response to RAI #3...):

a. Optimized ZIRLO™ LTA data from Byron, Calvert Cliffs, Catawba, and Millstone.

   i. Visual
   ii. Oxidation of fuel rods
   iii. Profilometry
   iv. Fuel rod length
   v. Fuel assembly length

b. Using the standard and Optimized ZIRLO™ database including the most recent LTA data, confirm applicability with currently approved fuel performance models (e.g., measured vs. predicted).

Confirmation of the approved models' applicability up through the projected end of cycle burnup for the Optimized ZIRLO™ fuel rods must be completed prior to their initial batch loading and prior to the startup of subsequent cycles. For example, prior to the first batch application of Optimized ZIRLO™, sufficient LTA data may only be available to confirm the models' applicability up through 45 GWd/MTU. In this example, the licensee would need to confirm the models up through the end of the initial cycle. Subsequently, the licensee would need to confirm the models, based upon the latest LTA data, prior to re-inserting the Optimized ZIRLO™ fuel rods in future cycles. Based upon the LTA schedule, it is expected that this issue may only be applicable to the first few batch implementations since sufficient LTA data up through the burnup limit should be available within a few years.

Safety Evaluation for CENPD-404-P-A, Addendum 1-A
Condition and Limitation 6

In the letter dated August 3, 2016 (Reference 21), the NRC staff concluded that:

> Westinghouse provided additional information from irradiation programs to comply with Conditions 6 and 7 of the SE in a letter LTR-NRC-13-6 dated February 25, 2013 (ADAMS [Agencywide Documents Access and Management System] Accession No. ML13070A188), and in the letter LTR-[NRC]-15-7, "Submittal of Responses to Draft RAIs and Revisions to Select Figures in LTR-NRC-13-6 to Fulfill Conditions 6 and 7 of the Safety Evaluation for WCAP-12610-P-A & CENPD-404-P-A Addendum 1-A (Proprietary/Non-Proprietary)," dated February 9, 2015 (ADAMS Accession No. ML15051A427).

The letter further explained that Westinghouse only intends to satisfy Conditions 6 and 7 through the information provided in LTR-NRC-13-6 and LTR-NRC-15-7. The data provided in LTR-NRC-13-6 and LTR-NRC-15-7 satisfy Conditions 6 and 7 and licensees no longer need to provide additional data when referencing WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A "Optimized ZIRLO™" in future LARS.

Therefore, the NRC staff concludes that this Condition and Limitation has been met.

3.3.10.7 Condition and Limitation 7

Cycle Data

The licensee is required to ensure that Westinghouse has fulfilled the following commitment: Westinghouse shall provide the NRC staff with a letter containing the following information (Based on the schedule described in response to RAI #11):

a. Vogtle growth and creep data summary reports.

b. Using the standard ZIRLO™ and Optimized ZIRLO™ database including the most recent Vogtle data, confirm applicability with currently approved fuel performance models (e.g., level of conservatism in W rod pressure analysis, measured vs. predicted, predicted minus measured vs. tensile and compressive stress).

Confirmation of the approved models' applicability up through the projected end of cycle burnup for the Optimized ZIRLO™ fuel rods must be completed prior to their initial batch loading and prior to the startup of subsequent cycles. For example, prior to the first batch application of Optimized ZIRLO™, sufficient LTA data may only be available to confirm the models' applicability up through 45 GWd/MTU. In this example, the licensee would need to confirm the models up through the end of the initial cycle. Subsequently, the licensee would need to confirm the models, based upon the latest LTA data, prior to re-inserting the Optimized ZIRLO™ fuel rods in future cycles. Based upon the LTA schedule, it is expected that this issue may only be applicable to the first few batch implementations since sufficient LTA data up through the burnup limit should be available within a few years.

Safety Evaluation for CENPD-404-P-A, Addendum 1-A
Condition and Limitation 7

In the letter dated August 3, 2016 (Reference 21), the NRC staff concluded that Westinghouse provided additional information from irradiation programs to comply with Conditions 6 and 7 of the SE in LTR-NRC-13-6 submitted in February 2013. Westinghouse provided additional information on February 9, 2015, in LTR-NRC-15-7. Westinghouse only intends to satisfy Conditions 6 and 7 through the information provided in LTR-NRC-13-6 and LTR-NRC-15-7. The data provided in LTR-NRC-13-6 and LTR-NRC-15-7 satisfy Conditions 6 and 7 and

licensees no longer need to provide additional data when referencing WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A "Optimized ZIRLO™" in future LARS.

Therefore, the NRC staff concludes that this Condition and Limitation has been met.

3.3.10.8 Condition and Limitation 8

Yield Strength (YS)

The licensee shall account for the relative differences in unirradiated strength (YS and UTS [ultimate tensile strength]) between Optimized ZIRLO™ and standard ZIRLO™ in cladding and structural analyses until irradiated data for Optimized ZIRLO™ have been collected and provided to the NRC staff.

a. For the Westinghouse fuel design analyses:

i. The measured, unirradiated Optimized ZIRLO™ strengths shall be used for beginning of life (BOL) analyses.

ii. Between BOL up to a radiation fluence of 3.0×10^{21} n/cm^2 [neutrons per square centimeters] (E>1 MeV [mega electron volt), pseudo-irradiated Optimized ZIRLO™ strength set equal to linear interpolation between the following two strength level points: At zero fluence, strength of Optimized ZIRLO™ equal to measured strength of Optimized ZIRLO™ and at a fluence of 3.0×10^{21} n/cm^2 (E>1 MeV), irradiated strength of standard ZIRLO™ at the fluence of 3.0×10^{21} n/cm^2 (E>1 MeV) minus 3 ksi [kilopounds per square inch].

iii. During subsequent irradiation from 3.0×10^{21} n/cm^2 up to 12×10^{21} n/cm^2, the differences in strength (the difference at a fluence of 3×10^{21} n/cm^2 due to tin content) shall be decreased linearly such that the pseudo irradiated Optimized ZIRLO™ strengths will saturate at the same properties as standard ZIRLO™ at 12×10^{21} n/cm^2.

b. For the CE fuel design analyses, the measured, unirradiated Optimized ZIRLO™ strengths shall be used for all fluence levels (consistent with previously approved methods).

Safety Evaluation for CENPD-404-P-A, Addendum 1-A
Condition and Limitation 8

PVNGS uses a CE fuel design, and therefore Condition and Limitation 8.a does not apply.

Because APS has stated that future analysis of Optimized ZIRLO™ will use the measured, unirradiated Optimized ZIRLO™ strengths for all fluence levels, the NRC staff has concluded that Condition and Limitation 8.b has been satisfied.

3.3.10.9 Condition and Limitation 9

LOCBART or STRIKIN-II early Peak Cladding Temperature (PCT)

As discussed in response to RAI #21 (Reference 3), for plants introducing Optimized ZIRLO™ that are licensed with LOCBART or STRIKIN-II and have a limiting PCT that occurs during blowdown or early reflood, the limiting LOCBART or STRIKIN-II calculation will be rerun using the specified Optimized ZIRLO™ material properties. Although not a condition of approval, the NRC staff strongly recommends that, for future evaluations, Westinghouse update all computer models with Optimized ZIRLO™ specific material properties.

Safety Evaluation for CENPD-404-P-A, Addendum 1-A
Condition and Limitation 9

APS has confirmed that the ECCS method of analysis has been updated to model the specific material properties. Therefore, the NRC staff has concluded that this condition and limitation has been met.

3.3.10.10 Condition and Limitation 10

Locked Rotor PCT

Due to the absence of high temperature oxidation data for Optimized ZIRLO™, the Westinghouse coolability limit on PCT during the locked rotor event shall be [[ ]].

Safety Evaluation for CENPD-404-P-A, Addendum 1-A
Condition and Limitation 10

This condition is specific to safety analysis methodologies used for Westinghouse PWRs, which use a cladding temperature limit on the locked rotor analysis. This locked rotor peak clad temperature limit does not apply to the safety analysis methodologies for CE plants such as PVNGS. Therefore, the NRC staff concludes that this condition and limitation is not applicable.

3.3.11 CENPD-404-P-A, Addendum 2-A

CENPD-404-P-A, Addendum 2-A (Reference 10), describes the Westinghouse fuel rod cladding corrosion model for ZIRLO™ and Optimized ZIRLO™ that replaces the corrosion model developed when ZIRLO™ was first licensed. Since (1) CENPD-404-P-A, Addendum 2-A, has been previously approved, (2) the licensee has proposed to revise TS 4.2.1 to support use of Optimized ZIRLO™, and (3) the supporting cladding material exemption for PVNGS is being approved concurrently in a separate SE, the NRC staff finds CENPD-404-P-A, Addendum 2-A, to be applicable to PVNGS.

The NRC staff's SE for CENPD-404-P-A, Addendum 2-A, contains four conditions and limitations. Licensees referencing Addendum 2-A of CENPD-404-P-A are expected to ensure compliance with the four conditions and limitations. The licensee has documented compliance with these four conditions and limitations. Each condition and limitation is restated below along

with the NRC staff's evaluation of APS's response. These evaluations are identical to those made as part of the related cladding material exemption that is documented in a separate SE.

3.3.11.1 Condition and Limitation 1

Corrosion Limit

The maximum TRDs [thermal reaction accumulated duties] are restricted to numbers corresponding to a cladding corrosion amount of 100 microns for licensing applications. The corrosion is defined as **[[**

]].

Safety Evaluation for CENPD-404-P-A, Addendum 2-A
Condition and Limitation 1

Since APS has indicated that the licensing calculations performed for the PVNGS NGF performance analysis consider a **[[** **]]** clad oxide thickness limited to a peak value of 100 microns, the NRC staff concludes that this condition and limitation has been met.

3.3.11.2 Condition and Limitation 2

Hydrogen Pickup Limit

The NRC staff requires that a hydrogen pickup limit of **[[** **]]** be implemented for ZIRLO™ and Optimized ZIRLO™ cladding.

Safety Evaluation for CENPD-404-P-A, Addendum 2-A
Condition and Limitation 2

APS has stated that the hydrogen pickup limit specified above from CENPD-404-P-A, Addendum 2-A, is used for ZIRLO™ and Optimized ZIRLO™ in the analysis done for PVNGS. Therefore, the NRC staff concludes that this condition and limitation has been met.

3.3.11.3 Condition and Limitation 3

Single Corrosion Limit

The NRC staff disapproves the Westinghouse assertion that a single corrosion limit could ensure cladding integrity without a separate hydrogen pickup limit. The hydrogen pickup limit in the current existing topical reports including References 9 and 10 **[**WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A**]** for ZIRLO® and Optimized ZIRLO™ cladding shall not be replaced.

Safety Evaluation for CENPD-404-P-A, Addendum 2-A
Condition and Limitation 3

APS has confirmed that the hydrogen pickup limit, as specified, was retained and evaluated in the PVNGS NGF licensing evaluation. Therefore, the NRC staff concludes that this condition and limitation has been met.

3.3.11.4 Condition and Limitation 4

Condition Removal and Fuel-Duty Index (FDI) Corrosion Model

Condition 4 in CENPD-404-P-A SE can be removed. And, the NRC staff disapproves the use of the FDI-based corrosion model for any future licensing applications.

Safety Evaluation for CENPD-404-P-A, Addendum 2-A
Condition and Limitation 4

APS confirmed that FDI-based corrosion analyses were not used. Therefore, the NRC staff concludes that this condition and limitation has been met.

3.3.12 CENPD-183-A

CENPD-183-A, "Loss of Flow, C-E Methods for Loss of Flow Analysis" (Reference 22), is currently included in PVNGS TS 5.6.5.b and the COLR list of analytical methods, but will be implemented differently for NGF than originally described. Therefore, the applicability and acceptability of CENPD-183-A is reevaluated here.

CENPD-183-A describes the assumptions, conservatisms, and basic methods used for analyzing events involving the loss of forced reactor coolant flow. The main body of the report describes a loss-of-flow analysis method for use with a computer code having transient core thermal hydraulic capabilities (referred to as the dynamic method). The appendix to CENPD-183-A describes a similar loss-of-flow analysis method for use with a steady-state core thermal-hydraulic code (referred to as the static method). The difference in implementation for NGF stems from the requirement for an updated flow coastdown curve due to the pressure drop changes introduced by NGF. Ultimately, however, the underlying methodology remains in effect for the current application.

Since CENPD-183-A has been previously approved for use at PVNGS, the underlying methodology remains unchanged, and the flow coastdown effects due to NGF are considered. Therefore, the NRC staff finds CENPD-183-A to continue to be applicable to PVNGS, Units 1, 2, and 3.

Similar to many topical report SEs of its era, the NRC staff's SE for CENPD-183-A (dated May 12, 1982) does not explicitly identify conditions and limitations. However, the licensee extracted three statements from within the SE that are essentially equivalent to conditions and limitations, and which will be treated as such in this evaluation. Licensees referencing CENPD-183-A are therefore expected to ensure compliance with these three conditions and limitations. The licensee has documented compliance with these three conditions and limitations. Each condition and limitation is restated below along with the NRC staff evaluation of APS's response.

3.3.12.1 Condition and Limitation 1

Approved Codes

The computer codes specifically approved by the NRC for use in conjunction with performing LOF [loss of flow] analyses using the approach described in CENPD-183 are:

a. COAST
b. QUIX
c. COSMO/W3
d. TORC/CE-1
e. CESEC

Therefore, no other computer codes may be used without prior NRC approval.

Safety Evaluation for CENPD-183-A
Condition and Limitation 1

The conditions imposed by the NRC in its SE for CENPD-183-A have been complied with as part of the LOF analyses supporting implementation of the NGF design at PVNGS. The computer codes utilized in the NGF LOF analyses for PVNGS differ from those cited in Condition 1. Since the time of submittal of CENPD-183-A, a revised set of computer codes has received NRC approval as replacements for those cited in Condition 1, specifically:

- the CENTS code was used to generate both the flow coastdown curve and the system response, replacing the COAST and CESEC codes,
- the transient neutronics response is modeled with the HERMITE code rather than the QUIX/COSMO codes, and
- the thermal-hydraulic (DNBR) response is modeled with the VIPRE-W or CETOP-D code replacing TORC code with DNB correlations, as discussed above.

Each of these code replacements has been previously approved by the NRC. Therefore, the NRC staff concludes that this condition and limitation has been met.

3.3.12.2 Condition and Limitation 2

Assumptions

These assumptions will result in lower DNBR and are therefore acceptable.

a. The assumptions referred to are:

i. Most adverse initial conditions
ii. Most adverse reactivity coefficients
iii. Maximum system response delay

Safety Evaluation for CENPD-183-A
Condition and Limitation 2

APS has confirmed that the initial condition assumptions above are consistent with those utilized in the PVNGS LOF analyses. The NRC staff has reviewed this information and concludes that this condition and limitation has been met.

3.3.12.3 Condition and Limitation 3

Fuel Damage Probability Distribution for COSMO/W-3

If COSMO/W-3 is used for DNBR calculations, the applicant is required to submit a fuel damage probability distribution for staff's approval.

Safety Evaluation for CENPD-183-A
Condition and Limitation 3

The probability density function correlation used for PVNGS has been generated with ABB-NV and WSSV CHF data. The ABB-NV and WSSV fuel failure data were generated as part of the fuel failure calculation based on the NRC-approved CHF correlation statistics in WCAP-14565-P-A, Addendum 2-P-A, and WCAP-16523-P-A. The data interface for the rods-in-DNB calculation process no longer requires the DNB probability density function data be provided in table form such as Table 2 of CENPD-183-A. Therefore, the NRC staff concludes that this condition and limitation has been met.

3.3.13 Conclusion on Analytical Methodologies Supporting NGF Implementation at PVNGS

The NRC staff has found each of the methodologies that is to be added to TS 5.6.5.b and the COLR list to be applicable to PVNGS and acceptable for use at PVNGS with NGF on the basis that each methodology has previously been generically approved and all conditions and limitations imposed by the NRC staff have been met or otherwise dispositioned for PVNGS on a plant-specific basis.

3.4 NGF Lead Fuel Assemblies

Eight lead fuel assemblies (LFAs) of NGF were introduced in PVNGS Unit 3 starting in Cycle 16. The LFAs were inspected following Cycle 18 operation (3 cycles of LFA operation). This inspection included visual examinations, guide tube eddy current testing, fiberscope visual inspection, and measurements of assembly length, shoulder gap, rod bow, and peripheral rod oxide.

APS reported the findings as follows:

- Visual examinations did not reveal any mechanical or structural abnormalities.
- Guide tube eddy current testing was performed on half of the assemblies and showed insignificant wall wear.
- Fiberscope visual inspection showed normal corrosion appearance.
- Assembly length measurements were taken for half of the LFAs and showed a maximum growth of 0.35 inches, which agrees well with data taken for other designs at PVNGS.
- Shoulder gap measurements were taken on half of the LFAs and showed that normal uniform positioning of rods was maintained and that rod growth was consistent with that seen in ZIRLO™-clad rods.
- Rod bow measurements were taken for two of the LFAs and compared favorably to channel closure limits.
- Oxide measurements on peripheral rods showed considerable margin to the limit.

The NRC staff finds the results of the LFA inspections reasonable and concludes that these results adequately demonstrate the performance of NGF and support full-core implementation at PVNGS.

3.5 Safety Analyses

3.5.1 Introduction

Using approved methodologies, APS has evaluated each fuel-related safety analysis in the PVNGS licensing basis in order to support implementation of NGF. Some AORs were found to remain bounding for NGF and others were adjusted and reevaluated using approved methodologies. These analyses include mechanical and core design, fuel performance and corrosion, thermal hydraulics, SRP Chapter 15 transients, ECCS performance/LOCA, and radiological evaluations, among others.

The objective of the NRC staff's review of these analyses is to determine that each have been performed using acceptable methods and that each demonstrates acceptable results according to the SAFDLs for PVNGS, the criteria of 10 CFR 50.46, or other applicable regulatory limits. Such a determination is necessary to support a conclusion that PVNGS can continue to operate safely following the proposed transition to NGF.

3.5.2 Mechanical Design

The NGF mechanical design was assessed against design criteria to ensure that it remains within the design basis established for PVNGS. Many of the analyses were previously performed for the LFA irradiation project (Reference 23). However, with the application of the grid spacer design changes approved in WCAP-16500-P-A, Supplement 2, a number of these analyses required updating. These grid spacer design changes result in a slightly increased pressure drop, which leads to slight changes in core operating conditions, fuel assembly burnup-dependent properties, scram times, and resultant structural loads. Additionally, some analyses required updating to account for the effects of full-batch or full-core implementation of NGF, rather than the small number of LFAs the analyses originally considered.

The analyses that were updated to include the above effects are the fuel assembly structural integrity analyses for both non-seismic/LOCA and seismic/LOCA events, CEA scram time and structural analyses, and fuel rod analyses (stress, strain, clad flattening, and maximum internal pressure). The following sections address these analyses, including the NRC staff's review and determination of acceptability.

3.5.2.1 Fuel Assembly Structural Analyses (Non-Seismic/LOCA)

The LFA analyses previously performed were adjusted to account for the spacer grid changes and the resulting operating conditions changes. The specific design criteria were reevaluated as follows.

Fuel Rod Growth. The LFA growth prediction remains unchanged by the grid design changes, as does the minimum shoulder gap. Therefore, the NRC staff has concluded that the LFA growth analysis remains applicable to NGF with the grid design changes.

Stress Limits. The limiting loadings are non-operational and unaffected by full-region[6] NGF implementation. The only non-seismic/LOCA load calculated was for holddown spring force, which was shown to remain applicable to NGF with the grid design changes. Therefore, the NRC staff has concluded that this criterion remains satisfied since the stress margins remain unchanged from the LFA analysis.

Tensile Load Requirement. Similar to the stress limits above, the tensile load is non-operational and unaffected by full-region NGF implementation. Therefore, the NRC staff has concluded that this criterion remains satisfied for NGF with the grid design changes.

Interface Criteria – Reactor Internals.

- **Axial Engagement of Nozzles.** The only effect due to full-core implementation with the grid design changes is a slight increase in differential thermal expansion between the fuel and internals. Regardless, the core insert pin height remains much greater than the maximum possible axial assembly movement, ensuring the assembly remains engaged for all operating conditions. Therefore, the NRC staff has concluded that this criterion remains satisfied for NGF with the grid design changes.

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[6] The term "full-region" is used by Westinghouse to signify the full batch of fresh fuel assemblies loaded into the reactor core during a refueling outage to support the next operating cycle; thus, each "full region" comprises approximately one-third to one-half of the reactor core.

- **Provision for Fuel Assembly Growth.** The differential thermal expansion between the fuel and internals mentioned above leads to a slight increase in clearance to the upper guide structure. Additionally, the LFA growth prediction remains applicable to NGF with the grid design changes. Therefore, the NRC staff concludes that this criterion remains satisfied for full-region implementation of NGF with the grid design changes.

- **Loads on Reactor Internals.** The holddown spring force and fuel uplift margin calculations, completed as part of the LFA analysis, were found to remain applicable to full-region NGF implementation with the WCAP-16500-P-A, Supplement 2-A, grid design changes. Therefore, the NRC staff concludes that this criterion remains satisfied.

Interface Criteria – Compatibility with Co-Resident Fuel Assemblies. The pressure drop associated with full-region implementation of NGF with the grid design changes was found to be between those of the STD fuel and the NGF LFAs. Therefore, the NRC staff has concluded that the LFA analysis conservatively bounds the effects of full-region NGF implementation and the criterion remains satisfied.

Holddown Force. Holddown spring forces remain unchanged and uplift forces are conservative compared to the LFA analysis. Therefore, the NRC staff has concluded that this criterion remains satisfied for full-region implementation of NGF with the grid design changes.

Spring Stress. Spring stress evaluations were performed at full compression and, therefore, are unaffected by full-region implementation of NGF with the grid design changes. The NRC staff concludes that this criterion remains satisfied.

CEA Impact Velocity Interface. The CEAs impact the fuel alignment tube sheet, not the fuel assembly. Therefore, the impact analyses remain unchanged and the NRC staff concludes that the criterion remains satisfied.

Instrument Tube/Top and Bottom Nozzle Interface. The clearance calculated for the LFAs is reduced by approximately half for full-region implementation of NGF with the grid design changes. However, the criterion is established only to show that a gap exists. Therefore, the NRC staff concludes that the criterion remains satisfied.

Top Nozzle Alignment and Engagement. Assembly growth predictions remain applicable and the only adjustment to engagement is the miniscule reduction due to the larger differential thermal expansion between the fuel and internals. Even with this slight reduction, the engagement criteria remain met for full-region implementation of NGF with the grid design changes.

The NRC staff has concluded that each of the specific design criteria above has been adequately met, demonstrating the acceptability, with respect to non-seismic/LOCA fuel assembly structural analyses, of full-region implementation of NGF with the WCAP-16500-P-A, Supplement 2, grid spacer design changes at PVNGS.

3.5.2.2 Fuel Assembly Structural Analyses (Seismic/LOCA)

The previously approved methodology from CENPD-178-P, Revision 1-P (Reference 24), was used to calculate seismic and branch line pipe break response loads on the NGF assemblies and grid spacers. The NRC staff issued an RAI regarding the type of apparatus used to excite the fuel assembly during forced vibration testing. The staff's concern regarded a possible discrepancy between the actual testing performed and the approved methodology, which specifies that a hydraulic shaker attached to the bottom of the assembly be used. This concern was raised since the staff was aware that the originally approved apparatus was replaced with an electro-mechanical shaker that attaches at the center of the assembly.

In response to the RAI (Reference 2), APS acknowledged that the shaker apparatus was changed. APS indicated that the substituted apparatus serves an identical purpose to the original, as in it is used to obtain fuel assembly natural frequencies, mode shapes, and damping. The application of these testing results is then applied following the CENPD-178-P, Revision 1-P, methodology.

The seismic/LOCA structural analysis and testing demonstrated that spacer grid strengths remain greater than the predicted impact loads that occur during a combined seismic and LOCA event, except for some peripheral STD assemblies in transition cores. For these assemblies, coolability and insertability analyses were performed as alternate justification, as specified in CENPD-178-P, Revision 1-P.

The coolability analysis considered a range of possible grid spacer deformations, which would give hot channel flow area blockages from zero to a maximum of **[[ ]]**. The analysis determined that no penalty to the ECCS performance analysis results is necessary for PVNGS for NGF or transition cores.

The NGF assemblies were evaluated with the co-resident STD assemblies and the CEAs for transition cores under seismic and LOCA conditions to demonstrate compliance with design criteria. The evaluations conclude that the resulting loadings and stresses satisfy their respective design criteria.

The NRC staff concludes that the results of the seismic/LOCA analyses performed using the approved methodology adequately demonstrate that the associated design criteria have been met and show that transition to NGF can be safely accomplished at PVNGS with respect to structural response to seismic/LOCA loadings and ability to safely shut down the reactor following a seismic/LOCA event.

The NRC staff notes that the analyses performed only consider BOL conditions, the conservatism of which has recently been challenged, as described in NRC IN 2012-09 (Reference 18). In the event that the plant experiences a seismic event with vibratory ground motion greater than the operating basis earthquake, as per 10 CFR Part 100, Appendix A requirements, the plant is required to shut down. According to the regulation, prior to restarting, the licensee must demonstrate that no functional damage has occurred to design features necessary for continued operation without undue risk to the health and safety of the public. This issue is discussed more fully below in Section 3.6.

3.5.2.3 CEA Scram Time Analyses

Scram times were calculated for each unit for full cores of both STD fuel and NGF. The NGF scram times are [[ ]]. The NRC staff issued RAI 2 requesting justification that the full-core CEA scram time analyses for STD and NGF bound those expected during transition cores.

In response to RAI 2 (Reference 2), APS explained that the difference in computed scram time for the two fuel designs is [[

]] for any transition core containing STD fuel and NGF. Additionally, the CEA insertion curves for both fuel designs are [[ ]] used for safety analysis. Therefore, the NRC staff concludes that the NGF scram time analysis results are acceptable and support the transition to NGF at PVNGS.

3.5.2.4 Fuel Rod Analyses

Fuel rod design analyses are broken into two sets of calculations. One fuel performance analysis is performed in FATES3B and is discussed in Section 3.5.4 of this SE. This section covers the mechanical performance analysis, which demonstrates NGF's ability to meet cladding collapse, cladding fatigue, cladding stress, and cladding strain criteria. This analysis does not have downstream impacts on other analyses. The calculations in this analysis were performed on a bounding basis using conservative inputs related to core physics and plant conditions, while explicitly accounting for the geometry and material properties of the fuel design.

Satisfaction of cladding burst acceptance criteria for all CE ZrB_2 fuels was demonstrated as part of CENPD-404-P-A, which concluded that cladding burst is precluded if PCT and engineering hoop stress remain below the values specified in Section 4.4.2.1 of CENPD-404-P-A. For NGF, an analysis was performed to show that these criteria remain valid and bounding. The results were shown to be well below these limits. Therefore, the NRC staff concludes that the acceptance criteria are satisfied for the NGF design.

3.5.2.5 Conclusion on Mechanical Design Analyses

The NRC staff has reviewed the mechanical design analyses provided and has determined that the analyses demonstrate that all mechanical design criteria have been met and that these analyses adequately support the transition to NGF at PVNGS, Units 1, 2, and 3.

3.5.3 Core Design

The purpose of this section is to demonstrate the capability for PVNGS to safely transition to NGF, with respect to neutronics and fuel cycle design, while remaining within limits for fundamental design parameters such as MTC of reactivity, peaking factors, and burnup. To accomplish this, a preceding cycle analysis of STD fuel with erbia burnable absorber was built upon with four subsequent cycle analyses. These analyses were developed to demonstrate representative core loading progressions that could be used during transition to NGF. The first three cycles represent the first, second, and third subsequent full-feed reloads (and associated fuel shuffles) of NGF into the core. The final of the four analyses represents a "near equilibrium"

full core of NGF (except for a single, central STD assembly, as discussed below) in order to demonstrate continued operation after transition.

The cycle analyses were performed in CASMO-4/SIMULATE-3 using the methodologies described in WCAP-16072-P-A (Reference 14) and CENPD-266-P-A (Reference 25). In the analyses, the center assembly is assumed to be a reinserted, twice-burnt STD assembly. When suitable STD fuel assemblies are no longer available for this use, NGF assemblies would be used, provided that peaking and fluence limits are met.

Neutronically, the main differences between the STD and NGF assemblies are a small decrease in pellet and rod diameters, a higher poisoned pellet stack density, and the use of IFBA rather than erbia. These differences lead to a slightly higher reactivity, softer neutron spectrum, and slightly higher MTC, though these effects can be readily controlled through varying fuel and burnable poison loading.

The analysis for each cycle follows the nuclear design process, which assures that design limits for parameters such as MTC, peaking factors, and burnup are all met. APS also provided a comparison of boron letdown curves and the peaking factors determined for each of the representative cycles.

Based on the results of these analyses and their compliance with design criteria, the NRC staff concludes that APS has adequately demonstrated that transition from STD to NGF can be accomplished at PVNGS while meeting fundamental design limits, with respect to nuclear design.

3.5.4 Fuel Performance

This section covers the fuel performance analyses performed using FATES3B that were mentioned in Section 3.5.2.4 of this SE. Some of the fuel performance parameters of interest include rod internal pressure, power-to-melt, axial densification factor, engineering factor on linear heat rate, and spent fuel rod internal pressure.

A fuel performance analysis was performed using approved methods (References 26, 27, 28, and 29) to demonstrate acceptable performance relative to fuel criteria over the design lifetime of NGF rods. Additionally, this analysis provided information to other downstream analyses. Similar analyses were also performed for co-resident STD rods. The analyses were performed to assess if there is acceptable power fall-off margin relative to what is predicted by the reactor physics analyses for the NGF implementation cycles mentioned in Section 3.5.3.

The fuel performance analysis produced bounding radial fall-off (RFO) curves for both NGF and STD fuel with and without burnable absorbers. On a cycle-to-cycle basis, the predicted powers are compared to these bounding RFO curves in order to verify that the results of the bounding analysis remain applicable and continue to bound the cycle of interest. This ensures that the limiting power-to-melt and the maximum allowable rod internal pressure limits will be met. Confirmation of satisfactory fuel performance results is performed for every reload. These results include the confirmation of fuel rod design criteria and results for use in the mechanical design, the non-LOCA analysis, the ECCS performance/LOCA analysis and the digital setpoints functions.

The licensee used the FATES3B fuel performance code to generate predictions of fuel performance parameters. The licensee applied bounding power histories, axial power shapes and peaking factor reduction with burnup with the intention of producing conservative results. However, the effects of thermal conductivity degradation (TCD) are not explicitly accounted for in FATES3B. As a result of accepting inputs from the FATES3B code, the STRIKIN-II code the licensee used for its LOCA analysis is similarly affected. If not otherwise compensated for, neglect of TCD in these current licensing basis analysis methods would result in a non-conservative underestimate of the fuel stored energy at higher burnups where TCD occurs.

To address the non-conservative neglect of TCD in current licensing basis analysis methods, the licensee originally proposed a voluntary commitment to impose a penalty on the RFO curves applicable to upcoming PVNGS core designs using NGF. This penalty would be specified in the form of an equivalent centerline fuel temperature reduction. To ensure that the margin reserved by this penalty adequately accounts for the neglect of TCD, this allowance would be included in the evaluation of the fuel performance criteria and the safety analysis inputs used in non-LOCA and LOCA safety analyses. This proprietary temperature allowance is detailed in Table 4-1 of Attachment 8 to the submittal (Reference 1) and reproduced below:

[[

]]

The NRC staff requested in RAI 6 that the licensee provide justification for the RFO penalty it proposed as a voluntary commitment. The licensee's basis for the proposed RFO penalty was discussed during an audit conducted on March 8, 2017 (Reference 30). The licensee subsequently provided a formal response to RAI 6 that was largely based upon proprietary analyses. These proprietary analyses were code-to-code comparisons intended to determine calculated differences in fuel temperatures and rod internal pressures between FATES3B and PAD5, a recently approved fuel thermal-mechanical performance code that explicitly accounts for the impacts of TCD. The licensee stated that the results of the calculations demonstrate that the conclusions made in the NGF LAR regarding the RFO curve penalty are unchanged. The licensee considered a comprehensive set of fuel performance criteria and assessed both LOCA and non-LOCA events. The NRC staff reviewed the licensee's justification for the RFO penalty and determined that the licensee's response provides an appropriate level of detail to address the staff's concern. The NRC staff observed that one of the inputs to the PAD5 calculation was slightly reduced relative to the value used in the original FATES3B analysis in order to ensure that the RFO curve penalty would be bounding. **[[**

]] Hence, future core designs relying upon this analysis would be required to be consistent with the reduced input. The NRC staff also observed that the PAD5 results calculated by the licensee are not necessarily bounded by the FATES3B-calculated results for all parameters at each burnup step; however, the PAD5 results are bounded by the limiting values assumed in the AORs for fuel average temperature and other key parameters.

The licensee's response further proposed upgrading the voluntary commitment to implement an RFO curve penalty from its original submittal into a license condition. The NRC staff agreed that imposition of a license condition is necessary in this circumstance, since credit for the more restrictive RFO penalty would be necessary to support the NRC staff's approval of the proposed LAR. In particular, without the RFO curve penalty, non-conservative predictions of fuel temperature, stored heat, or other fuel-performance-related parameters could occur.

The NRC staff reviewed the final version of the license condition proposed by the licensee in its letter dated December 15, 2017, and determined that APS must adhere to the following license condition respecting the use of CE 16×16 NGF at PVNGS.

> 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.
>
> 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.

Contingent upon the imposition of the above license condition, the NRC staff has confidence that the neglect of TCD by the current licensing basis analysis methods in use at Palo Verde will be acceptably addressed for NGF through the reservation of sufficient margin via a penalty to the RFO curve. The above license condition states explicitly that it is applicable to NGF for each future reload cycle for which the FATES3B code is used. Therefore, should the licensee transition in a future operating cycle to NRC-approved analytical methods that explicitly account for TCD, performance of the actions described in this license condition would no longer be necessary.

The NRC staff has concluded that the licensee's approach for ensuring fuel performance limits are met on a cycle-to-cycle basis is acceptable on the basis that (1) the bounding RFO curves are generated using an acceptable methodology, (2) a license condition will be applied to address the non-conservative neglect of TCD in current analysis methods, and (3) comparisons will be made for each cycle to ensure that predicted fuel performance remains bounded.

3.5.5 Core Thermal-Hydraulic Design

3.5.5.1 Introduction

This section describes the core thermal-hydraulic (T-H) analysis methodologies performed in support of the NGF transition LAR. The section includes a brief description of the T-H methods to be used with NGF, discussion of the compatibility of the NGF assembly with the STD fuel during transition cycles, and other DNB-related effects.

3.5.5.2 T-H Methods

The T-H methods used for the analyses to support the transition to NGF are the same as those described in the current UFSAR, with the exception of the following five approved changes:

- The NRC-approved Westinghouse version of the VIPRE-01 subchannel analysis code (i.e., VIPRE-W) is used in DNBR calculations.

- The NRC-approved ABB-NV CHF correlation is used in place of the CE-1 CHF correlation to predict DNBR in the co-resident STD fuel during transition cores.

- The NRC-approved WSSV, WSSV-T CHF correlation is applied to mixing vane and IFM grid regions, and the ABB-NV CHF correlation is applied to non-mixing vane grid regions to predict DNBR in NGF assemblies.

- The NRC-approved SCU methodology was incorporated and supplemented to [[ ]] to calculate an overall uncertainty factor.

- The NRC-approved WLOP CHF correlation can be used to supplement the ABB-NV and WSSV correlations to predict DNBR at low-pressure conditions as an alternative to the Macbeth correlation.

As discussed in the previous sections, these methodologies have been reviewed, approved, and found applicable for use at PVNGS, Units 1, 2, and 3, by the NRC staff.

3.5.5.3 NGF Assembly

From a hydraulic perspective, significant design differences incorporated into NGF assemblies as compared to STD fuel, include the addition of mixing vanes and IFM grids and a slightly increased assembly flow area due to a smaller fuel rod diameter. Though the additional features contribute to a larger overall assembly pressure drop, Westinghouse stated that the overall thermal performance has been improved. The design of the NGF assembly has been previously approved by the NRC in WCAP-16500-P-A and its two supplements. This LAR provides justification, and ultimately NRC approval, for the use of NGF assemblies at PVNGS, Units 1, 2, and 3.

3.5.5.4 Compatibility with STD fuel

Transition to NGF will require several fuel cycles involving operation with both NGF and STD present in the core simultaneously. Differences in T-H characteristics between the fuels have been examined to ensure that any adverse effects imposed by either type of co-resident assemblies onto the other type are adequately accounted for.

As discussed above in Section 3.3.1.6, the NGF assemblies may experience flow starvation due to flow diversion away from the NGF mixing vane grid region in transition cores with STD fuel. However, the licensee has demonstrated that thermal margin gains from the WSSV CHF correlation will offset the adverse impacts of flow diversion in that region.

Another potential issue associated with transition cycles is that the difference in pressure drop between the different assemblies affects inter-assembly crossflow, leading to GTRF concerns. Therefore, APS performed full-scale hydraulic testing, including a 500-hour accelerated wear test, on the NGF and STD fuel designs to investigate the GTRF characteristics of the NGF assembly and the mixed-core impacts on the GTRF performance of both the NGF and STD assemblies. The NRC staff notes that the full-scale testing of the NGF assemblies was performed without the inclusion of the minor spacer grid changes detailed in WCAP-16500-P, Supplement 2-A. Due to the insignificance of the T-H effects these changes cause, the staff has concluded that the full-scale hydraulic tests performed continue to apply to NGF assemblies that include the Supplement 2 grid spacer changes. Ultimately, the wear analysis performed met Westinghouse's field wear criteria for transition cycles. Therefore, the NRC staff is reasonably assured that co-resident NGF and STD assemblies will have adequate GTRF performance.

3.5.5.5 DNB Effects

Limiting assembly candidates are selected based on two criteria: **[[**

]]. The candidates then enter a **[[** **]]** VIPRE-W execution process used to determine the limiting assemblies and CETOP-D correction factors. This process is largely similar to the TORC/CETOP-D process previously used, and yields results that are more conservative (i.e., result in a larger overpower penalty).

The online reactor monitoring and protection systems rely on CETOP-D, which **[[**
]]. Therefore, in applying the CETOP-D model during the first transition cycle, **[[**
]]. During the second transition cycle, however, the remaining twice-burnt STD fuel assemblies would no longer be limiting assembly candidates. Therefore, the CETOP-D model based on the NGF design with WSSV/ABB-NV CHF correlations can be applied with the corresponding SCU DNBR safety limit of 1.25. In either case, the appropriate rod bow penalty is applied according to the methodology in CENPD-225-P-A, approved for use with NGF in WCAP-14656-P, Addendum 2-A.

Condition III and IV events that result in DNB (i.e., the calculated DNBR falls below the appropriate limit) are analyzed to quantify the number of rods that would undergo DNB and be assumed to fail. The NRC staff issued an RAI regarding the methodology used for the fuel failure analyses. Specifically, differences between the NGF analysis and the AOR were of interest, including the relevance of overall time in DNB. In response to the RAI (Reference 2),

APS explained that the [[ ]] approach was applied to execute a steady-state core thermal-hydraulic VIPRE-W model where the boundary conditions provided by the system transient analyses [[ ]]. The [[ ]] analyses determined that the number of rods predicted to be in DNB indicate that the current UFSAR dose analyses remain satisfied.

APS explained that the overall time in DNB was used as a criterion for DNB propagation based on the bounding clad strain evaluation. The analysis identified the time in DNB necessary to reach the NRC-imposed cladding strain limit of 29.3 percent from CEN-372-P-A. Based on this, DNB propagation can be precluded for events that result in overall times in DNB less than the identified time, which was conservatively rounded down to 4 seconds.

For low-pressure events (e.g., main steam line break, post-reactor trip), the approved WLOP correlation was used in VIPRE-W and compared to the applicable correlation-based DNBR limit.

Based on the above information, combined with the demonstrated non-LOCA and LOCA Chapter 15 event performance (evaluated below in Sections 3.5.7 and 3.5.8 of this SE), the NRC staff concludes that the STD and NGF assemblies are hydraulically compatible and the introduction of NGF is expected to provide acceptable T-H performance and adequate margin to DNB.

3.5.6 Fuel Rod Corrosion

The licensing requirements for fuel rod corrosion are listed in WCAP-16500-P-A and CENPD-404-P-A, Addendum 2-A. Specifically, for Optimized ZIRLO™-clad NGF (and the ZIRLO™ clad STD) [[ ]] is limited to the licensed peak value of 100 microns. Additionally, clad hydrogen pickup is limited to a best-estimate volumetric average value of [[ ]] at end-of-life (EOL). The maximum TRDs in the cladding oxidation models are also limited to values corresponding to a thickness of 100 microns.

The calculations were performed using approved TRD-based models based on projected conditions for a number of PVNGS future cycles. The analysis of NGF implementation at PVNGS showed that maximum predicted oxide thickness, maximum predicted volume-average hydrogen pickup and maximum predicted TRD limits are satisfied with significant margin. Additionally, it was shown that mixed cores and full cores of NGF were both able to meet these limits. Therefore, the NRC staff concludes that the NGF design can be expected to meet corrosion limits during and following transition to NGF at PVNGS.

3.5.7 UFSAR Chapter 15 Non-LOCA Transient Events

3.5.7.1 Introduction

All UFSAR Chapter 15 non-LOCA transient events were evaluated consistent with currently approved methodologies to ensure their applicability to NGF. Each of these evaluations was based on the rated thermal power of 3990 megawatt thermal (MWt) combined with uncertainty, yielding an analysis power of 4070 MWt. The analyses were evaluated using an analytical DNBR limit of 1.38, which is based on approved DNBR safety limits with some added discretionary margin for conservatism.

In total, the UFSAR Chapter 15 transients consist of thirty-five non-LOCA analyses and two additional considerations from Chapter 15 appendices.[7] Each was examined to determine the effects of transitioning to NGF on the applicability and acceptability of the analysis. It has been previously found that five of these analyses are not applicable to System 80 PWRs such as PVNGS, and therefore require no further consideration. Two of the Chapter 15 events, LOCA and Radiological Material Release, are addressed in separate sections of this SE.

It was stated that the fuel transition has no impact on four of the applicable analyses. The events stated to be unaffected include the startup of an inactive reactor coolant pump (RCP), inadvertent deboration, inadvertent operation of the ECCS, and steam generator (SG) tube rupture events. The NRC staff issued an RAI that asked for justification that these events are not impacted by the fuel transition.

In response to the RAI (Reference 2), APS explained specifically why each event was not affected by the introduction of NGF. The NRC staff has considered each of these events and the explanations provided by APS and agrees with the determination that these events remain unaffected by the change in fuel design.

The remaining twenty-four non-LOCA analyses (and two other considerations from UFSAR Chapter 15 appendices) are affected by NGF implementation. Of these affected events, it was determined that eight of them are bounded by another related Chapter 15 event, and therefore, they 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.

Table 1: Bounded Events from UFSAR Chapter 15

UFSAR Section	Event	Bounding Chapter 15 Event(s)
15.1.1	Feedwater (FW) Temperature Decrease	This event is bounded by the steam flow increase event (UFSAR Section 15.1.3) and the two inadvertent dump valve opening events (UFSAR Section 15.1.4).
15.1.2	Increase in Main FW Flow	This event is bounded by the steam flow increase event (UFSAR Section 15.1.3) and the two inadvertent dump valve opening events (UFSAR Section 15.1.4).
15.2.1	Loss of External Load	This event is bounded by the Loss of Condenser Vacuum event (UFSAR Section 15.2.3).
15.2.2	Turbine Trip	This event is bounded by the Loss of Condenser Vacuum event (UFSAR Section 15.2.3).
15.2.4	Main Steam Isolation Valve Closure	This event is bounded by the Loss of Condenser Vacuum event (UFSAR Section 15.2.3).

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

15.2.6	Loss of Non-Emergency Alternating Current (AC) 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 FW Flow	This event is bounded by the Loss of Condenser Vacuum event (UFSAR Section 15.2.3).
15.5.2	Chemical and Volume Control System Malfunction - Pressurizer Level Control System Malfunction with Loss of Offsite Power (LOP)	This event is bounded by the Total Loss of Reactor Coolant Flow event (UFSAR Section 15.3.1).

The NRC staff issued an RAI requesting justification for the determinations that each of these events is bounded. APS provided details regarding each of these determinations, discussed below.

The steam flow increase (UFSAR Section 15.1.3) and inadvertent dump valve opening events (UFSAR Section 15.1.4) bound the FW temperature decrease (UFSAR Section 15.1.1) and the increase in main FW flow (UFSAR Section 15.1.2) events. This is because these events result in the most limiting RCS cooldown and resultant power increase. Specifically, the parameter of concern for these events is the minimum hot channel DNBR. However, the resulting NSSS responses are bounded by those caused by the steam flow increase and inadvertent opening of steam generator atmospheric dump valve (with combined single failure) events. The FW temperature and flow events create less significant RCS cooldowns and resultant power increases when compared to the bounding events, and therefore are not as challenging with respect to the minimum hot channel DNBR.

The loss of condenser vacuum (LOCV) (UFSAR Section 15.2.3) event bounds the loss of external load (UFSAR Section 15.2.1), turbine trip (UFSAR Section 15.2.2), main steam isolation valve closure (UFSAR Section 15.2.4), loss of non-emergency AC power to the station auxiliaries (LOAC) (UFSAR Section 15.2.6), and loss of normal FW flow (UFSAR Section 15.2.7) events. Each of these events results in a pressurization of the RCS due to the isolation of steam and/or FW flow through various means, the most limiting of which occurs during the LOCV event. This isolation results in RCS pressurization and thus an increase in DNBR. This alleviates DNB concerns for most of these events (aside from the long-term portion of the LOAC event, discussed below), so the parameter of interest is RCS pressure. Since the LOCV event results in the greatest pressurization of the RCS, it is determined to be the bounding event for this subset of Chapter 15 analyses.

The total loss of reactor coolant flow (TLOF) event bounds the fuel performance portion (i.e. DNBR) of the loss of the LOAC event and the Chemical and Volume Control System Malfunction – Pressurizer Level Control System (PLCS) Malfunction with LOP event. Specifically, the results of the LOAC event are identical to those presented for the LOF event, since the LOAC is the initiating event for the LOF analysis. The results of the LOF analysis are used as justification for both events. Additionally, the DNBR degradation experienced during the PLCS malfunction events was determined to be less significant than for the LOAC/LOF event. Therefore, the LOF event is determined to be the bounding event for this subset of analyses.

The NRC staff concludes that the events are, in fact, bounded and do not require separate evaluation aside from the bounding events. Additionally, the NRC staff notes that these bounding events were examined and the results are considered as part of the review of the reanalyzed Chapter 15 events in the following section.

3.5.7.2 Reanalyzed UFSAR Chapter 15 Non-LOCA Events

As referred to above, there are sixteen Chapter 15 non-LOCA events and the two considerations from Chapter 15 appendices that are affected by NGF implementation and not bounded by other events. For each of these events, the licensee determined that the CENTS evaluations, completed, as part of the AORs, remain applicable for transient system response, and do not require reanalysis. The NRC staff issued an RAI regarding justification that the CENTS cases in the AOR remain unaffected by the transition to NGF.

In response to the RAI (Reference 2), APS explained that the CENTS evaluations were not reanalyzed due to the introduction of NGF for two reasons. The first, for events analyzed at constant flow, is that these events are analyzed at the TS RCS flow limits, regardless of the effects of the core pressure drop. Therefore, the changes introduced by NGF would have no effect on the response of the model for these events. The second, for events with significant flow coastdown (e.g., LOP events), is that the flow coastdown curves used in the AORs for these events were conservatively chosen and continue to bound the NGF-specific flow coastdown, as shown in the figure provided in the RAI response. Therefore, the NRC staff finds the use of the AOR CENTS evaluations acceptable for use in the reanalyzed Chapter 15 analyses.

The NRC staff's review of the reanalyzed Chapter 15 non-LOCA transients focuses on ensuring that approved methodologies are implemented appropriately and that results (overall transient system response, DNBR, linear heat generation rate (LHGR), and predicted fuel failures) are acceptable.

Increase in Secondary System Heat Removal Analyses (UFSAR Section 15.1)

In scenarios where offsite power is assumed to be available, the increase in main steam flow (IMSF) (UFSAR Section 15.1.3) and inadvertent opening of a steam generator atmospheric dump valve (IOSGADV) (UFSAR Section 15.1.4) events bound the decrease in FW temperature and increase in main FW flow events. The analyses were performed using the current UFSAR methodologies with the appropriate updated CHF correlations for NGF. This, combined with fuel rod and hot channel changes associated with implementation of NGF, resulted in revised DNBR values, but did not affect overall transient system response. The resultant minimum DNBR (mDNBR) values for the IMSF and IOSGADV Condition II events remain above the established limit of 1.38. Therefore, the NRC staff concludes that these events have been adequately analyzed and the results ensure that SAFDLs are not violated.

However, when **either** Condition II event is combined with a LOP, the resultant mDNBRs fall below the limit of 1.38. Therefore, these resulting Condition III events (i.e., anticipated transient plus LOP) require fuel failure analyses. The calculated fuel failures associated with these events are bounded by the current UFSAR AORs. The NRC staff issued an RAI requesting that APS justify that the AOR fuel failure analyses do, in fact, bound the NGF analyses. In response to the RAI, APS indicated that a bounding fuel failure fraction of 5.5 percent is assumed in the AOR dose analyses. Additionally, it was confirmed that this assumption remains bounding to

the fuel failure levels calculated for these events with NGF considered, and therefore remains conservative.

In these analyses, it was found that the overall time in DNB was less than 4 seconds and, therefore, it was concluded that DNB propagation would not occur. Due to these facts, the NRC staff concludes that the analysis results of the IMSF and IOSGADV events, when combined with LOP (i.e., which results in Condition III events), are acceptable.

Analysis of the main steam line break (MSLB) event is contained in Section 15.1.5 of the PVNGS UFSAR. Although the MSLB event is classified as a limiting fault (i.e., Condition IV event), the licensee is typically able to demonstrate that SAFDLs will not be violated and no fuel damage will occur. UFSAR Section 15.1.5 analyses include the HFP and hot, zero-power (HZP) post-trip MSLB events with and without LOP and the HFP pre-trip MSLB event, each evaluated for postulated breaks both inside and outside containment.[8] Additionally, UFSAR Section 15.1.6 analysis evaluates steam system piping failures inside and outside containment during Mode 3 operations. Each of the four post-trip Modes 1 and 2 MSLB events (i.e., HFP, HFP+LOP, HZP, and HZP+LOP cases) was analyzed using approved methodologies. Of these, the HFP+LOP case is most limiting. Each of the other three analyses resulted in mDNBR values greater than 5 using the Macbeth correlation. The HFP+LOP analysis resulted in a mDNBR of 2.38, a slightly reduced value than obtained in the past, but still well above the Macbeth SAFDL limit of 1.30. For each analysis, it was confirmed that the LHGR safety limit was not exceeded and the overall system transient response was not changed. Therefore, the NRC staff concludes that the post-trip phase of the MSLB events has been adequately analyzed and the results conservatively demonstrate that SAFDLs are not violated.

The pre-trip phase of the Modes 1 and 2 MSLB event was also analyzed for postulated breaks inside and outside containment using the current UFSAR methodology. Overall transient system response remains unchanged and only DNBR is affected by transition to NGF. For the pre-trip phase of the MSLB outside containment analysis, it was found that the mDNBR calculated using CETOP-D with the WSSV correlation was greater than the limit of 1.38. However, for the pre-trip phase of the MSLB inside containment, CETOP-D with the WSSV correlation yielded an mDNBR below the limit. In order to prevent the MSLB outside containment event from violating the mDNBR limit, APS has indicated that two options may be used for a given reload analysis: a detailed VIPRE-W subchannel modeling or, if necessary, adjusting the TS LCO applicable to initial thermal margin. These methods provide additional means of ensuring that a core design can be selected that conservatively precludes fuel failures for this Condition IV event. Therefore, the NRC staff concludes that the MSLB event can be adequately analyzed and the results will be confirmed to adhere to acceptable limits.

The Mode 3 Post-Trip MSLB event was also reanalyzed using the current UFSAR methodology. Only the most limiting case, with respect to RCS cold leg temperature and SG tube plugging, was analyzed and included a concurrent LOP. This case bounds all other variants and the results of the analysis demonstrate that the DNBR and LHGR limits are satisfied and transient system response is unchanged. Therefore, the NRC staff concludes that the Mode 3 Post-Trip MSLB event has been adequately analyzed, and the results conservatively demonstrate that SAFDLs are not violated for this Condition IV event.

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[8] The "pre-" and "post-trip" terminology denotes that analysis of the steam line break event has been split into two discrete phases, pre- and post-trip.

Decrease in Secondary System Heat Removal Analyses (UFSAR Section 15.2)

Table 1 of this SE illustratres that many of the UFSAR Section 15.2 analyses are bounded by the LOCV event. This event was analyzed using the current UFSAR methodology and the results demonstrate that the overall transient system response remains unchanged. Additionally, the CETOP-D WSSV analysis yielded an mDNBR of 1.80, well above the limit. Therefore, the NRC staff concludes that the LOCV event has been adequately analyzed and the results demonstrate that SAFDLs are not violated.

The remaining unbounded UFSAR Section 15.2 analysis covers the FW System Pipe Break (FWSPB) event, which is evaluated consistent with the current UFSAR methodology. The analysis results demonstrate that the overall transient system response remains unchanged. The CETOP-D WSSV analysis yielded an mDNBR of 1.40, which is above the limit of 1.38. Therefore, the NRC staff concludes that the FWSPB event has been adequately analyzed and the results demonstrate that SAFDLs are not violated.

Decrease in Reactor Coolant Flow Rate Analyses (UFSAR Section 15.3)

Three of the four events in UFSAR Section 15.3 are applicable to PWRs. The first is the total TLOF event, which partially bounds the UFSAR Section 15.2.6 event (i.e., loss of non-emergency ac power to station auxiliaries). UFSAR Appendix 15D analysis methods were used to evaluate the TLOF event. This includes replacing the COAST code described in the original approved methodology (CENPD-183-A) with CENTS, as discussed above in Section 3.3.12.1 of this SE. Additionally, a revised RCP coastdown curve is used to account for pressure drop effects introduced by NGF. The resulting mDNBR from CETOP-D with WSSV is greater than the limit of 1.38. Additionally, the LHGR has been verified to remain below the limit. Therefore, the NRC staff concludes that the TLOF event has been adequately analyzed and the results demonstrate that SAFDLs are not violated.

Since the flow controller malfunction causing coastdown event is specific to boiling-water reactors (BWRs) and is inapplicable to PVNGS, the next Chapter 15.3 event analyzed is the single RCP rotor seizure with LOP. Analysis of this event is described in UFSAR Section 15.3.3. The flow coastdown curve used as part of this analysis has been updated to reflect the impacts of transition to NGF. Ultimately, the conclusions for this limiting fault remain unchanged and bounded by the UFSAR Section 15.3.4 analysis. Therefore, the NRC staff concludes that the single RCP rotor seizure with LOP event has been adequately analyzed and it has been conservatively demonstrated that SAFDLs are not violated.

The remaining UFSAR Section 15.3 event is the RCP shaft break with LOP, which is classified as a limiting fault (Condition IV event). The analysis was performed consistent with current UFSAR methodology and resulted in no change to the overall transient system response, aside from the effect on the RCS flow coastdown curve mentioned previously. Consistent with the prior UFSAR analysis, the mDNBR limit is not met for the event and a fuel failure analysis is required. Similar to the previous fuel failure analysis for the IOSGADV+LOP and IMSF+LOP events, it was predicted that time in DNB would be less than 4 seconds and therefore it has been concluded that DNB propagation will not occur. The calculated fuel failures for the event remain bounded by the current UFSAR analysis. Therefore, the NRC staff concludes that the RCP shaft break with LOP event has been adequately analyzed and that resultant fuel failures have been adequately predicted for this Condition IV event.

Reactivity and Power Distribution Anomalies Analyses (UFSAR Section 15.4)

Of the eight UFSAR Section 15.4 events, two were determined to be unaffected by transition to NGF and another is specific to BWRs. The remaining five events were analyzed. The first two of these are the Uncontrolled CEA Withdrawals from either Subcritical or Low Power (e.g., HZP) or from HFP. These two analyses were both performed consistent with current UFSAR methodologies and neither resulted in a change to overall transient system response. For both events, it was shown using CETOP-D with WSSV that the mDNBR was greater than the limit. Therefore, the NRC staff concludes that both Uncontrolled CEA Withdrawal events have been adequately analyzed and the results demonstrated that SAFDLs are not violated.

The third of the analyzed events is the Single Full-Length CEA Drop event. In this event, the COLR Figure 3.1.5-1 allows the plant to maintain the initial power for 15 minutes, after which the plant must reduce power following the power reduction curve. During this 15-minute period, the thermal margin is protected by the initial required over-power margin, which is reserved in the COLSS. The reduction of power at the 15-minute mark, following the power reduction curve, ensures that mDNBR values remain above the limits. The updated analysis was performed and the results confirm that the above statements remain valid for transition to NGF and that mDNBR and LHGR limits remain satisfied. Therefore, the NRC staff concludes that the Single Full-Length CEA Drop event has been adequately analyzed and it has been demonstrated that SAFDLs are not violated.

The fourth of the analyzed Chapter 15.4 events is the Inadvertent Loading of a Fuel Assembly into the Improper Position (misload) event. Analysis of this event requires that the worst-case undetectable fuel misload be identified. Detectable misloads prevent reactor startup and therefore are not considered in the analysis. APS originally requested the use of an unapproved statistical methodology for analyzing the worst-case undetectable misload. However, the NRC staff raised questions concerning the adequacy of this methodology during two audits (References 30 and 31) and subsequently issued RAI 8 regarding the methodology. In response to RAI 8 (Reference 2), APS explained that the methodology described in Section 7.4.7 of Attachments 7 and 8 to the submittal would be revised to reflect the use of the currently approved method rather than the unapproved statistical approach.

The results of the analysis with current licensing-basis methods predicted limited fuel failure, which, according to the PVNGS licensing basis, is permitted for an infrequent event (Condition III event), provided the offsite radiological dose consequences remain below 10 percent of the 10 CFR Part 100 limits. This dose criterion is also applicable to two other analyzed infrequent events, the IOSGADV+LOP and the AOO from the SAFDL[9] events. The predicted doses for the misload event were determined to be lower than or the same as those predicted for the AOO from the SAFDL event and therefore the applicable dose limit is satisfied. Therefore, the NRC staff finds the misload event to be adequately analyzed and considers the results acceptable.

The final UFSAR Section 15.4 event is the CEA ejection. The analysis for this limiting fault was performed consistent with the current UFSAR methodology and the results indicate no change to the overall transient system response. The CEA ejection event is the most limiting of all design-basis non-LOCA Chapter 15 events with respect to violation of the DNBR safety limit and resultant fuel failure levels. The 20 percent power CEA ejection is the most limiting DNB

[9] AOO from the SAFDL limit is a composite, bounding event from UFSAR Appendix 15E described further below.

thermal margin value and is therefore used to bound all other power conditions. Of the four UFSAR Chapter 15 events that require fuel failure analyses, this event predicts the largest number of fuel failures. Therefore, a DNB propagation analysis was performed. As described in CEN-372-P-A, DNB propagation is precluded if the maximum cladding strain is predicted to be less than 29.3 percent. The analysis results predicted cladding strain to be less than 1 percent for the 20 percent power case. However, the CEA ejection event requires that fuel enthalpy and rod internal pressure also be examined to ensure melt and ballooning or bursting do not occur. The analysis results were compared to these criteria as well and showed that the criteria have been met. Additionally, the fuel failure levels predicted for the event remain bounded by the current UFSAR analysis. Therefore, the NRC staff concludes that the CEA ejection event has been adequately analyzed, that appropriate fuel enthalpy and rod internal pressure criteria have been met, and that resultant fuel failures have been adequately predicted. Additionally, the NRC staff notes that the predicted fuel failures for this Condition IV event have been previously determined to be acceptable, as described in the PVNGS UFSAR.

Increase in RCS Inventory Analyses (UFSAR Section 15.5)

The UFSAR Section 15.5 events (i.e., inadvertent operation of ECCS and malfunction of the chemical and volume control system / pressurizer level control system with LOP) were determined to be unaffected by the fuel transition or bounded by other events.

Decrease in RCS Inventory Analyses (UFSAR Section 15.6)

Of the UFSAR Section 15.6 events, three are related to LOCA or radioactive material release and are discussed in other sections of this SE. The fuel transition was determined to have no impact on the SG Tube Rupture event. Additionally, the Radiological Consequences of MSL Failure Outside Containment event is specific to BWRs and is therefore inapplicable.

Thus, the only UFSAR Section 15.6 analysis discussed in this section is the Double-Ended Break of a Letdown Line Outside Containment event. The analysis was performed consistent with the current UFSAR methodology and the results show that the overall system transient response is unchanged. The CETOP-D WSSV results yield an mDNBR greater than 1.50, satisfying the limit of 1.38. Therefore, the NRC staff concludes that the event has been adequately analyzed, and it has been demonstrated that SAFDLs are not violated.

UFSAR Appendix 15D and 15E Considerations

PVNGS UFSAR Appendix 15D includes a description of the methodology used in the analysis of the TLOF (UFSAR Section 15.3.1) event. As part of the evaluation of that event, the Appendix 15D methods were evaluated to ensure they remain applicable. This evaluation indicated that only the flow coastdown curve needed to be updated and that the remainder of the analysis remained applicable. The NRC staff's review agreed with the licensee's determination and concluded that, with an updated coastdown curve, the Appendix 15D methodologies remain applicable and are appropriate for use in the TLOF analysis.

PVNGS UFSAR Appendix 15E is comprised of a composite event designed to bound, with respect to DNBR, all infrequent events, which, for PVNGS includes AOOs in combination with a single active failure. Specifically, the composite event considered in Appendix 15E involves a loss of flow from operation at the DNBR SAFDL. Due to the definition of the event, it is inherent that the DNBR SAFDL is violated, and a fuel failure analysis is required. The analysis was

performed using the current UFSAR methodology, and the results demonstrate that the overall system transient response is unchanged. The mDNBR obtained by CETOP-D with the WSSV correlation was 1.18, violating the DNBR limit as expected. The resulting fuel failure analysis is bounded by the current UFSAR analysis. Therefore, the NRC staff concludes that the Appendix 15E event has been adequately analyzed and that resultant fuel failures have been adequately predicted. Additionally, the NRC staff notes that the predicted fuel failures for this postulated event have been previously determined to be within applicable limits, as described in the PVNGS UFSAR.

3.5.7.3 Conclusion on UFSAR Chapter 15 Non-LOCA Events

The NRC staff has reviewed the UFSAR Chapter 15 analyses and appendix considerations provided by APS. For each analysis, the applicability to PVNGS, the impact of the fuel design change, and the potential for bounding with other analyses were evaluated. Each event excluded from reanalysis, in any of these ways, was appropriately justified to the NRC staff. The remaining events were analyzed using approved methodologies and considered the effects of transition to, and full-core operation of, NGF. It was demonstrated that overall system response to each of the transients was unchanged and that SAFDL limits were met for all but five of the events. These events (Sheared Shaft/Seized RCP, Pre-trip MSLB, CEA Ejection, Loss of Flow from a SAFDL, and Inadvertent Loading of a Fuel Assembly into an Improper Position) are all infrequent or limiting events, for which the PVNGS current licensing basis permits exceedance of the DNBR SAFDL. Acceptability of these event analyses against applicable dose limits has previously been demonstrated through fuel failure analyses. For PVNGS, these analyses have been updated to account for the transition to and operation with NGF and have shown that any predicted fuel failures remain bounded by the current PVNGS UFSAR. Therefore, the NRC concludes that APS has adequately demonstrated that the UFSAR Chapter 15 events have been appropriately analyzed, and the results are acceptable and support the transition to NGF at PVNGS.

3.5.8 ECCS Performance LOCA

The ECCS performance has been analyzed for LOCA conditions. The analyses are performed in accordance with an acceptable evaluation model based on conservative modeling requirements prescribed in Appendix K to 10 CFR Part 50. A number of postulated LOCAs of different sizes, locations, and other properties has been analyzed to provide assurance that the most severe postulated LOCAs are calculated. Specifically, the spectrum of postulated LOCAs has been divided into two categories based on break size: large break (LBLOCA) and small break (SBLOCA).

The results of the licensee's LOCA analyses aim to demonstrate that PVNGS satisfies the requirements of 10 CFR 50.46, thereby ensuring adequate cooling of the fuel rod cladding. Specifically, these requirements include:

- PCT remains below 2200 °F.

- Maximum local cladding oxidation (MCO) remains less than 17 percent of the cladding thickness.

- Maximum hydrogen generation remains less than 1 percent of the hypothetical hydrogen generation that would occur if all cladding metal reacted.

- A coolable geometry is always maintained.

- Long-term cooling is maintained following successful initial operation of the ECCS.

The PVNGS LOCA AORs have been reanalyzed to consider the effects of transitioning to NGF. Some changes introduced by NGF that could affect the LOCA analyses include the use of Optimized ZIRLO™ cladding, changes in cladding and pellet dimensions, and changes in hydraulic performance due to the presence of IFM grids and vaned mid-grids. A range of SBLOCAs and LBLOCAs was considered that the licensee deemed sufficient to provide assurance that the most severe case has been analyzed. Additionally, mixed-core effects were accounted for to support the transition from STD fuel to NGF. The analyses were performed following the methodologies described in WCAP-16500-P-A and its Supplement 1, which act as the core reference report for NGF. Additionally, as required by the approvals, the ECCS evaluation codes have been updated to include properties specific to the Optimized ZIRLO™ cladding material. The following sections summarize the methods for each analysis (LBLOCA and SBLOCA), the results, and the NRC staff's review thereof.

3.5.8.1 Large Break LOCA

LBLOCA Methodology and Analysis

The Westinghouse Appendix K evaluation model (EM) for LBLOCA for CE plants is referred to as the 1999 EM. This methodology has been augmented for applicability to ZIRLO™ and Optimized ZIRLO™ by CENPD-404-P-A and its Addendum 1 (References 9 and 8), respectively. Additionally, the methodology was supplemented by WCAP-16072-P-A for use with IFBA. The CE LBLOCA 1999 EM is based on a combination of codes, each of which is designed to model different stages or phenomena postulated during the LBLOCA event. Specifically, the codes and their purposes in the analysis are presented below in Table 2.

Table 2: CE LBLOCA 1999 EM Analysis Codes

Code	Analysis Purpose
CEFLASH-4A	Hydraulic analysis of RCS during blowdown
COMPERC-II	Refill/reflood of RCS and containment minimum pressure Calculation of FLECHT-based reflood heat transfer coefficient (HTC) for rod heat-up analysis.
HCROSS and PARCH	Steam cooling HTCs
STRIKIN-II	Rod heat-up analysis to determine PCT and MCO
COMZIRC	Core-wide cladding oxidation
FATES3B	Initial steady-state fuel rod conditions

A number of conservative assumptions are embedded in these analyses, some of which are necessary to conform to the requirements of Appendix K to 10 CFR Part 50.

One conservative assumption is that all safety injection flow delivered to the broken leg is lost directly to containment. Additionally, the analysis uses the most limiting initial fuel rod conditions (with respect to resultant PCT) determined by burnup-dependent sensitivity calculations with the 1999 EM, using initial conditions calculated by FATES3B. As noted above

in Section 3.5.4 of this SE, an RFO curve penalty will be applied to ensure conservative prediction of the fuel stored heat by FATES3B.

Another conservative assumption is consideration of the most limiting single failure of ECCS equipment as part of the analysis. The specific failures considered in the current LBLOCA analysis for PVNGS include:

- no failure with maximum safety injection (SI) pump flow rates
- failure of a high-pressure safety injection (HPSI) pump with minimum SI flow rates
- failure of a low-pressure safety injection (LPSI) pump with minimum SI flow rates
- failure of an emergency diesel generator (EDG) with minimum SI flow rates

A third conservative assumption the licensee cited in its submittal is that the LOCA event is postulated to occur with a concurrent LOP; therefore, all ECCS pumps must await EDG startup prior to activation. However, the conservatism of this assumption was not clear to the NRC staff, since the limiting postulated scenario among those considered was the no-failure case with maximum SI flow. In particular, the NRC staff questioned whether an earlier start of the ECCS pumps could result in a more severe containment pressure reduction that would ultimately result in a net delay in core reflood. The NRC staff observed in particular that GDC 35 and the Commission's Opinion on the 10 CFR 50.46 rulemaking (Reference 32) both stipulate that the ECCS shall be designed such that its system safety function is accomplished, not only under postulated conditions with only emergency onsite power available, but also under conditions with offsite power available. Therefore, the NRC staff issued RAI 7 to request justification for the licensee's assumption that it is unnecessary to analyze explicitly a LOCA scenario with offsite power available.

The licensee's response to RAI 7 contained two parts:

- First, the licensee provided its rationale that the information requested in RAI 7 is unnecessary. The licensee stated its view that permission for not explicitly analyzing the offsite-power-available case was granted in previous NRC staff approvals of the CE LBLOCA evaluation model. The NRC staff's review of this information did not agree with the licensee's position. In particular, the NRC staff found that a number of points made by the licensee were based on earlier versions of the CE LBLOCA evaluation model, for which a different single failure was limiting (i.e., failure of a LPSI pump). Although a partially relevant sensitivity study in a later supplement to the CE LBLOCA evaluation model was discussed in the licensee's RAI response, the NRC staff found that the ECCS flowrates considered did not bound PVNGS. Ultimately, the licensee's response did not provide adequate evidence that the NRC staff's previous reviews had determined that analysis of the onsite-power-available case was unnecessary for PVNGS.

- The second part of the licensee's response provided the information requested by RAI 7. The licensee performed a sensitivity study for PVNGS that revealed that, even under the most pessimistic set of assumptions considered, only a slight increase in PCT would occur for the onsite-power-available case that would be bounded by discretionary conservatism in the analysis. (In other sensitivity cases with less pessimistic assumptions, the licensee calculated that a PCT reduction would occur if offsite power were assumed available.)

The NRC staff considered RAI 7 to have been acceptably addressed based on the analysis described in the second part of the response.

Considering the conservatisms and potential failures discussed above, a spectrum of guillotine break sizes (i.e., arrived at by applying discharge coefficients of 0.6, 0.8, and 1.0) on the RCP discharge leg was analyzed, including assessment of transient mixed-core flow-redistribution effects on NGF assemblies. Other important plant data used in the analyses can be found in Table 8-1 in Attachment 8 to the submittal (Reference 1).

LBLOCA Results

Tables 8-2 and 8-3 in Attachment 8 to the submittal summarize the analyzed LBLOCA results for PCT and cladding oxidation.

Paragraph 50.46(b)(1) of 10 CFR specifies that the PCT shall not exceed 2200 °F. It can be seen that for all break sizes analyzed, the PCT remains below the 2200-°F limit. The NRC staff notes that a slight increase in the predicted PCT is seen in the NGF analyses relative to the AOR. Regardless, the PCT remains below the limit for all cases. Therefore, the NRC staff concludes that the criterion has been met according to a conservative methodology, and the analyses adequately support the transition to NGF at PVNGS.

Paragraph 50.46(b)(2) of 10 CFR specifies that the calculated total oxidation of the cladding shall nowhere exceed 17 percent of the total cladding thickness before oxidation. In Information Notice 98-29 (Reference 33), the NRC staff clarified that the "total oxidation" of the cladding includes pre-accident oxidation as well as the calculated accident-induced oxidation. PVNGS's LBLOCA analysis for Optimized ZIRLO™-clad NGF without IFBA calculated a bounding MCO of 15.78 percent for the 0.6 double-ended guillotine break in the pump discharge leg event. This limiting result corresponds to a very early time in life for the hot rod with the maximum stored energy in the fuel. The pre-accident oxidation for the condition would be very small, but was explicitly considered, as required. Other times in core life are bounded by this analysis, since the accident-induced oxidation amount decreases faster than the pre-accident oxidation accumulates. Additionally, for higher burnups, the RFO curve imposed by the core design process restricts the limiting rod LHGR. Therefore, the NRC staff concludes that the LBLOCA analysis adequately demonstrates that the calculated total oxidation limit specified in 10 CFR 50.46(b)(2) has been met according to a conservative methodology, and therefore supports transition to NGF at PVNGS.

Paragraph 50.46(b)(3) of 10 CFR specifies that the calculated total hydrogen generated from the cladding-water chemical reaction shall not exceed 1 percent of the hypothetical amount that would be generated if all cladding (excluding the plenum region) were to react. In this case, pre-accident effects are not included. The PVNGS LBLOCA analysis for Optimized ZIRLO™-clad NGF without IFBA calculated a bounding core-wide oxidation (CWO) of 0.813 percent for the 0.8 double-ended guillotine break in the pump discharge leg event. This bounding CWO meets the specified limit of 1 percent. Therefore, the NRC staff concludes that the CWO criterion has been met according to a conservative methodology, and the analysis adequately supports transition to NGF at PVNGS.

Paragraph 50.46(b)(4) of 10 CFR specifies that calculated changes in core geometry shall be such that the core remains amenable to cooling. This requirement is addressed in part via analysis described in the fuel assembly structural analysis sections of this SE, which covers

LOCA loadings and combined seismic and LOCA loadings (Sections 3.5.2.1 and 3.5.2.2 above). It was demonstrated that combined seismic and LOCA loads would not cause changes in the core geometry sufficient to affect core coolability. Furthermore, as noted in the Commission's Opinion on the 10 CFR 50.46 rulemaking (Reference 32), compliance with the applicable criteria for PCT and MCO supports the determination that a coolable geometry will be maintained under design-basis LOCA conditions. Therefore, the NRC staff concludes that the criterion for maintaining a coolable core geometry has been met, and the analysis adequately supports transition to NGF at PVNGS.

The final criterion, 10 CFR 50.46(b)(5), requires that the calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. This criterion, referred to as long-term cooling (LTC), is addressed below in Section 3.5.8.4.

The NRC staff concludes that the LBOCA analysis results are acceptable and support the transition to NGF at PVNGS on the bases that the analysis has been performed following approved, conservative methodologies and has adequately demonstrated that the ECCS Performance / LOCA acceptance criteria described in 10 CFR 50.46 have been met.

3.5.8.2 Small Break LOCA

SBLOCA Methodology and Analysis

The SBLOCA ECCS performance analysis used the CENPD-137P, Supplement 2, version of the Westinghouse SBLOCA EM for CE PWRs (References 34, 35, and 36), referred to herein as S2M. This methodology was augmented for applicability to ZIRLO™ and Optimized ZIRLO™ by CENPD-404-P-A and its Addendum 1 (References 9 and 8), respectively. Additionally, the methodology was supplemented by WCAP-16072-P-A for use with IFBA (Reference 14). The S2M methodology is based on a combination of codes, each of which is designed to model different stages or phenomena postulated during the SBLOCA event. Specifically, the codes and their purposes in the analysis are presented below in Table 3.

Table 3: CE S2M SBLOCA EM Analysis Codes

Code	Analysis Purpose
CEFLASH-4AS	Hydraulic analysis of RCS until time of safety injection tank (SIT) injection
COMPERC-II	Hydraulic analysis of RCS after SIT injection Only used for larger SBLOCA cases with prolonged SIT flow and significant core voiding Not run for the PVNGS limiting analysis since SIT injection was not credited
STRIKIN-II	Rod heat-up analysis to determine PCT and MCO during initial forced convection period
PARCH	Rod heat-up analysis to determine PCT and MCO during subsequent period of pool boiling
FATES3B	Initial steady-state fuel rod conditions

A limited selection of break sizes (0.065, 0.070, and 0.075 ft^2) was analyzed to bracket the limiting PVNGS break of 0.070 ft^2. The limiting break size is generally determined to be the largest SBLOCA for which the PCT occurs at approximately the same time that injection from

the SITs starts. In the case the licensee determined to be limiting (0.070 ft^2), the PCT was calculated to occur approximately 70 seconds after SIT injection would have started. However, SIT injection is conservatively not credited in the analysis, so the hot rod heating transient is terminated solely by the injection from one HPSI pump.

The COMZIRC code is not used in the SBLOCA analysis since CWO is conservatively represented as the rod-average cladding oxidation of the hot rod, as determined by STRIKIN-II and PARCH. Further conservatisms are introduced in the analysis by using fuel rod conditions that result in the maximum initial stored energy in the fuel, and assuming the failure of an EDG, which ultimately causes the loss of one HPSI and one LPSI pump. Based on this single failure and the assumption that flow to the broken loop is completely spilled to the containment, only 73 percent of the flow from a single HPSI pump is credited in the analysis. LPSI flow is not credited at all, since the RCS pressure during the analyzed event remains above the LPSI pump shutoff head.

The licensee deemed mixed-core analysis of NGF and STD assemblies unnecessary since the resulting changes in core hydraulic losses have a negligible effect on the outcome of the SBLOCA analysis. The NRC staff agreed with the licensee's reasoning, because, unlike the rapid, inertially dominated LBLOCA event, SBLOCA behavior is governed largely by gravitational forces and gradually varying phenomena such as inventory loss and pressure differences.

SBLOCA Results

Tables 8-7 and 8-10 in Attachment 8 to the submittal (Reference 1) summarized the analyzed SBLOCA results for PCT and oxidation.

Paragraph 50.46(b)(1) of 10 CFR specifies that the PCT shall not exceed 2200 °F. Although a slight increase in the predicted PCT is seen in the NGF analyses relative to the AOR with STD fuel, the PCT remains well below the regulatory limit for all analyzed cases. Therefore, the NRC staff concludes that the criterion for PCT has been met according to a conservative methodology, and the analyses adequately support the transition to NGF at PVNGS.

Paragraph 50.46(b)(2) of 10 CFR specifies that the calculated total oxidation of the cladding shall nowhere exceed 17 percent of the total cladding thickness before oxidation. In Information Notice 98-29 (Reference 33), the NRC staff clarified that the "total oxidation" of the cladding includes pre-accident oxidation as well as the calculated accident-induced oxidation. PVNGS's SBLOCA analysis for Optimized ZIRLO™-clad NGF calculated a bounding MCO of 4.5 percent for the 0.070 ft^2 break event, showing significant margin to the 17 percent limit. Therefore, the NRC staff concludes that the SBLOCA analysis adequately demonstrates that the calculated total oxidation limit specified in 10 CFR 50.46(b)(2) has been met according to a conservative methodology, and supports transition to NGF at PVNGS.

Paragraph 50.46(b)(3) of 10 CFR specifies that the calculated total hydrogen generated from the cladding-water chemical reaction shall not exceed 1 percent of the hypothetical amount that would be generated if all cladding (excluding the plenum region) were to react. In this case, pre-accident effects are not included. The PVNGS SBLOCA analysis for Optimized ZIRLO™-clad NGF conservatively represented CWO as the rod-average cladding oxidation of the hot rod. Despite this conservatism, the SBLOCA results show significant margin to the 1 percent limit, with a bounding CWO of 0.33 percent for the limiting small break. Therefore, the

NRC staff concludes that the CWO criterion has been met according to a conservative methodology, and the analysis adequately supports transition to NGF at PVNGS.

Paragraph 50.46(b)(4) of 10 CFR specifies that calculated changes in core geometry shall be such that the core remains amenable to cooling. This requirement is addressed in part via analysis described in the fuel assembly structural analysis sections of this SE, which covers LOCA loadings and combined seismic and LOCA loadings (Sections 3.5.2.1 and 3.5.2.2 above). It was demonstrated that combined seismic and LOCA loads would not cause changes in the core geometry sufficient to affect core coolability. Furthermore, as noted in the Commission's Opinion on the 10 CFR 50.46 rulemaking (Reference 32), compliance with the applicable criteria for PCT and MCO supports the determination that a coolable geometry will be maintained under design-basis LOCA conditions. Therefore, the NRC staff concludes that the criterion for maintaining a coolable core geometry has been met, and the analysis adequately supports transition to NGF at PVNGS.

The final criterion, 10 CFR 50.46(b)(5), requires that the calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. This criterion, referred to as LTC, is addressed below in Section 3.5.8.4.

The NRC staff concludes that the SBLOCA analysis results are acceptable and support the transition to NGF at PVNGS on the bases that the analysis has been performed following approved, conservative methodologies and has adequately demonstrated that the ECCS Performance / LOCA acceptance criteria described in 10 CFR 50.46 have been met.

3.5.8.3 Other LOCA Considerations

Two additional small-LOCA scenarios are analyzed in the PVNGS licensing basis. These are the Inadvertent Opening of a Pressurizer Safety Valve event and the Bottom Mounted Instrumentation Break event. The licensee determined that neither analysis would be significantly impacted by the transition to NGF. Because a rationale for this conclusion was not provided in the licensee's submittal, the NRC staff reviewed Section 6.3.3.3 of the PVNGS UFSAR for further information regarding these two small-LOCA scenarios.

Regarding the postulated inadvertent opening of a pressurizer safety valve, the NRC staff found that (1) the peak cladding temperature for this event occurs during the blowdown phase and (2) the results for this event are far from being limiting, both among all analyzed LOCA scenarios (i.e., including large breaks) as well as among the subcategory of small-LOCA scenarios. Although the licensee did not provide sufficient information to convince the NRC staff that the calculated result for this scenario shown in Table 6.3.3.3-5 of the PVNGS UFSAR would necessarily bound the proposed configuration with NGF, it is clear that this case is far from being limiting. Therefore, in light of the substantial margin available, the NRC staff concludes that a reanalysis of this scenario is not necessary to demonstrate that the requirements of 10 CFR 50.46 are satisfied for the proposed transition to NGF.

Regarding the postulated 0.003-ft^2 rupture of a bottom-mounted instrument tube, according to Section 6.3.3.3.7 of the PVNGS UFSAR, the core will remain covered and cooled during this event. The NRC staff did not observe a specific calculation of PCT for this event in the UFSAR; in particular, it is not clear whether a blowdown peak in the cladding temperature would occur in this scenario due to a departure from nucleate boiling. In any event, for similar reasons to those

discussed above regarding the inadvertent opening of a pressurizer safety valve, the NRC staff does not consider analysis of the bottom-mounted instrument tube break to be necessary to support the transition to NGF.

Therefore, in both cases described above (i.e., inadvertent opening of a pressurizer safety valve and the bottom-mounted instrument tube rupture), fuel transition effects are not expected significantly to affect these analyses that have been determined to be substantially non-limiting according to the current AOR.

Transition mixed-core effects for the LBLOCA and SBLOCA analyses were also considered. As part of these analyses, the effects of differential-pressure induced flow redistribution (see Section 3.3.1.6 above) and the reduced core flow area (relative to a full core of NGF assemblies) were examined. For physical reasons noted above in Section 3.5.8.2, mixed-core effects were found to have negligible impact on SBLOCA analyses. For LBLOCA analyses, the calculated mixed-core results showed a slight improvement in thermal-hydraulic performance relative to the full-core NGF results. The slight decrease in core pressure drop for the mixed-core cases led to increases in reflood rates and slight reductions to PCT, MCO, and CWO. As such, it was determined that the full-core NGF analyses bound the transition core scenarios. Therefore, the NRC staff has concluded that the transition core effects for the analyses have been adequately considered and are bounded by the full-core NGF LOCA analyses.

3.5.8.4 Post-LOCA Long-Term Cooling

The LTC EM consists of a boric acid precipitation analysis and a decay heat removal analysis. The precipitation analysis demonstrates that the maximum boric acid concentration remains below the solubility limit, ensuring that precipitation does not occur. The decay heat removal analysis demonstrates that that the decay heat in the fuel can be removed for the long term and that the core remains covered with two-phase liquid, ensuring that temperatures remain acceptably low. Both analyses were performed in accordance with the Westinghouse post-LOCA LTC EM for CE PWRs, CENPD-254-P-A, with one exception. In particular, the EM defined in CENPD-254-P-A (Reference 37) consists of four codes:

- CEPAC
- NATFLOW
- CELDA
- BORON

The licensee has proposed to replace the BORON code with the SKBOR code, and all of the boric acid precipitation results presented in the LAR were performed with SKBOR.

Ultimately, the results of the LTC analyses are used to create an LTC plan that plant operators would use to mitigate a LOCA event.

Boric Acid Precipitation Analysis

As a result of concerns with the CENPD-254-P-A methodology that were identified in 2005, the NRC has formally suspended this topical report (References 38 and 39). In suspending CENPD-254-P-A, the NRC staff observed that, although all aspects of the topical report are no longer approved, the overall framework and general approach of CENPD-254-P-A remain valid.

In particular, the NRC staff determined that, until a supplement to CENPD-254-P-A is issued, licensees should address four main items with respect to post-LOCA LTC boric acid precipitation analyses (Reference 38):

- The assumed mixing volume within the reactor vessel should be justified, and the void fraction should be accounted for when computing the boric acid concentration.

- The time-dependent behavior of the mixing volume should be considered, and the boric acid precipitation analysis should account for the variation in the mixing region while also considering the pressure drop in the RCS loops. The resulting boric acid concentration should remain below the solubility limit.

- The solubility limit should be justified, especially if credit is taken for containment pressures greater than 14.7 pounds per square inch absolute (psia) or for additives in the sump water.

- For Appendix K-based models, the decay heat multiplier must be 1.2 at all times.

In as much as the LTC evaluation model in CENPD-254-P-A has, up to the present time, not been revised to address the concerns identified by the NRC staff in 2005, it is necessary to address these issues for PVNGS on a plant-specific basis in this SE.

As noted previously, the boric acid precipitation analysis for PVNGS was performed using the SKBOR code in lieu of the BORON code described in CENPD-254-P-A. However, the licensee's submittal did not provide adequate description of the SKBOR code, nor did it provide a basis for the acceptability of the code. Therefore, the NRC staff issued RAI 5 requesting further information concerning the SKBOR code and its implementation for the PVNGS plant-specific analysis.

In response to RAI 5 (Reference 2), the licensee provided a 27-page proprietary technical description for the SKBOR code. The technical description covers key conservation and boron tracking equations, a simplified void model, simplified nodal diagrams, and a description of the code inputs and outputs.

As implemented in the analysis for Palo Verde, the SKBOR code considered only two basic volumes, one representing the effective vessel mixing volume (denoted within SKBOR as "CORE"), and the other representing the remaining system inventory (i.e., "SUMP"). The code allows the boric acid concentration (BAC) for makeup water to be specified independently between the injection and recirculation phases of the LOCA event. Specifically, the refueling water tank concentration of 4400 ppm is used during the injection phase, and the computed sump concentration is used during the recirculation phase.

Mass conservation is ensured by SKBOR. Vapor boiled off from the core is returned to the sump as condensate containing no boric acid. While this model is relatively simple, from a physical perspective, the containment atmosphere would become saturated with water vapor very quickly following a LOCA. Thus, the model treats the effects of the liquid mass lost as vapor to the containment atmosphere as a bias that is not tracked over time during the event. The licensee allowed for 40,000 pounds (lbm) of water vapor generation to saturate the containment atmosphere, which the NRC staff checked and found reasonable for an assumed containment pressure of 20 psia. Since the containment atmosphere is assumed to remain at a

constant pressure and an allowance has been taken for fully saturating this volume with water vapor, the NRC staff found it reasonable for the simplified model to assume that the additional vapor generated by core boil-off is condensed within the containment and returned to the SUMP volume.

The effective vessel mixing volume in SKBOR (i.e., the CORE volume) includes 50 percent of the lower plenum volume. The inclusion of this volume is intended to reflect that some amount of mixing will occur during LTC between the coolant in the lower plenum and that in the core. As noted in the NRC staff's SE for the Extended Power Uprate of Waterford 3 (Reference 40), the NRC staff found the assumption of 50 percent mixing in the lower plenum to be acceptable based on comparison to testing at the BACCHUS facility. For similar reasons, the NRC staff finds this assumption acceptable for PVNGS.

The boric acid solubility limit used, 32 percent by weight, is based on an RCS pressure of 20 psia. The NRC staff checked this value against published data and found it acceptable. The licensee's analyses have found that containment pressure does not drop below 20 psia for at least 24 hours following a LBLOCA. In its review, the NRC staff further noted that current licensing basis calculations for PVNGS credit 20 psia of containment pressure in its boric acid precipitation calculations. Therefore, no additional credit for containment accident pressure is associated with this LAR. Additionally, in determining the boric acid precipitation limit, no credit for additives in the containment sump water was assumed.

As part of RAI 5, the NRC staff further requested information concerning important post-processing steps associated with the SKBOR methodology, including the determination of the core void fraction, calculation of RCS loop pressure drop, and determination that hot leg entrainment criteria are satisfied. The licensee provided further information regarding these topics as follows:

- The presence of significant voiding in the reactor core following the limiting cold-leg break would reduce the available mass of water in the reactor vessel; hence, void fraction must be accounted for to estimate accurately the available margin to boric acid precipitation. Voiding in the CORE volume in SKBOR is determined using the Yeh correlation. Voiding in the mixing volume region above the core is determined using the core exit void fraction adjusted for differences in flow area. Thus, core void fraction was accounted for in the PVNGS analyses through use of the Yeh correlation.

- Consideration of RCS loop differential pressure is necessary to ensure that the mixing volume has not been overestimated. Excessive loop differential pressure could depress the two-phase mixture level in the reactor core, thereby reducing the mass of water available to dissolve boric acid. SKBOR performs a determination of loop differential pressure to ensure that there is sufficient margin in the calculated mixing volume to account for the impacts of loop pressure drop. In particular, a post-processing check is used to verify that the loop differential pressure is less than or equal to the difference in static head between the downcomer and core. The loop differential pressure was confirmed to satisfy its acceptance criterion for the Palo Verde analysis.

- A calculation of the potential for liquid entrainment in the RCS hot legs is performed to determine the earliest time at which simultaneous hot- and cold-leg injection may be initiated. The liquid film entrainment threshold in the RCS hot legs is evaluated by

applying both the Wallis-Steen liquid entrainment onset criterion and the Ishii-Grolmes inception criteria. As noted by the licensee, these correlations are valid under conditions where the liquid volume fraction is low and flow of the liquid phase is in the turbulent regime. The potential for entrainment of ECCS coolant injected into the hot legs by steam exiting the core was demonstrated to fall below the entrainment criterion 1 hour post-LOCA using the Ishii-Grolmes criteria and a steaming rate of 72.43 pounds per second (lbm/s). This calculation was performed using Appendix K decay heat models with a multiplier of 1.2.

Boric Acid Precipitation Analysis Results

The boric acid precipitation analysis demonstrates that a minimum flow rate of 415 gallons per minute to both the hot and cold legs of the RCS, initiated at 3 hours following the limiting, large cold-leg break, will maintain the BAC below the solubility limit for both NGF and STD fuel designs. With no hot side injection flow, the licensee calculated that the BAC would reach the solubility limit at 3.4 hours.

The NRC staff's review of the boric acid precipitation calculation for Palo Verde found that the four items associated with the suspension of CENPD-254-P-A that licensees should address regarding boric acid precipitation analysis have been met. Specifically:

- The mixing volume accounts for void fraction in the reactor vessel and only credits 50 percent of the lower plenum volume.

- The time-dependent mixing volume accounts for loop pressure drop.

- The BAC limit (32 percent by weight) is justified based on containment pressure analyses and does not take credit for additives.

- An Appendix K model was used with a multiplier of 1.2 on decay heat.

The NRC staff also found the licensee's practice of assuming no credit for boric acid transport out of the core via entrainment to be conservative and appropriate.

The NRC staff further performed a limited-scope confirmatory analysis using the TRACE code for comparison against the results computed by the licensee using the SKBOR code. The NRC staff's primary motivations for performing such a confirmatory analysis were (1) the SKBOR code has not been generically reviewed and approved by the NRC staff and (2) the SKBOR code relies upon simplified modeling approaches (e.g., two-node system). The TRACE model generated by the NRC staff for confirmatory calculations was built with the intent of maximizing consistency with the SKBOR model used by the licensee. However, because the native models in TRACE differ significantly from those used in SKBOR, some important nodalization differences (e.g., increased noding resolution, more complete and realistic vessel geometry) were necessary to ensure overall consistency of the simulated physical behavior (e.g., avoiding excessive carryover, establishing representative containment / sump boundary conditions). Similarly to SKBOR, modeling of loop differential pressure was not included in the NRC staff's simplified TRACE model.

The results of the NRC staff's confirmatory calculations with TRACE indicated that the licensee's SKBOR calculations predicted the time to boric acid precipitation conservatively.

However, the NRC staff did observe differences between the TRACE and SKBOR results, some of which can be attributed to the different modeling practices (e.g., regarding core void fraction, intermixing between different nodal volumes). In light of these differences, the NRC staff considered inclusion of the conservatisms discussed above appropriate for assuring the conservatism of the results predicted by the simplified SKBOR model.

Based upon the review described above, the NRC staff found that an acceptable boric acid precipitation analysis has been performed for PVNGS, which has demonstrated that, with appropriate operator actions according to the LTC plan, the BAC would be maintained below the solubility limit. Additionally, the solubility limit has been justified and the four items associated with the suspension of CENPD-254-P-A have been acceptably addressed. Therefore, the NRC staff concludes that APS has adequately shown that boric acid precipitation can be prevented during a postulated LOCA event at PVNGS for both STD fuel and NGF.

Post-LOCA Decay Heat Removal

The decay heat removal analysis is performed using CEPAC, NATFLOW, and CELDA, following the approach outlined in CENPD-254-P-A (Reference 37). This analysis categorizes each analyzed break according to two criteria:

- The first criterion identifies breaks small enough such that the RCS could be refilled post-LOCA. In this case, once the RCS has been cooled down to the shutdown cooling (SDC) entry temperature, SDC could be aligned to remove long-term decay heat and prevent boric acid precipitation.

- The second criterion identifies breaks large enough that the break flow and simultaneous hot- and cold-leg injection could remove decay heat and prevent boric acid precipitation in the long-term.

CEPAC is used to calculate the SG cooldown transient, providing input for the CELDA and NATFLOW analyses. Additionally, the amount of FW used during the cooldown is calculated. NATFLOW is used to calculate the natural circulation flow rate and RCS temperatures under natural circulation following a LOCA. The results are used to determine when the RCS is cooled down to the SDC entry temperature. CELDA is used to calculate the long-term T-H response of the RCS following a LOCA for a series of break sizes and provides RCS inventory output that is significant in justifying the LTC plan.

Since the LTC methodology implementation defines a subset of the spectrum of break sizes that satisfy both small-break and large-break LTC strategies (i.e., an overlap region for the two criteria bulletized above), this analytical approach is consistent with LOCA emergency operating procedures and the underlying philosophy of CENPD-254-P-A. Although issues identified with the methodology for determining boric acid precipitation resulted in the formal suspension of CENPD-254-P-A, the long-term core cooling analysis is a part of the overall framework of the topical report that the NRC staff still considers valid. Using this approach the licensee demonstrated that decay heat can be removed in the long term for any size LOCA, and that the operator can be expected to correctly identify and initiate the appropriate means of LTC. Therefore, the NRC staff concludes that the decay heat removal analysis, in combination with the boric acid precipitation analysis, ensures that the 10 CFR 50.46(b)(5) criterion for LTC has been adequately met for PVNGS. It should be noted, however, that the CENPD-254-P-A

methodology does not account for the impacts of post-LOCA debris that are associated with Generic Safety Issue 191 (GSI-191). As discussed below in Section 3.5.8.5, the licensee currently remains in the process of addressing the impacts of post-LOCA debris.

The results of the decay heat removal analysis and the LTC plan derived from these results are discussed more specifically in the following section.

Decay Heat Removal Analysis Results and LTC Plan

The LTC plan is created using a 6-step process that incorporates the results of the decay heat removal analysis and satisfies the assumptions and results of the boric acid precipitation analysis. These steps are summarized below.

- CEPAC examines the SG cooldown and associated condensate storage tank (CST) inventory depletion. SG cooldown is initiated at 2 hours post-LOCA at 75 °F per hour. Upon CST depletion (9.25 hours post-LOCA), the SGs are considered to be lost as heat sinks.

- NATFLOW determines the time to reach RCS SDC entry temperature (7.7 hours). Combined with the above step, this shows that SGs are available as heat sinks for at least an hour after SDC entry temperature is established, allowing sufficient time to initiate SDC.

- For the above results, the LTC plan decision time is selected to be 8 hours. At this time, the operator makes the determination of whether to initiate SDC or continue simultaneous hot and cold leg injection and has over an hour to implement the selected procedure before the SGs are lost as heat sinks.

- CELDA determines the largest break for which the RCS refills before the decision time. For all break sizes up to and including the determined break (0.0290 ft^2), it is shown that the RCS has sufficient inventory to initiate SDC.

- CELDA determines the smallest break for which the break flow rate serves as an adequate means of RCS heat removal after the SGs are lost as heat sinks. For the determined break (0.0110 ft^2) and larger, it is confirmed that the core remains covered with two-phase liquid after the time the SGs are lost as heat sinks. Combined with the above step, this shows that for breaks between 0.0110 ft^2 and 0.0290 ft^2, either LTC strategy can be applied successfully. Breaks outside this range should apply the appropriate strategy.

- From these results, the LTC plan decision pressure is determined. This is the pressure criterion against which the operator compares the indicated pressurizer pressure at the decision time in order to select the appropriate course of action. This decision pressure has been selected to be 200 psia, based on the resultant pressures from the range of overlap in break sizes mentioned in the previous step. This decision pressure ensures that an adequate action will be selected by the operator regardless of the indicated pressurizer pressure, accounting for a measurement uncertainty of 100 psia.

The LTC plan has been formulated according to approved methodologies and shows that the operator has adequate time and means of making the appropriate decision and implementing it

before the SGs are lost as heat sinks. Therefore, the NRC staff concludes that the LTC plan is acceptable for use at PVNGS with NGF.

LTC Conclusion

It has been demonstrated that boric acid precipitation will not occur and that decay heat can be removed for any LOCA break size while keeping the core covered, thereby ensuring that core temperatures would remain acceptably low during LTC. Based on the results of these analyses, an acceptable LTC plan has been developed to ensure that operators have sufficient means and time to implement an appropriate LTC strategy. The plan includes an overlap region where both small-break and large-break LTC strategies can be successfully implemented, and contains an allowance for instrument uncertainty. As such, the NRC staff has confidence that an appropriate decision will be made regardless of the indicated pressure. Additionally, it was demonstrated that auxiliary feedwater flow would be sufficient to support the LTC plan. Therefore, the NRC staff concludes that APS has adequately demonstrated PVNGS's LTC capability for both STD fuel and NGF, satisfying the requirements of 10 CFR 50.46(b)(5).

3.5.8.5 GSI-191

Concerns related to the collection of debris on recirculation sump strainers, ingestion within the RCS, and subsequent accumulation on fuel assemblies in the reactor core have been grouped under GSI-191. Although all PWRs have made significant modifications to enhance LTC performance in the presence of post-LOCA debris (e.g., installation of larger ECCS recirculation strainers, modifications to thermal insulation inside containment), many PWR licensees, including APS, have not fully resolved all issues associated with GSI-191.

With specific regard to the licensee's proposed transition to NGF, fuel assembly testing in support of the resolution of GSI-191 was performed by the PWROG, as detailed in WCAP-17057-P, Revision 1 (Reference 41), as part of a comprehensive evaluation of the issue discussed more fully in WCAP-16793-NP-A (Reference 42). The testing addressed the collection of debris and the potential effects of blockage on selected fuel assembly designs being manufactured at the time the tests were conducted. The licensee's LAR stated that the NGF design is less limiting than those fuel designs included in the PWROG's test program. In assessing this statement, the NRC staff reviewed proprietary information concerning the NGF design in WCAP-16500-P-A (Reference 5) and the PWROG testing program in WCAP-17507-P, Revision 1. The NRC staff's review of this information found that the NGF design is substantially similar to the CE fuel design explicitly considered in WCAP-17057-P, Revision 1, **[[ ]]**. Therefore, the NRC staff agrees that the conclusions made in WCAP-17057-P regarding the tested CE fuel design should extend to the NGF the licensee has proposed to load at PVNGS.

The licensee is continuing its efforts to resolve concerns associated with GSI-191 in accordance with the policy outlined by the Commission in its Staff Requirements Memorandum to SECY-12-0093 (References 43 and 44). Final resolution of GSI-191 issues is necessary to obtain adequate confidence in the licensee's LTC plan in the presence of post-LOCA debris. The NRC staff expects the licensee to reflect the change in fuel design as part of its final resolution of GSI-191 issues.

3.5.8.6 ECCS Performance / LOCA Conclusion

The NRC staff has reviewed the ECCS performance / LOCA analyses and concluded that the analyses have been appropriately performed and the results demonstrate adequate performance of the ECCS, considering the transition to and operation with full cores of NGF. Therefore, the NRC staff finds the use of NGF at PVNGS appropriate and justified.

3.5.9 Containment Response

Introduction of NGF at PVNGS will increase the pressure drop across the core. Ultimately, this results in a slight reduction to RCS flowrate, essentially increasing the RCS hot-leg temperature. RCS cold-leg temperature and operating pressure are unchanged, but the increased hot-leg temperature results in a slightly higher SG operating pressure. 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, mass and energy (M&E) release AORs have still been examined to ensure continued applicability at PVNGS.

In RAI 9.a, the NRC requested quantitative results to confirm the AOR M&E releases for the LOCA event remain bounding for the long-term sump temperature response. In response to RAI 9.a, the licensee provided the NRC staff with the fuel parameters that are the focus of evaluation: core average linear heat rate, pellet and cladding geometry, centerline temperature, decay heat, and metal-water reaction. The response stated that for these parameters, the AOR analysis is bounding for NGF fuel. For the core average linear heat rate, the AOR uses a value of 5.925 kilowatt per foot (kW/ft), while the maximum value for a scaled core power of 4070 MWt is 5.735 kW/ft, demonstrating the AOR is bounding.

For the pellet and cladding geometry, the primary dimensions of pellet outside radius, cladding inside radius, and cladding outside radius were all reduced for NGF, resulting in less heat transfer from the fuel to the coolant than would result in the AOR geometry. Higher heat transfer is conservative for the M&E release analyses and therefore, the AOR is bounding.

For the centerline temperature, the licensee stated that since the heat transfer is driven by the centerline temperature and the AOR centerline temperatures are based on Erbia fuel composition plus a 50 °F addition, AOR centerline temperatures bound all the NGF centerline temperatures.

For the decay heat, the AOR assumed 102 percent thermal power, which is bounding for NGF. For the metal-water reaction, the LBLOCA M&E analysis is biased to extract the most energy from the fuel to heat the water to produce steam. This results in lower fuel clad temperatures and negligible metal-water reaction.

For the short term 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 reactor coolant system pressure and temperature) with the NGF transition.

The NRC requested PVNGS to explain how the AOR FW temperature produces bounding results for the M&E release analysis for the Postulated Secondary System Pipe Ruptures Inside Containment in RAI 9.b. In response to RAI 9.b, the licensee stated that the AOR FW temperature used in the evaluation was 450 °F, while the full-power FW temperature for NGF is 448 °F. A higher FW temperature results in more energy transferred to the steam generator secondary side causing a higher containment pressure and temperature during the blowdown phase. The higher FW temperature used in the AOR analysis establishes the AOR as bounding.

In RAI 9.c, the NRC requested the licensee justify quantitatively that the NGF operating conditions will remain bounded by the AOR main steam line break containment M&E source energy. In response to RAI 9.c, the licensee described that a hand calculation was completed to determine the total fuel-region energy for both the AOR and NGF fuel by using the average temperature, fuel density, specific heat, and volume. The calculation determined that the AOR has a higher total fuel-region energy than NGF, 34.67 mega British thermal units (MBtu) versus 34.53 MBtu. The higher total fuel-region energy results in higher-energy steam release during the blowdown phase and leads to higher pressure and temperature in the containment, which is a conservative response.

To summarize the discussion above, an evaluation of NGF's impact on the LOCA M&E release AOR was performed. 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 AOR remains applicable. Additionally, an evaluation of NGF's impact on the M&E release for secondary system pipe rupture inside containment (i.e., MSLB) was performed. Similarly, NGF's impacts were determined to be either bounded or inconsequential; therefore, the NRC staff concludes that the secondary system pipe rupture inside containment AOR also remains applicable.

Finally, the containment sub-compartment M&E release AORs are considered. The transients for these analyses are of rather short duration (i.e., ~1 second) and do not allow enough time for significant interaction between the primary and secondary systems. Therefore, the NRC staff concludes that, since the initial conditions of the analyses remain unchanged for NGF, the AOR results remain applicable.

3.5.10 Radiological Source Term Evaluations

Source terms used for evaluating radiological consequences of postulated accidents are based on the methodology in UFSAR Chapter 15. NGF does not represent a significant departure in fundamental fuel design, and its implementation does not result in a significant change in overall reactor operation. NGF cores are still comprised of square lattices of cylindrical fuel rods encasing UO_2 fuel pellets and are to be operated under licensed power factors up to licensed burnup limits with typical operating histories. System model input changes are small and transient impacts are insignificant. Therefore, the NRC staff concluded in its SE on WCAP-16500-P-A that NGF does not introduce changes that meaningfully affect Chapter 15 source terms for LOCA, non-LOCA, or fuel-handling accidents. Additionally, the UFSAR Section 11.1 RCS specific activities and activation products for 1 percent failed fuel conditions are unaffected and remain conservative and bounding. As a result, the NRC staff considers the licensee's radiological source term evaluations applicable to NGF.

3.5.11 Radiological Accident Evaluations

For the same reasons described in Section 3.5.10 of this SE, it has been determined that all limiting offsite and control room dose consequences determined in UFSAR Chapter 15 and Section 6.4.7 remain bounding and applicable to NGF at PVNGS.

Compliance with Regulatory Guide (RG) 1.25 has been considered. Though RG 1.25 specifies a maximum fuel rod discharge pressure of 1200 pounds per square inch gauge (psig), approval of WCAP-16072-P-A concludes that fuel rod pressures up to 1500 psig would not invalidate the analysis assumptions in RG 1.25 with respect to iodine decontamination. Therefore, the NGF maximum fuel rod discharge pressure is acceptably specified as 1500 psig.

3.5.12 Setpoints

The setpoints analysis uses the Modified SCU methodology described in CEN-356(V)-P-A, Revision 01-P-A (Reference 45) as modified by WCAP-16500-P-A, Supplement 1, Revision 1 for NGF. This methodology stochastically combines uncertainties to calculate COLSS DNB power operating limits and CPCS DNBR addressable uncertainty constants. The overall uncertainty factors obtained ensure that these calculations are conservative to at least a 95/95 probability and confidence level.

3.5.13 Structural

Compliance with American Society of Mechanical Engineers (ASME) requirements for Class 1 structures, systems, and components (SSCs) has been evaluated for implementation of NGF using the methodology in UFSAR Section 3.9. The only effects introduced by implementation of NGF are associated with branch line pipe break LOCA loads, which results in minor changes to predicted loads. The AORs for the affected Class 1 SSCs have been evaluated and continue to satisfy ASME code limits.

3.5.14 Design, Systems and Components

Analyses of fluid systems, including RCS flow and hydraulic loads, have been evaluated. Results show a decrease in overall RCS flow rate, consistent with expectations for the increased pressure drop due to the introduction of NGF. Hydraulic loads for the reactor vessel and internals were calculated and have been used in the structural analyses of the systems.

3.5.15 Other Issues

TS Safety Limit 2.1.1.1 specifies the DNBR safety limit. This limit is based on the use of the CE-1 CHF correlation. The WSSV and ABB-NV CHF correlations will be used in the NGF safety and setpoint analyses. However, because of existing hardware limitations, the Core Protection Calculator (CPC) algorithm will retain the CE-1 correlation. Since the CPC thermal-hydraulic algorithm retains the CE-1 correlation, any change to the DNBR-Low trip setpoint and allowable value would introduce inconsistency between the trip setpoint and the control room monitors. To ensure that the plant operators have consistency between the trip setpoint and their control room monitors (i.e., a human factors concern), the DNBR-Low trip setpoint and allowable value will remain set at 1.34.

Credit for the improved WSSV and ABB-NV correlations (i.e., the improved DNBR limit of 1.25) is achieved by adjusting of the BERR1 uncertainty constant used in the DNBR calculations such that the CPC trip at a DNBR of 1.34 using the CE-1 correlation assures that the bounding DNBR safety limit of 1.25 for the WSSV and ABB-NV correlations will not be violated. The NRC staff observed that an analogous approach has been approved in the past for other CE PWRs with similar hardware limitations (e.g., Reference 46); likewise, the NRC staff finds this approach acceptable for PVNGS.

TS Safety Limit 2.1.1.2 provides a peak fuel centerline temperature limit and specifies that it shall be adjusted for burnable poisons per CENPD-382-P-A. However, with the implementation of IFBA burnable absorbers in WCAP-16072-P-A, it has been determined that such adjustment is not necessary. Therefore, the TS Safety Limit 2.1.1.2 is not required to be changed.

APS submitted a LAR (Reference 47) regarding an updated spent fuel pool criticality analysis, including NGF as a fuel type. The criticality analysis LAR was approved by NRC (Reference 49) with this fuel transition to support long-term operation with NGF at PVNGS.

Implementation of NGF will not affect compliance with Fukushima-related NRC orders EA-12-049 or EA-12-051.

3.6 IN 2012-09: Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength

NRC Information Notice (IN) 2012-09 (Reference 18) was issued by the NRC staff to inform addressees of operating experience involving evaluations of fuel assembly structural response to external loads and associated issues the NRC staff identified during reviews of fuel designs for design certification applications. Specifically, the SRP Section 4.2 (Reference 4) guidance suggests that grid crush strength would be most limiting at the BOL. However, operating experience regarding the effects of in-reactor service on fuel assembly component response to externally applied forces challenges this guidance and suggests that EOL may be more limiting.

The PWR Owners Group (PWROG) is working on a resolution to the issue presented in IN 2012-09 for Westinghouse and CE designs, but no solution has been reviewed or approved by the NRC staff as of the issuance of this SE. Therefore, APS proposed in its submittal that a license condition be imposed in this SE in order to allow its approval prior to dispositioning the unresolved issue presented by IN 2012-09.

However, during the NRC staff's review of the LAR, the need for imposing a license condition was discussed further with the licensee in two audits and several teleconferences. In these discussions, the licensee argued that a license condition would not, in fact, be necessary, because an existing regulatory requirement (paragraph V.(a)(2) of Appendix A to 10 CFR Part 100) already requires that PVNGS shut down following an operating basis earthquake. Furthermore, prior to resuming operations, the licensee must demonstrate to the Commission that no functional damage has occurred that would result in undue risk to the public health and safety. The NRC staff agreed with the licensee's understanding of the regulatory requirements applicable to PVNGS that are associated with the operating basis earthquake, and further observed that these requirements have been exercised in the recent past for a different plant (e.g., 2011 earthquake near Mineral, VA).

Furthermore, the NRC staff understands that grid deformation for NGF fuel may occur for vibratory ground motion between the operating basis earthquake and safe shutdown

earthquake, but that will not occur below the operating basis earthquake. As a result, under conditions where the potential exists for fuel assembly spacer grid deformation, the licensee is already subject to a regulatory requirement to shut down PVNGS and demonstrate that there is no undue risk to the public health and safety prior to the resumption of operations. The NRC staff further notes that, since the seismic/LOCA analysis for transition cores predicts the deformation of peripheral STD assemblies, such deformation should be considered in demonstrating that no functional damage has occurred.

Based on the discussion above, the NRC staff concluded that a license condition regarding the issues identified in IN 2012-09 is not necessary for the adequate protection of public health and safety.

4.0 LICENSE CONDITION

APS must adhere to the following license condition respecting the use of CE 16×16 NGF at PVNGS:

> 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.
>
> 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.

5.0 SUMMARY

Based upon the NRC staff's prior approval of NGF and the licensee's current and future compliance with the conditions and limitations specified in the approved methodologies used for safety analyses, the staff finds the proposed changes to TS 4.2.1 and TS 5.6.5.b acceptable along with the use of NGF clad with Optimized ZIRLO™ at PVNGS, Units 1, 2, and 3. The NRC staff further reviewed the analyses performed by the licensee to support NGF implementation for transition cycles and full-core conditions. These analyses included LOCA and non-LOCA events, as well as mechanical and hydraulic analyses. The NRC staff found that these analyses

support the intended transition to NGF. The NRC staff further reviewed the results of the licensee's NGF LFA program and determined that they support the proposed LAR. Therefore, based on the foregoing evaluation, the NRC staff finds the proposed LAR acceptable. The NRC staff's conclusion is contingent upon imposition of the license condition specified in Section 4.0 of this SE.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arizona State official was notified of the proposed issuance of the amendments on November 16, 2017. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on October 4, 2016 (81 FR 68469). 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., APS, letter to U.S. Nuclear Regulatory Commission, "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3 License Amendment Request and Exemption Request to Support the Implementation of Next Generation Fuel," dated July 1, 2016 (ADAMS Accession No. ML16188A336).

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Date: January 23, 2018

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 – ISSUANCE OF AMENDMENTS TO REVISE TECHNICAL SPECIFICATIONS TO SUPPORT THE IMPLEMENTATION OF NEXT GENERATION FUEL (CAC NOS. MF8076, MF8077, AND MF8078; EPID L-2016-LLA-0005) DATED JANUARY 23, 2018

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