Enclosure Attachments 10 and 11 contain PROPRIETARY information to be withheld under 10 CFR 2.390

When separated from Attachments 10 and 11, this transmittal is decontrolled.

10 CFR 50.90
10 CFR 50.12

MARIA L. LACAL
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102-08012-MLL/MDD
November 26, 2019

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

References:

1. Arizona Public Service Company (APS) letter number 102-07727, *License Amendment Request and Exemption Request to Support the Implementation of Framatome High Thermal Performance Fuel*, dated July 6, 2018, Agencywide Documents Access and Management System (ADAMS) Accession Numbers ML18187A417 (non-proprietary) and ML18187A418 (proprietary).

2. U.S. Nuclear Regulatory Commission (NRC) correspondence to APS, *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Third Regulatory Audit Plan for November 7-8, 2019, in Support of Framatome High Thermal Performance Fuel License Amendment Request and Exemption (EPID L-2018-LLA-0194 and EPID L-2018-LLE-0010)*, dated October 30, 2019, ADAMS Accession Number ML19301A905.

Dear Sir:

Subject: **Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3**
Docket Nos. STN 50-528, 50-529, and 50-530
Renewed Operating License Nos. NPF-41, NPF-51, and NPF-74
Supplemental Response to NRC Request for Additional Information Regarding License Amendment and Exemption Requests Related to the Implementation of Framatome CE16HTP Fuel

By letter dated July 6, 2018 (Reference 1), APS submitted to NRC a license amendment request (LAR) pursuant to the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR) and an exemption request pursuant to the provisions of 10 CFR 50.12, to request approval of proposed changes to the PVNGS Technical Specifications (TS) to support the implementation of Framatome CE16HTP fuel with M5® cladding and gadolinia as a burnable absorber. The Enclosure to this letter describes additional correspondence between APS and NRC related to these requests.

The NRC staff conducted a third regulatory audit in Rockville, Maryland, on November 7 and 8, 2019 (Reference 2). The Enclosure to this letter, which includes eleven (11) attachments, provides APS supplemental responses to NRC staff individual Requests for Additional Information (RAIs), based on reviews and discussions that occurred during this audit.

Enclosure Attachment 1 includes two proposed License Conditions that will support NRC staff approval of the license amendment request. The attachments also include proposed TS and TS Bases changes that are different than those originally submitted with the LAR (Reference 1). Consistent with the PVNGS *Quality Assurance Program Description*, the proposed TS changes have been reviewed and approved by the PVNGS Plant Review Board. The proposed TS changes and the APS supplemental responses to the NRC RAIs do not affect the conclusions of the 10 CFR 50.91(a) no significant hazards consideration determination provided in the LAR.

APS, Framatome, and Westinghouse affidavits have been attached to the enclosure to support withholding of proprietary information from public disclosure pursuant to 10 CFR 2.390. Attachments 10 and 11 include proprietary content and, when those attachments are separated from this transmittal, this transmittal is decontrolled.

If you have any questions about this request, please contact Michael D. DiLorenzo, Licensing Department Leader, at (623) 393-3495.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: November 26, 2019

Sincerely,

Thomas N. Weber... for

Maria L. Lacal

MLL/MDD/mg

Enclosure: Supplemental Response to NRC Request for Additional Information Regarding License Amendment and Exemption Requests Related to the Implementation of Framatome CE16HTP Fuel

cc: S. A. Morris — NRC Region IV Regional Administrator
S. P. Lingam — NRC NRR Project Manager for PVNGS
C. A. Peabody — NRC Senior Resident Inspector for PVNGS

ENCLOSURE

**Supplemental Response to NRC
Request for Additional Information
Regarding License Amendment and Exemption Requests
Related to the Implementation of
Framatome CE16HTP Fuel**

Introduction

On April 5, 2018, the U.S. Nuclear Regulatory Commission (NRC) staff held a Category 1 public and partially closed pre-submittal meeting with Arizona Public Service Company (APS) staff and APS contractors in Rockville, Maryland. The purpose of the meeting was to discuss a forthcoming license amendment request (LAR) and exemption request to support the implementation of Framatome (formerly AREVA) advanced CE16HTP fuel with M5® cladding at the Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3. The meeting notice and agenda, dated March 23, 2018, are available in the NRC's Agencywide Documents Access and Management System (ADAMS) at Accession No. ML18088B259 [Reference 1]. APS provided presentation slides for the meeting, which are at ADAMS Accession Nos. ML18088A012 (non-proprietary) and ML18103A000 (proprietary) [Reference 2]. The NRC meeting summary, dated April 18, 2018, is at ADAMS Accession No. ML18102B212 [Reference 3].

On July 6, 2018, APS submitted the LAR to the NRC pursuant to the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), and the exemption request for the use of M5® cladding pursuant to the provisions of 10 CFR 50.12. This submittal is at ADAMS Accession Nos. ML18187A417 (non-proprietary) and ML18187A418 (proprietary) [Reference 4]. The NRC staff issued seven (7) acceptance review questions on October 2, 2018, which are at ADAMS Accession Nos. ML18271A039 (non-proprietary) and ML18271A035 (proprietary) [Reference 5]. The APS responses, dated October 18, 2018, are at ADAMS Accession No. ML18296A466 (non-proprietary) [Reference 6]. The NRC accepted the LAR for review on November 13, 2018, as documented at ADAMS Accession No. ML18312A332 [Reference 7].

The first of three NRC regulatory audits occurred in Rockville, Maryland, on January 22 and 23, 2019. The NRC audit plan is at ADAMS Accession No. ML19011A108 [Reference 8]. APS audit entrance presentation slides are at ADAMS Accession No. ML19060A298 (non-proprietary) [Reference 9]. The NRC audit summary is at ADAMS Accession No. ML19218A293 [Reference 10].

On April 5, 2019, the NRC's Reactor Assessment and Human Factors Branch (IRAB) issued three (3) requests for additional information (RAIs) to APS, regarding a proposed time critical operator action to trip reactor coolant pumps (RCPs) following a small break loss of coolant accident. These RAIs are at ADAMS Accession No. ML19098A187 [Reference 11]. The APS responses, dated May 17, 2019, are at ADAMS Accession No. ML19137A118 [Reference 12].

The second NRC regulatory audit occurred at the PVNGS site in Arizona during the period of June 17-20, 2019. The NRC audit plan is at ADAMS Accession No. ML19154A469 [Reference 13] and the audit summary is at ADAMS Accession No. ML19235A256 [Reference 14]. Following this audit, the NRC staff issued sixty-four (64) additional RAIs to APS, which are at ADAMS Accession Nos. ML19234A320 (non-proprietary) and ML19234A321 (proprietary) [Reference 15]. The APS responses, dated October 4, 2019,

are at ADAMS Accession Nos. ML19277J457 (non-proprietary) and ML19277J458 (proprietary) [Reference 16].

The third NRC regulatory audit occurred in Rockville, Maryland, on November 7 and 8, 2019. The NRC audit plan is at ADAMS Accession No. ML19301A905 [Reference 17]. The audit scope included NRC staff review of faulted condition analyses and 10 CFR 50.46 emergency core cooling system (ECCS) performance analyses for Westinghouse fuel that will be co-resident with Framatome fuel in PVNGS transition cores, as well as discussions with APS regarding certain RAIs addressed in the October 4 response letter [Reference 16].

This Enclosure and its Attachments provide supplemental APS responses to the RAIs included in Reference 15. These supplemental responses address the reviews and discussions that occurred during the third regulatory audit.

APS Response to NRC RAIs

The following sections provide a high level summary of the supplemental APS responses contained herein. The following is not intended to be a comprehensive summary of all of the information contained in the Attachments to this Enclosure.

The LAR and its associated exemption request [Reference 4] have been revised by APS letters to NRC dated October 18, 2018 [Reference 6], May 17, 2019 [Reference 12], and October 4, 2019 [Reference 16], and are supplemented further by the APS responses herein.

The APS responses do not affect the conclusions of the 10 CFR 50.91(a) no significant hazards consideration determination provided in the LAR.

License Conditions (Enclosure Attachment 1)

APS proposes two License Conditions in response to NRC staff RAIs. The first License Condition serves to resolve SNPB RAI-25 by prohibiting the use of mixed fresh fuel types in a PVNGS reload core fuel batch, with the exception of Lead Test Assemblies (LTAs) as defined in Technical Specifications (TS). The second License Condition serves to resolve SNPB RAI-26 by revising an existing License Condition in Amendment No. 205 to the PVNGS Renewed Facility Operating Licenses [Reference 18], to require that Thermal Conductivity Degradation (TCD) be addressed if the Westinghouse FATES3B fuel performance code is used in the analysis of future Westinghouse fuel designs.

Enclosure Attachment 1 provides a set of clean License Condition pages to show how Appendix D of the PVNGS Renewed Facility Operating Licenses would appear following incorporation of the two proposed changes.

Technical Specifications (Enclosure Attachments 2 and 3)

The LAR [Reference 4] requested NRC approval of changes to the PVNGS TS to support the implementation of Framatome CE16HTP fuel with M5® cladding and gadolinia as a burnable absorber. The proposed TS changes included the addition of a reactor core Safety Limit (SL) for the peak fuel centerline temperature for the Framatome fuel [TS 2.1.1]; administrative changes to the description of fuel assemblies used in PVNGS reactor cores [TS 4.2.1]; and the addition of several topical reports for analytical methods to be used in the determination of core operating limits, including Framatome large break and small break loss of coolant accident evaluation models, the Framatome COPERNIC fuel performance code, the Framatome BHTP critical heat flux (CHF) correlation, and the Electric Power Research Institute (EPRI) VIPRE-01 thermal-hydraulic code [TS 5.6.5].

APS and NRC personnel discussed the proposed TS changes during the June 2019 and November 2019 regulatory audits, including how they would be applied to mixed core or transition core designs in which multiple fuel types would be co-resident in a reactor core. As a result of these conversations, APS has modified the TS changes originally proposed in the LAR [Reference 4] to clarify their intent and reduce the potential for future misinterpretation or misunderstanding.

Enclosure Attachment 2 provides mark-ups of the revised PVNGS TS pages for NRC review and approval. Enclosure Attachment 3 provides a set of clean TS pages showing how the proposed mark-ups would appear upon incorporation. These attachments are complete in that they reflect all of the TS changes proposed by APS, including revisions made as a result of regulatory audits and discussions with NRC staff. Consistent with the PVNGS *Quality Assurance Program Description*, the TS proposed changes have been reviewed and approved by the PVNGS Plant Review Board.

Technical Specification Bases (Enclosure Attachment 4)

Enclosure Attachment 4 provides mark-ups of the revised PVNGS TS Bases. The TS Bases mark-ups are provided for information only. This attachment is complete in that it reflects the TS Bases changes that APS will implement in support of the Framatome fuel LAR.

Affidavits (Enclosure Attachments 5, 6, and 7)

Enclosure Attachments 5, 6, and 7 provide, respectively, APS, Westinghouse, and Framatome affidavits pursuant to 10 CFR 2.390, to support withholding of proprietary information from public disclosure. The proprietary information to be withheld supports the APS supplemental responses to the NRC RAIs, and is contained in Enclosure Attachments 10 and 11. Non-proprietary versions of those Enclosure Attachments are provided as Enclosure Attachments 8 and 9.

It is noted that the proprietary information in Enclosure Attachment 11 is the sole property of Framatome. Enclosure Attachment 10, however, includes information that is proprietary

to all three parties. The superscripts A(PS), F(ramatome), and W(estinghouse) have been applied to the brackets in Enclosure Attachment 10 to designate the affidavit that supports withholding of information from public disclosure. It is also noted that the W superscript designation includes additional notations as required by the Westinghouse affidavit. The A superscript designation may denote information that is the property of APS, or information that is the property of either Framatome or Westinghouse but held by APS subject to contractual obligations to prevent its public disclosure.

APS Supplemental Responses (Enclosure Attachments 8, 9, 10, and 11)

The third regulatory audit plan [Reference 17] identified 14 NRC RAIs to be addressed during the November 2019 audit. The APS supplemental responses to 11 of the 14 RAIs (SNPB RAI-10, SNPB RAI-17, SNPB RAI-19, SNPB RAI-22, SNPB RAI-25, SNPB RAI-26, SNPB RAI-28, SNPB RAI-29, SNPB RAI-30, SRXB RAI-3, and CHF RAI-04) are provided in Enclosure Attachments 8 (non-proprietary) and 10 (proprietary). Supplemental responses are not provided herein for 3 of the 14 RAIs (that is, BHTP RAI-03, BHTP RAI-05, and BHTP RAI-06), because discussions with NRC staff during the November 2019 audit concluded that additional information did not need to be docketed in response to these three RAIs.

With regard to CHF RAI-04, APS notes that the supplemental response withdraws a request for NRC approval of a proposed process for implementing new combinations of CHF correlations and thermal-hydraulic codes. The supplemental response to this RAI also clarifies that the CETOP-D thermal-hydraulic code may be used in transient analyses as an alternative to the VIPRE code.

In addition to the RAIs identified in the third regulatory audit plan, Enclosure Attachments 8 (non-proprietary) and 10 (proprietary) include an APS supplemental response to SRXB RAI-5. This supplemental response modifies the last paragraph of that response to add information related to Control Element Assembly (CEA) Ejection, based on discussions during the November 2019 audit.

The APS supplemental response to SNPB RAI-19, related to small break loss of coolant accidents, is cross-referenced to Framatome material in Enclosure Attachments 9 (non-proprietary) and 11 (proprietary).

Enclosure Attachments

Attachment 1 – License Conditions
Attachment 2 – Technical Specification Page Mark-Ups, Affected Pages: 2.0-1, 4.0-1, 5.6-3, and 5.6-7
Attachment 3 – Clean Technical Specification Pages, Affected Pages: 2.0-1, 4.0-1, 5.6-3, 5.6-7, and 5.6-8
Attachment 4 – Technical Specification Bases Page Mark-Ups (Provided for Information Only), Affected Pages: B 2.1.1-3, B 2.1.1-4, and B 3.5.1-2
Attachment 5 – Affidavit from Arizona Public Service Company Submitted in Accordance with 10 CFR 2.390 to Consider Enclosure Attachment 10 as a Proprietary Document
Attachment 6 – Affidavit from Westinghouse Submitted in Accordance with 10 CFR 2.390 to Consider Enclosure Attachment 10 as a Proprietary Document
Attachment 7 – Affidavit from Framatome Submitted in Accordance with 10 CFR 2.390 to Consider Enclosure Attachments 10 and 11 as Proprietary Documents
Attachment 8 – APS Supplemental Responses to NRC Request for Additional Information [NON-PROPRIETARY VERSION]
Attachment 9 – Framatome ANP-3640Q2NP, Revision 0, Palo Verde Units 1, 2 and 3 Small Break LOCA Summary Report NRC RAI Responses [NON-PROPRIETARY VERSION]
Attachment 10 – APS Supplemental Responses to NRC Request for Additional Information [PROPRIETARY VERSION]
Attachment 11 – Framatome ANP-3640Q2P, Revision 0, Palo Verde Units 1, 2 and 3 Small Break LOCA Summary Report NRC RAI Responses [PROPRIETARY VERSION]

References

1. S. P. Lingam (NRC), *Forthcoming Partially Closed Pre-Application Meeting with Arizona Public Service Company to Discuss a License Amendment Request for Palo Verde Nuclear Generating Station, Units 1, 2, and 3 RE: Framatome (AREVA) Fuel (EPID L-2018-LRM-0020)*, March 23, 2018 [NRC ADAMS Accession No. ML18088B259]
2. T. N. Weber (APS), *NRC Pre-Submittal Meeting, Implementation of Framatome CE16HTP Fuel, Palo Verde Units 1, 2, and 3*, April 5, 2018 [NRC ADAMS Accession Nos. ML18088A012 (non-proprietary) and ML18103A000 (proprietary)]
3. S. P. Lingam (NRC), *Summary of April 5, 2018, Partially Closed Meeting with Arizona Public Service Company to Discuss Upcoming License Amendment Request Regarding Transition to Framatome Fuel for Palo Verde Nuclear Generating Station, Units 1, 2, and 3 (EPID L-2018-LRM-0020)*, April 18, 2018 [NRC ADAMS Accession No. ML18102B212]
4. Letter 102-07727-MLL/SMM from M. L. Lacal (APS) to Document Control Desk (NRC), *Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529, and 50-530, License Amendment Request and Exemption Request to Support the Implementation of Framatome High Thermal Performance Fuel*, July 6, 2018 [NRC

ADAMS Accession Nos. ML18187A417 (non-proprietary) and ML18187A418 (proprietary)]

5. Letter from M. D. Orenak (NRC) to R. S. Bement (APS), *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Supplemental Information Needed for Acceptance of Requested License Amendments and Exemptions RE: Implementation of Framatome High Thermal Performance Fuel (EPID L-2018-LLA-0194 and EPID L-2018-LLE-0010)*, October 2, 2018 [NRC ADAMS Accession Nos. ML18271A039 (non-proprietary) and ML18271A035 (proprietary)]
6. Letter 102-07807-MLL/MDD from M. L. Lacal (APS) to Document Control Desk (NRC), *Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529, and 50-530, Supplemental Information Regarding License Amendment Request and Exemption Request to Support the Implementation of Framatome High Thermal Performance Fuel*, October 18, 2018 [NRC ADAMS Accession No. ML18296A466 (non-proprietary)]
7. Letter from M. D. Orenak (NRC) to R. S. Bement (APS), *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Acceptance Review for License Amendment and Exemption requests Regarding the Implementation of Framatome High Thermal Performance Fuel (EPIDs L-2018-LLA-0194 and L-2018-LLE-0010)*, November 13, 2018 [NRC ADAMS Accession No. ML18312A332]
8. Letter from S. P. Lingam (NRC) to R. S. Bement (APS), *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – First Regulatory Audit Plan for January 22-23, 2019, in Support of Framatome High Thermal Performance Fuel License Amendment Request and Exemption (EPID L-2018-LLA-0194 and EPID L-2018-LLE-0010)*, January 15, 2019 [NRC ADAMS Accession No. ML19011A108]
9. Letter 102-07869-MLL/MDD from M. L. Lacal (APS) to Document Control Desk (NRC), *Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529, and 50-530, Renewed Operating License Nos. NPF-41, NPF-51, and NPF-74, Audit Presentation Slides Regarding License Amendment Request and Exemption Request to Support the Implementation of Framatome CE16HTP™ Fuel*, March 1, 2019 [NRC ADAMS Accession No. ML19060A298 (non-proprietary)]
10. Letter from S. P. Lingam (NRC) to R. S. Bement (APS), *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Regulatory Audit Summary for the January 22-23, 2019, Audit for the License Amendment and Exemption Requests Associated with Framatome High Thermal Performance Fuel (EPID L-2018-LLA-0194 and EPID L-2018-LLE-0010)*, August 13, 2019 [NRC ADAMS Accession No. ML19218A293]
11. Email from S. P. Lingam (NRC) to M. S. Cox (APS), *Palo Verde 1, 2, and 3 – Official RAIs from IRAB for Framatome HTP Fuel LAR and Exemption (EPIDs L-2018-LLA-0194 and L-2018-LLE-0010)*, April 5, 2019 [NRC ADAMS Accession No. ML19098A187]
12. Letter 102-07921-MLL/MDD from M. L. Lacal (APS) to Document Control Desk (NRC), *Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529, and 50-530, Renewed Operating License Nos. NPF-41, NPF-51, and NPF-74, Response to NRC Staff Request for Additional Information from Reactor Assessment and Human Performance Branch Regarding License Amendment and Exemption Requests Related to the Implementation of Framatome High Thermal Performance Fuel*, May 17, 2019 [NRC ADAMS Accession No. ML19137A118]

13. Letter from S. P. Lingam (NRC) to R. S. Bement (APS), *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Second Regulatory Audit Plan for June 17-20, 2019, in Support of Framatome High Thermal Performance Fuel License Amendment Request and Exemption (EPID L-2018-LLA-0194 and EPID L-2018-LLE-0010)*, June 4, 2019 [NRC ADAMS Accession No. ML19154A469]
14. Letter from S. P. Lingam (NRC) to R. S. Bement (APS), *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Regulatory Audit Summary for the June 17-20, 2019, Audit for the License Amendment and Exemption Requests Associated with Framatome High Thermal Performance Fuel (EPID L-2018-LLA-0194 and EPID L-2018-LLE-0010)*, September 9, 2019 [NRC ADAMS Accession No. ML19235A256]
15. Letter from S. P. Lingam (NRC) to R. S. Bement (APS), *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Request for Additional Information for Amendment and Exemption Request Associated with Revising Technical Specifications to Support the Implementation of Framatome High Thermal Performance Fuel (EPID L-2018-LLA-0194 and EPID L-2018-LLE-0010)*, August 29, 2019 [NRC ADAMS Accession Nos. ML19234A320 (non-proprietary) and ML19234A321 (proprietary)]
16. Letter 102-07986-MLL/MDD from M. L. Lacal (APS) to Document Control Desk (NRC), *Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529, and 50-530, Renewed Operating License Nos. NPF-41, NPF-51, and NPF-74, Response to NRC Request for Additional Information Regarding License Amendment and Exemption Requests Related to the Implementation of Framatome CE16HTP Fuel*, October 4, 2019 [NRC ADAMS Accession Nos. ML19277J457 (non-proprietary) and ML19277J458 (proprietary)]
17. Letter from S. P. Lingam (NRC) to R. S. Bement (APS), *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Third Regulatory Audit Plan for November 7-8, 2019, in Support of Framatome High Thermal Performance Fuel License Amendment Request and Exemption (EPID L-2018-LLA-0194 and EPID L-2018-LLE-0010)*, October 30, 2019 [NRC ADAMS Accession No. ML19301A905]
18. Letter from S. P. Lingam (NRC) to R. S. Bement (APS), *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)*, January 23, 2018 [NRC ADAMS Accession Nos. ML17319A103 (proprietary) and ML17319A107 (non-proprietary)]

ENCLOSURE
ATTACHMENT 1

License Conditions

APPENDIX D

ADDITIONAL CONDITIONS

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

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

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

Amendment Number	Additional Conditions	Implementation Date
207	APS is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, internal flooding, internal fire, and seismic; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on a screening of other external hazards using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in license amendment 207 dated October 10, 2018. Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach). APS will complete the implementation items listed in the Enclosure of APS letter 102-07546, dated July 19, 2017, to the NRC and in Attachment 1, Table 1-1 of APS letter 102-07690, dated May 9, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the enclosure will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.	The license amendment shall be implemented within 90 days of the date of issuance.

Amendment Number	Additional Conditions	Implementation Date
209	Arizona Public Service Company (APS) is approved to implement the risk-informed completion time (RICT) program specified in license amendment 209 dated May 29, 2019. The risk assessment approach and methods, shall be acceptable to the NRC, be based on the as-built, as-operated, and maintained plant, and reflect the operating experience of the plant as specified in RG 1.200. Methods to assess the risk from extending the completion times must be PRA methods accepted as part of this license amendment, or other methods approved by the NRC. If the licensee wishes to use a newly developed method, and the change is outside the bounds of this license condition, the licensee will seek prior NRC approval, via a license amendment. APS will complete the implementation items listed in the Enclosure of APS letter 102-07587, dated November 3, 2017, to the NRC and in Attachment 1, Table 1-1 of APS letter 102-07691, dated May 18, 2018, as updated by APS letter 102-07801, dated October 5, 2018, prior to implementation of RICTs. All issues identified will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the RICT program.	Prior to implementation of RICT program.

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

ENCLOSURE
ATTACHMENT 2

Technical Specification Page Mark-Ups
Affected Pages: 2.0-1, 4.0-1, 5.6-3, and 5.6-7

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

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

2.1.1.2 ~~In MODES 1 and 2, the peak fuel centerline temperature shall be maintained < 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-382-P-A).~~ Replace with Insert "A"

2.1.2 Reactor Coolant System (RCS) Pressure SL

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

2.2 SL Violations

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

2.2.2 If SL 2.1.2 is violated:

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

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

INSERT "A" to 2.1.1.2 (page 2.0-1)

2.1.1.2 In MODES 1 and 2,

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

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

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

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

Replace with insert "B"

~~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)

INSERT "B" to 4.2.1 (page 4.0-1)

The reactor shall contain 241 fuel assemblies.

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

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

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

1. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.
2. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.2.
3. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.4.
4. Boron Dilution Alarm System for Specification 3.3.12.
5. CEA Alignment for Specification 3.1.5.
6. Regulating CEA Insertion Limits for Specification 3.1.7.
7. Part Strength CEA Insertion Limits for Specification 3.1.8.
8. Linear Heat Rate for Specification 3.2.1.
9. Azimuthal Power Tilt - T_q for Specification 3.2.3.
10. DNBR for Specification 3.2.4.
11. Axial Shape Index for Specification 3.2.5.

Add Insert "C"

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

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

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

(continued)

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

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

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

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

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

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

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

Add Insert "D"

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)

INSERT "C" to 5.6.5.a (page 5.6-3)

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

INSERT "D" to 5.6.5.b (page 5.6-7)

27. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors." [Methodology for Specification 3.2.1, Linear Heat Rate]
28. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." [Methodology for Specification 3.2.1, Linear Heat Rate]
29. BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code." [Methodology for Specification 3.2.1, Linear Heat Rate]
30. BAW-10241(P)(A), "BHTP Correlation Applied with LYNXT." [Methodology for Specification 3.2.4, DNBR]
31. EPRI-NP-2511-CCM-A, "VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores." [Methodology for Specification 3.2.4, DNBR]

ENCLOSURE
ATTACHMENT 3

Clean Technical Specification Pages

Affected Pages: 2.0-1, 4.0-1, 5.6-3, 5.6-7, and 5.6-8

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

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

2.1.1.2 In MODES 1 and 2,

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

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

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

2.1.2 Reactor Coolant System (RCS) Pressure SL

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

2.2 SL Violations

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

2.2.2 If SL 2.1.2 is violated:

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

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

4.0 DESIGN FEATURES

4.1 Site Location

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

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies.

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

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

4.2.2 Control Element Assemblies

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

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

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

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

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

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

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

4. Boron Dilution Alarm System for Specification 3.3.12.

5. CEA Alignment for Specification 3.1.5.

6. Regulating CEA Insertion Limits for Specification 3.1.7.

7. Part Strength CEA Insertion Limits for Specification 3.1.8.

8. Linear Heat Rate for Specification 3.2.1.

9. Azimuthal Power Tilt - T_q for Specification 3.2.3.

10. DNBR for Specification 3.2.4.

11. Axial Shape Index for Specification 3.2.5.

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

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

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

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

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

(continued)

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

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

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

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

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

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

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

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

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

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

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

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

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

(continued)

5.6 Reporting Requirements

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

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

5.6.7 Tendon Surveillance Report

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

5.6.8 Steam Generator Tube Inspection Report

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

a. The scope of inspections performed on each SG.

b. Active degradation mechanisms found.

c. Nondestructive examination techniques utilized for each degradation mechanism.

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

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

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

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

(continued)

ENCLOSURE
ATTACHMENT 4

Technical Specification Bases Page Mark-Ups
(Provided for Information Only)

Affected Pages: B 2.1.1-3, B 2.1.1-4, and B 3.5.1-2

BASES

APPLICABLE SAFETY ANALYSES (continued)

h. Log Power Level — High trip;

i. Reactor Coolant Flow — Low trip; and

j. Steam Generator Safety Valves.

The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation used in the protection system design as a measure of the core power is proportional to core power.

The SL represents a design requirement for establishing the protection system trip setpoints identified previously. LCO 3.2.1, "Linear Heat Rate (LHR)," and LCO 3.2.4, "Departure From Nucleate Boiling Ratio (DNBR)," or the assumed initial conditions of the safety analyses (as indicated in the UFSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

~~SL 2.1.1.1 and SL 2.1.1.2 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature remains below melting.~~

INSERT "A"

~~Prior to the Next Generation Fuel (NGF) implementation:~~

~~The minimum value of the DNBR during normal operation and design basis AOOs is limited to 1.34, based on a statistical combination of CE-1 Critical Heat Flux (CHF) correlation and engineering factor uncertainties, and is established as an SL. Additional factors such as rod bow and spacer grid size and placement will determine the limiting safety system settings required to ensure that the SL is maintained.~~

~~Following NGF implementation:~~

~~The minimum value of the DNBR during normal operation and design basis Anticipated Operational Occurrences AOOs is limited to 1.34 using the ABB-NV correlation for the first NGF transition core. This value is based on a statistical combination of CHF correlation and engineering factor uncertainties, and is established as a SL for the first NGF transition core. For the second NGF transition core and~~

(continued)

BASES

SAFETY LIMITS (continued)

~~subsequent cores with NGF, the minimum value of the DNBR during normal operation and design is AOOs is limited to 1.25 using the WSSV and ABB-NV correlations. This value is based on a statistical combination of CHF correlation and engineering factor uncertainties. Additional factors such as rod bow and placement will determine the limiting safety system settings required to ensure that the SL is maintained.~~

INSERT "B" AS A NEW PARAGRAPH

~~The WSSV and ABB-NV correlations are used in the safety and setpoint analyses. However because of existing hardware limitations, the CPC algorithm will retain the CE-1 correlation and the DNBR-Low trip setpoint and Allowable Value of 1.34. To maintain consistency with the CPC setpoint, the safety limit value will remain at 1.34 after the first NGF transition core. The adjustment to the lower DNBR limit will be made within the safety and setpoint analyses.~~

INSERT "C" TO START THIS PARAGRAPH

Maintaining the dynamically adjusted peak LHR to ≤ 21 kW/ft or peak fuel centerline temperature ~~≤ 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-382-P-A),~~ ensures that fuel centerline melt will not occur during normal operating conditions or design AOOs.

INSERT "D" AS A NEW PARAGRAPH

~~The~~ design melting point of new fuel with no burnable poison is 5080°F. The melting point is adjusted downward from this temperature depending on the amount of burnup and amount and type of burnable poison in the fuel. The 58°F per 10,000 MWD/MTU adjustment for burnup was accepted by the NRC in Topical Report CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel," August 1992. Adjustments for burnable poisons are established based on NRC approved Topical Report CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.

A steady state peak linear heat rate of 21 kW/ft has been established as the Limiting Safety System Setting to prevent fuel centerline melting during normal steady state operation. Following design basis anticipated operational occurrences, the transient linear heat rate may exceed 21 kW/ft provided the fuel centerline melt temperature is not exceeded. However, if the transient linear heat rate does not exceed 21 kW/ft, then the fuel centerline melt temperature is also not exceeded.

(continued)

INSERT "A" to B 2.1.1 (page B 2.1.1-3)

SL 2.1.1.1:

The minimum value of the DNBR during normal operation and design basis AOOs is based on a statistical combination of the applicable CHF correlation and engineering factor uncertainties, and is established as an SL. Additional factors such as rod bow and spacer grid size and placement will determine the limiting safety system settings required to ensure that the SL is maintained.

The minimum value of the DNBR during normal operation and design basis AOOs is dependent on the fuel types present in the reactor core, and which fuel type had been irradiated prior to the current operating cycle. The fuel types include Westinghouse supplied Standard (i.e., CE16STD) fuel, Westinghouse supplied Next Generation Fuel (i.e., CE16NGF) fuel, and Framatome supplied High Thermal Performance (i.e., CE16HTP) fuel.

1. For a core where CE16STD fuel is limiting, the DNBR analytical limit is 1.34 using the CE-1 or ABB-NV CHF correlation.

2. For a core where CE16NGF fuel is limiting, the DNBR analytical limit is 1.25 using the WSSV and ABB-NV CHF correlations.

3. For a core where CE16HTP fuel is limiting, the DNBR analytical limit is 1.27 using the BHTP CHF correlation.

4. For a mixed core where multiple types are limiting, the most conservative DNBR analytical limit will be used in conjunction with the CHF correlation for each limiting fuel type.

As noted in the preceding discussion, the WSSV, ABB-NV and BHTP CHF correlations may be used in safety and setpoint analyses. However, because of existing hardware limitations, the CPC algorithm will retain the CE-1 correlation and the DNBR-Low trip setpoint and Allowable Value of 1.34.

INSERT "B" to B 2.1.1 (page B 2.1.1-4)

SL 2.1.1.2:

INSERT "C" to B 2.1.1 (page B 2.1.1-4)

For Westinghouse supplied fuel, the

INSERT "D" to B 2.1.1 (page B 2.1.1-4)

For Framatome supplied fuel, the design melting point of new fuel is 4901 °F. The melting point is adjusted downward from this temperature depending on the amount of burnup in the fuel. The 13.7 °F per 10,000 MWD/MTU adjustment for burnup was accepted by the NRC for burnups up to 62 GWD/MTU in Topical Report BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," January 2004.

BASES

BACKGROUND (continued)

Additionally, the isolation valves are interlocked with the pressurizer pressure instrumentation channels to ensure that the valves will automatically open as RCS pressure increases above SIT pressure and to prevent inadvertent closure prior to an accident. The valves also receive a Safety Injection Actuation Signal (SIAS) to open. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the SITs will be available for injection without reliance on operator action.

During operations at RCS pressure greater than 430 psia the SIT isolation valves are procedurally locked open and motive power is removed with the breakers locked open, which is conservative with respect to SR 3.5.2.5.

The open and closure interlocks are tested as described in UFSAR 7.6.2.2.2 (Reference 7). The open interlock is functionally tested per Reference 8 (TRM, T3.5 (ECCS); TSR 3.5.200.4). The SIAS function to open these valves is tested per Reference 8 using the method described in Reference 7.

The SIT gas and water volumes, gas pressure, and outlet pipe size are selected to allow three of the four SITs to partially recover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three SITs are adequate for this function is consistent with the LOCA assumption that the entire contents of one SIT will be lost via the break during the blowdown phase of a LOCA.

APPLICABLE SAFETY ANALYSES

The SITs are taken credit for in both the large and small break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the SITs. Reference to the analyses for these DBAs is used to assess changes to the SITs as they relate to the acceptance limits.

INSERT "E"

In performing the LOCA calculations, conservative assumptions are made concerning the availability of SI flow. These assumptions include signal generation time, equipment starting times, and delivery time due to system piping. In the early stages of a LOCA with a loss of offsite power, the SITs provide the sole source of makeup water to the RCS. (The assumption of a loss of offsite power is required by regulations.) This is because the LPSI pumps and HPSI pumps cannot deliver flow until the Diesel Generators (DGs) start, come to rated speed, and go through their timed loading sequence. In cold leg breaks, the entire contents of one SIT are assumed to be lost through the break during the blowdown and reflood phases.

(continued)

INSERT "E" to B 3.5.1 (page B 3.5.1-2)

either partially or completely

ENCLOSURE
ATTACHMENT 5

Affidavit from Arizona Public Service Company
Submitted in Accordance with 10 CFR 2.390
to Consider Enclosure Attachment 10
as a Proprietary Document

AFFIDAVIT

STATE OF ARIZONA)
) ss.
CITY OF PHOENIX)

1. My name is Bruce Rash. I am employed by Arizona Public Service Company ("APS"). My present capacity is Vice President, Nuclear Engineering, for the Palo Verde Nuclear Generating Station ("PVNGS"), and in that capacity I am authorized to execute this Affidavit.

2. APS is the operating agent for PVNGS. I am familiar with the policies established by APS to determine whether certain APS information is proprietary and confidential, and to ensure the proper application of these policies.

3. I am familiar with APS information in the following document: Attachment 10 to the enclosure for APS Correspondence 102-08012, "Supplemental Response to NRC Request for Additional Information Regarding License Amendment and Exemption Requests Related to the Implementation of Framatome CE16HTP Fuel," referred to herein as "Document." Information contained in this Document has been classified by APS as proprietary in accordance with the policies established by APS for the control and protection of proprietary and confidential information.

4. The information contained in this Document is proprietary and confidential in nature and of the type customarily held in confidence by Framatome (formerly Areva, Inc.), Westinghouse, and APS, and not made available to the public. Based on my experience in the nuclear industry, I am aware that other companies also regard the type of information contained in the Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding proprietary information from public

disclosure is made in accordance with 10 CFR 2.390. The information qualifies for withholding from public disclosure under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. APS applied the following criteria to determine that the information contained in the Document should be classified as proprietary and confidential:

(a) APS has a non-disclosure agreement with Westinghouse Electric Company LLC ("Westinghouse"), Framatome, Inc. ("Framatome"), and Structural Integrity Associates, Inc. (SI) (the NDA is referred to as the "Westinghouse-Framatome-SI-APS NDA"), under which Westinghouse and Framatome have provided to APS certain proprietary and confidential information contained in the Document.

(b) The information reveals details of Westinghouse's, APS's, and/or Framatome's research and development plans and programs, or the results of these plans and programs.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive commercial advantage for Westinghouse, APS, and/or Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive commercial advantage for Westinghouse, APS, and/or Framatome on product optimization or marketability.

(e) The unauthorized use of the information by one of Westinghouse's, APS's, and/or Framatome's competitors would permit the offending party to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(f) The information contained in the Document is vital to a competitive commercial advantage held by Westinghouse, APS, and/or Framatome, would be helpful to

their competitors, and would likely cause substantial harm to the competitive position of Westinghouse, APS, and/or Framatome.

(g) It reveals aspects of past, present, or future Westinghouse, Framatome, or APS funded development plans and programs of potential commercial value.

7. In accordance with APS's policies governing the protection and control of proprietary and confidential information, the information contained in this Document has been made available, on a limited basis, to others outside APS only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. APS's policies require that proprietary and confidential information be kept in a secured file or area and distributed on a need-to-know basis. The information contained in the Document has been kept in accordance with these policies.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief, and if called as a witness I would competently testify thereto. I declare under penalty of perjury under the laws of the State of Arizona that the above is true and correct.

Bruce Rash

SUBSCRIBED before me this 25TH

day of November, 2019.

DONNA NORMAN
Notary Public - State of Arizona
MARICOPA COUNTY
My Commission Expires
August 09, 2021

Donna Norman

NOTARY PUBLIC, STATE OF ARIZONA
MY COMMISSION EXPIRES: August 9, 2021
Reg. #:

ENCLOSURE
ATTACHMENT 6

Affidavit from Westinghouse
Submitted in Accordance with 10 CFR 2.390
to Consider Enclosure Attachment 10
as a Proprietary Document

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

(1) I, Korey L. Hosack, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of 102-08012-MLL/MDD, "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3; Docket Nos. STN 50-528, 50-529, and 50-530; Renewed Operating License Nos. NPF-41, NPF-51, and NPF-74; Supplemental Response to NRC Request for Additional Information Regarding License Amendment and Exemption Requests Related to the Implementation of Framatome CE16HTP Fuel" be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

  (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

  (ii) Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

AFFIDAVIT

(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

AFFIDAVIT

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 2019/11/14

Korey L. Hosack, Manager
Licensing, Analysis, & Testing

ENCLOSURE
ATTACHMENT 7

Affidavit from Framatome
Submitted in Accordance with 10 CFR 2.390
to Consider Enclosure Attachments 10 and 11
as Proprietary Documents

AFFIDAVIT

1. My name is Phil Opsal. I am Product Licensing Manager for Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the attachment to Arizona Public Service Company (APS) letter number 102-08012-MLL/MDD entitled, "Supplemental Response to NRC Request for Additional Information Regarding License Amendment and Exemption Requests Related to the Implementation of Framatome CE16HTP Fuel," and in Licensing Report ANP- 3640Q2P, Revision 0, entitled, "Palo Verde Units 1, 2 and 3 Small Break LOCA Summary Report NRC RAI Responses," referred to herein as "Documents." Information contained in these Documents has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. These Documents contain information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in these Documents as proprietary and confidential.

5. These Documents have been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in these Documents be withheld from public disclosure. The request for withholding of proprietary information is

made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in these Documents is considered proprietary for the reasons set forth in paragraphs 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in these Documents has been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Phil Opsal

Phil Opsal

Commonwealth of Virginia)
) ss.
City of Lynchburg)

SUBSCRIBED before me this 25th day of November, 2019.

Heidi Hamilton Elder

Heidi Elder
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/22
Reg. # 7777873

HEIDI HAMILTON ELDER
COMMONWEALTH OF VIRGINIA
REGISTRATION NO. 7777873
MY COMM. EXPIRES 12/31/2022
NOTARY PUBLIC

ENCLOSURE
ATTACHMENT 8

APS Supplemental Responses to NRC Request for Additional Information

[NON-PROPRIETARY VERSION]

SNPB RAI-10

The regulatory basis for this RAI is Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50 and 10 CFR 50.46.

ANP-10337P-A, "PWR [Pressurized-Water Reactor] is an NRC-approved methodology for a faulted condition (earthquakes and postulated pipe breaks in the reactor coolant system) analysis for cores containing Framatome fuel designs. However, the Palo Verde core will have three different fuel designs, including CE STD and CE NGF designs. Explain how the faulted condition analysis will be performed for Palo Verde mixed core conditions.

Supplemental Response to SNPB RAI-10

The APS response to this RAI, dated October 4, 2019 (Reference 1), stated that faulted condition analyses for PVNGS transition (mixed core) configurations are performed by both Framatome and Westinghouse, using their respective approved methodologies. Each fuel vendor is responsible for evaluating the effects of seismic and Loss of Coolant Accident (LOCA) loads for its own fuel types. Westinghouse and Framatome have exchanged proprietary technical information to facilitate the PVNGS evaluations by each vendor.

The Reference 1 response to this RAI addressed the Framatome faulted condition analysis methodology (Reference 2) and its application for PVNGS. The following supplements the Reference 1 response with additional information for the Westinghouse faulted condition analysis methodology and its application for PVNGS.

The Westinghouse faulted condition analysis methodology is described in CENPD-178-P, Revision 1-P, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading," dated August 1981 (Reference 3). The NRC initially approved the application of this methodology for PVNGS in October 1984 (Reference 4), and approved its application again in January 2018 for a PVNGS transition from Westinghouse CE16STD to Westinghouse CE16NGF (Reference 5).

The supplemental response to SNPB RAI-29 herein describes PVNGS fuel transitions that may involve Framatome CE16HTP fuel, and associated mixed core row configurations that were analyzed by Westinghouse using the Reference 3 methodology. Consideration of End-of-Life (EOL) grid crush is addressed in the supplemental response to SNPB RAI-30 herein.

SNPB RAI-17

Realistic, with allowance for uncertainty, or conservative modeling of the fuel in the reactor core is necessary to ensure that appropriately conservative figures of merit are predicted for comparison against the acceptance criteria in 10 CFR 50.46(b). Please justify how both the Framatome and Westinghouse LOCA analyses for Palo Verde would address the potential suite of mixed core configurations that could ensue following implementation of the proposed license amendment.

a. Please describe whether and how each vendor's existing LOCA analysis for both small and large breaks consider a bounding core configuration that would address the impacts of potential variations in core composition on the predicted LOCA figures of merit.

b. Please clarify and justify the conditions under which each analysis would be deemed applicable to a given mixed core configuration, and the conditions under which an explicit analysis of a particular mixed core configuration would become necessary.

Supplemental Response to SNPB RAI-17

The APS response to this RAI dated October 4, 2019 (Reference 1), addressed PVNGS transitions from one fuel type to another fuel type and the potential effects of co-resident fuel assemblies on Emergency Core Cooling System (ECCS) performance analyses. As stated in the Reference 1 response, large-break and small-break LOCA analyses are performed by both Framatome and Westinghouse, using their respective methodologies for their respective fuel types, to ensure that calculated figures of merit remain bounding for anticipated mixed core configurations during fuel transitions. Transition core configurations take into consideration PVNGS fuel management practices that utilize full reload batches of fresh fuel from a single fuel supplier.

The following supplements the Reference 1 response with additional information regarding Westinghouse ECCS performance analyses for PVNGS.

Parts a and b Response

APS asked Westinghouse to analyze both large-break and small-break LOCAs with respect to the following scenarios to assess transition (mixed core) effects:

- A transition from a full core of CE16STD to a full core of CE16HTP (except for the hot assembly).
- A transition from a full core of CE16NGF to a full core of CE16HTP (except for the hot assembly).
- A transition from a mixed core of CE16NGF and CE16STD to a full core of CE16HTP (except for the hot assembly).
- A transition from a full core of CE16HTP (except for the hot assembly) to a full core of CE16NGF.

For large-break LOCA transition scenarios, Westinghouse performed quantitative analyses guided by qualitative analyses regarding core configurations to be analyzed. The quantitative analyses utilized the User-Controlled Interface (UCI) parameter "mixed_core" to perform evaluations of PVNGS mixed core configurations with the 1999 Evaluation Model (EM). This parameter, which was described in the PVNGS 10 CFR 50.46 annual report to NRC for calendar year 2008 (Reference 6), is not intended for licensing applications of the 1999 EM, which must use a uniform core representation of one fuel assembly type for conformance to the 1999 EM licensed methodology. This parameter does, however, facilitate the quantitative analysis of mixed core effects that may arise as a result of co-resident, thermal-hydraulically dissimilar fuel types. These mixed core analyses and evaluations, combined with results referenced from applicable licensed Analyses of Record (AORs), were used to develop the (qualitative) technical justification of expected mixed core configurations.

For small-break LOCA transition scenarios, Westinghouse performed a qualitative analysis of mixed core effects for the **[[** **]]^{Wa,c}** S2M EM, as described in the Reference 1 response to this RAI.

The Westinghouse LBLOCA mixed core studies revealed the following:

- The mixed core configurations that were analyzed showed that calculated figures of merit **[[** **]]^{Wa,c}**.
- The mixed core configurations that were analyzed resulted in figures of merit that were **[[** **]]^{Wa,c}** licensing AOR cases.

Westinghouse concluded the following with respect to both large-break and small-break LOCAs:

- For a transition from a full core of CE16STD to a full core of CE16HTP (except for the hot assembly), the existing PVNGS licensing basis AORs for a full core of CE16STD would bound the Westinghouse fuel response throughout the transition, with no additional Peak Cladding Temperature (PCT) or oxidation penalty associated with the transition.
- For a transition from a full core of CE16NGF to a full core of CE16HTP (except for the hot assembly), and for the transition from a full core of CE16HTP (except for the hot assembly) to a full core of CE16NGF, the existing PVNGS licensing basis AOR for the transition from CE16STD to CE16NGF (including a full core of CE16NGF) would bound the Westinghouse fuel response throughout the transition, with no additional PCT or oxidation penalty associated with the transition.
- For a transition from a mixed core of CE16STD and CE16NGF to a full core of CE16HTP (except for the hot assembly), the existing PVNGS licensing basis AOR for the transition from CE16STD to CE16NGF (including a full core of CE16NGF) would bound the Westinghouse fuel response throughout the transition, with no additional PCT or oxidation penalty associated with the transition.

SNPB RAI-19

As discussed in the LAR, the SBLOCA analysis methods proposed for Palo Verde include a deviation from the approved EMF-2328(P)(A) methodology. **[[**

]][F] To ensure that the methodology continues to appropriately predict a SBLOCA transient for demonstrating compliance with 10 CFR 50.46(b) acceptance criteria, please address the following:

a. **[[** **]]**[F]

b. **[[** **]]**[F]

c. **[[** **]]**[F]

d. **[[** **]]**[F]

Supplemental Response to SNPB RAI-19

The supplemental response is provided in Enclosure Attachments 9 (non-proprietary) and 11 (proprietary).

SNPB RAI-22

The description of fuel assemblies contained in proposed TS 4.2.1 would replace the names of three specific types of cladding that have been approved for use at Palo Verde with the term "zirconium-alloy clad." The term "zirconium-alloy clad" in proposed TS 4.2.1 has not been defined, and its intent is not clear with respect to either conventional or coated cladding types. As such, it is not clear that replacement of specific cladding alloys with an undefined generic term would create an enforceable TS requirement capable of satisfying regulatory requirements in 10 CFR 50.36, "Technical specifications."

a. Please clarify the intended definition of the term "zirconium-alloy clad" and discuss how the wording of the proposed TS and its basis would assure an unambiguous interpretation that satisfies applicable regulatory requirements. In particular, the requested information is necessary to confirm satisfaction of 10 CFR 50.36(c)(4), which states, in part that "Design features to be included [in the "Design Features" section of the technical specifications] are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety" and are not covered under 10 CFR 50.36(c)(1)-(3).

b. Marked up TS page 4.0-1 in Attachment 2 of the LAR dated July 6, 2018, does not highlight addition of the term "zirconium-alloy clad" as a proposed change to TS 4.2.1. Please clarify whether this is an omission, and, as necessary, provide a corrected markup of page 4.0-1.

Supplemental Response to SNPB RAI-22

Part a Response

APS has updated the proposed Technical Specification 4.2.1 wording as follows to use the term "NRC approved cladding material" (see Enclosure Attachments 2 and 3):

The reactor shall contain 241 fuel assemblies.

a. Each assembly shall consist of a matrix of fuel rods with an NRC approved cladding material with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. Each unit-specific COLR shall contain an identification of the fuel types and cladding material in the reactor, and the associated COLR methodologies.

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

A corresponding change to Technical Specification 5.6.5 has been proposed to require identification of the cladding material in use in each reactor in the unit specific Core Operating Limits Reports (see Enclosure Attachments 2 and 3).

Part b Response

The editorial error that previously appeared on marked-up TS page 4.0-1 has been corrected by subsequent mark-ups (see Enclosure Attachments 2 and 3).

SNPB RAI-25

Historically, the analytical methods of domestic fuel vendors have generally been developed for application to reactor cores where all fresh fuel assemblies have been manufactured by that same vendor. In light of U.S. nuclear plants' historical reliance upon a single fuel supplier for one or typically a number of fuel cycles, however, such a restriction may not be specified explicitly in each topical report (or its corresponding safety evaluation) in proposed Palo Verde TS 5.6.5. To assure that the administrative controls in proposed Palo Verde TS 5.6.5 are sufficient to assure operation of the facility in a safe manner, in accordance with 10 CFR 50.36(c)(5),

a. Please clarify whether implementation of the proposed LAR would permit operation with mixed batches of fresh fuel in the reactor core of any unit at Palo Verde.

b. If implementation of the proposed license amendment would permit operation with mixed batches of fresh fuel under existing regulatory requirements applicable to Palo Verde, then please either

    i. provide justification for the acceptability of using mixed batches of fresh fuel, considering the applicability of the full suite of reload analysis and COLR methodologies, including the potential for increased uncertainties associated with mixed batches of fresh fuel and any validation of the analytical methods for such conditions, or

    ii. propose a binding restriction that would forbid operation with mixed batches of fresh fuel.

c. If, following implementation of the proposed license amendments, operation with mixed batches of fresh fuel would not be permitted under existing regulatory requirements applicable to Palo Verde, then please identify the specific requirement(s) that would preclude operation with mixed batches of fresh fuel.

Supplemental Response to SNPB RAI-25

Part a Response

As discussed in the Reference 1 response to SNPB RAI-1, the normal reload process will continue to be based on using a full reload batch from a single fuel supplier, and the normal fuel transition process will continue to be based on transitioning from a full core of one type of fuel to a full core of another type of fuel.

APS is no longer requesting authorization in the proposed license amendment to permit operation with mixed batches of fresh fuel in the reactor core of any unit at Palo Verde. Lead Test Assemblies per Technical Specification 4.2.1.b are not considered mixed fresh fuel.

Part b Response

APS is no longer requesting authorization for mixed fresh fuel core designs with this license amendment request. The implications of mixed fresh fuel in the areas of core design, thermal-hydraulics, transient analysis, and Core Operating Limits Supervisory System/Core Protection Calculator (COLSS/CPC) setpoints were addressed during the June 2019 regulatory audit. The following license condition is proposed (see Enclosure Attachment 1):

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

Part c Response

Because APS is proposing a license condition, no response is required.

SNPB RAI-26

The proposed LAR would implement changes that may obviate future NRC review of certain types of fuel transitions that have historically been subject to review. As such, please justify whether a revision is necessary to the license condition imposed in Amendment No. 205 (which applies a restriction specific to Westinghouse NGF) to ensure its compatibility with the proposed license amendments. In particular, to assure compliance with the acceptance criteria in 10 CFR 50.46(b), as well other regulatory requirements for fuel integrity deriving from GDC 10, please clarify whether the terms of the license condition should apply, not only to NGF, but to any future Westinghouse-supplied fuel designs introduced at Palo Verde to which FATES3B would be applied.

Supplemental Response to SNPB RAI-26

Amendment 205 to the Palo Verde Nuclear Generating Station Renewed Operating License added a License Condition to Appendix D of Facility Operating License Nos. NPF-41, NPF-51, and NPF-74 as part of the NRC approval for use of Westinghouse Next Generation Fuel. To address the issue identified in this RAI, the following change (in bold font) to the Amendment 205 license condition is proposed:

> APS shall apply a radial power fall off (RFO) curve penalty, equivalent to the fuel centerline temperature reduction in Section 4 of Attachment 8 to the Palo Verde license amendment request dated July 1, 2016, to accommodate the anticipated impacts of thermal conductivity degradation (TCD) on the predictions of FATES3B at high burnup for Westinghouse Next Generation Fuel **or to future Westinghouse-supplied fuel designs introduced at PVNGS to which the FATES3B fuel performance code would be applied**.
>
> To ensure the adequacy of this RFO curve penalty, as part of its normal reload process for each cycle that analysis using FATES3B is credited, APS shall verify that the FATES3B analysis is conservative with respect to an applicable confirmatory analysis using an acceptable fuel performance methodology that explicitly accounts for the effects of thermal conductivity degradation (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.

SNPB RAI-28

In order to confirm that introduction of Framatome HTP fuel will support continued compliance with the acceptance criteria in 10 CFR 50.46(b) for other types of co-resident fuel, please

a. describe the Westinghouse large- and small-break LOCA analyses performed for mixed core conditions involving Framatome HTP fuel,

b. provide the results of these Westinghouse large- and small-break LOCA analyses and confirm that the acceptance criteria of 10 CFR 50.46(b) remain satisfied.

Supplemental Response to SNPB RAI-28

The APS response to this RAI dated October 4, 2019 (Reference 1), stated that Westinghouse analysis work related to a planned PVNGS CE16STD to CE16HTP fuel transition was then underway, and that results would be made available for NRC staff review at a later date. Westinghouse analysis work is now complete and consequently the following paragraphs supplement the Reference 1 response with additional information about the transition core Westinghouse LOCA evaluations for PVNGS.

Part a Response

Please see the supplemental response to SNPB RAI-17 herein for a description of the Westinghouse large-break and small-break LOCA analyses that were performed for mixed cores with co-resident Framatome CE16HTP fuel. Also please note that the transitions described in the supplemental response to SNPB RAI-17 herein include more scenarios than just the planned PVNGS CE16STD to CE16HTP transition discussed in the Reference 1 APS response to this RAI.

Part b Response

Please see the supplemental response to SNPB RAI-17 herein. Westinghouse has concluded that (a) existing PVNGS licensing basis large-break and small-break LOCA AORs for a full core of CE16STD would bound Westinghouse fuel response throughout the transition scenario which begins with a full core of CE16STD, with no additional PCT or oxidation penalty to account for co-resident CE16HTP; and (b) existing PVNGS licensing basis large-break and small-break LOCA AORs for a full core of CE16NGF would bound Westinghouse fuel response throughout all of the evaluated transition scenarios, with no additional PCT or oxidation penalty to account for co-resident CE16HTP. Thus the figures of merit previously calculated by Westinghouse for the licensing basis LOCA AORs would remain applicable and in compliance with 10 CFR 50.46(b). Additionally, Westinghouse concluded that co-resident CE16HTP fuel during these transition scenarios would not invalidate the existing Westinghouse post-LOCA Long-Term Cooling (LTC) licensing basis AORs for PVNGS, relative to decay heat removal and boric acid precipitation, thus assuring continued compliance with 10 CFR 50.46(b).

SNPB RAI-29

Section 2.3.2 of Attachment 10 of the LAR dated July 6, 2018, describes the fuel assembly structural analyses, including externally applied forces such as earthquakes and postulated pipe breaks. The analyses supporting the Advanced CE16 HTP fuel assembly and the use of the approved Framatome methodology in ANP-10337P-A is described. However, there is little discussion on the impact of a mixed core containing CE16STD, CE16NGF, and CE16 HTP fuel bundles on the predicted response of each fuel assembly design (i.e., margin to respective design criteria).

a. Discuss the sensitivity studies conducted to identify the limiting mixed core configurations with respect to each fuel assembly design, predicted results (i.e., margin to respective design criteria), and how future core loading patterns will demonstrate that these calculations remain bounding.

b. Each fuel vendor has separately analyzed mixed core configurations and the performance of their respective fuel assemblies. Provide a comparison of Westinghouse and Framatome dynamic model predictions (e.g. horizontal accelerations, impact loads) and identify and disposition inconsistencies.

c. Discuss the methods used to assess differences in spacer grid (and mid-grid mixing grids) axial location and height.

Supplemental Response to SNPB RAI-29

The APS response to this RAI dated October 4, 2019 (Reference 1), addressed Framatome faulted condition analyses for PVNGS, including mixed core configurations. The following paragraphs supplement the Reference 1 response with additional information about the Westinghouse faulted condition analyses for PVNGS, including mixed core configurations.

Part a Response

The Westinghouse faulted condition analyses for PVNGS included a number of mixed core configurations that addressed the following transition scenarios:

- A transition from a full core of Westinghouse CE16STD to a full core of Framatome CE16HTP. Representative **[[**

  ]]^{Wa,c} for this transition.

- A transition from a full core of Westinghouse CE16NGF to a full core of Framatome CE16HTP, with an optional Westinghouse CE16STD assembly in the center of the core. Representative **[[**

  ]]^{Wa,c} for this transition.

- A transition from a full core of Framatome CE16HTP to a full core of Westinghouse CE16NGF, with an optional Westinghouse CE16STD assembly in the center of the

core. Representative **[[**

]]^{Wa,c} for this transition.

Mixed core configurations for these transitions were defined based on PVNGS fuel management practices, core design groundrules, and recent core design patterns. All three fuel types - CE16STD, CE16NGF, and CE16HTP - were analyzed at Beginning-of-Life (BOL) with still-water damping, consistent with the current PVNGS licensing basis methodology. Sensitivity analyses for the mixed core configurations included analysis for the required cases of frequency-shifted time histories **[[**
]]^{Wa,c}.

The Westinghouse analysis results will remain bounding for PVNGS transition core designs because fuel management practices and core design groundrules effectively establish **[[**

]]^{A}. Regardless, Criterion III of 10 CFR Part 50 Appendix A requires the incorporation of these analyses into the reload core design process to ensure that design inputs (analyzed configurations) are effectively translated into design outputs (core load maps).

Westinghouse analytical results for the three transition scenarios described above were as follows:

- For the Operating Basis Earthquake (OBE), positive margins to grid strength design criteria were maintained for both **[[** **]]**^{Wa,c} loads for all three transition scenarios, for CE16STD mid-grids and top grids, and for CE16NGF mid-grids, top grids, and Intermediate Flow Mixing (IFM) grids. All fuel assembly components satisfy the acceptance criteria.
- For the combination of a Safe Shutdown Earthquake (SSE) and a LOCA, positive margins to grid strength design criteria were maintained for both **[[** **]]**^{Wa,c} loads for the CE16NGF to CE16HTP and the CE16HTP to CE16NGF transition scenarios, for CE16STD mid-grids and top grids, and for CE16NGF mid-grids, top grids, and Intermediate Flow Mixing grids. For these two transitions, it is noted that the CE16STD results are for an optional single assembly placed in the center of the core. All fuel assembly components satisfy the acceptance criteria. The guide tube stresses for the combined SSE and LOCA meet the more restrictive OBE acceptance criteria.
- For the combination of a SSE and a LOCA, positive margins to grid strength design criteria were maintained for both **[[** **]]**^{Wa,c} loads for the CE16STD to CE16HTP transition scenario, for inboard (away from the core

shroud) CE16STD mid-grids and for both inboard and peripheral (adjacent to the core shroud) CE16STD top grids. For this transition, however, peripheral CE16STD mid-grids were potentially susceptible to grid deformation, as evidenced by a negative margin of **[[**

]]$^{\mathrm{Wa,c}}$. This susceptibility to grid deformation is similar to, and actually below, what is calculated for a full core of CE16STD fuel, where a negative margin of **[[**

]]$^{\mathrm{Wa,c}}$ exists. All other fuel assembly components satisfy the acceptance criteria. The guide tube stresses for the combined SSE and LOCA meet the more restrictive OBE acceptance criteria.

This last conclusion above is mitigated by the fact that the Westinghouse analytical results for the CE16STD to CE16HTP transition are similar to, but less severe than, the analytical results for the PVNGS CE16STD to CE16NGF transition that the NRC staff approved in January 2018 (Reference 5). For the CE16STD to CE16NGF transition, peripheral CE16STD mid-grids had a negative margin of **[[**

]]$^{\mathrm{Wa,c}}$. The NRC stated the following in Section 3.5.2.2 of the PVNGS Safety Evaluation (Reference 5) with regard to these results:

"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 **[[** **]]**$^{\mathrm{Wa,c}}$. 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."

For those peripheral STD assemblies where the predicted seismic/LOCA impact loads during a combined seismic and LOCA event were larger than the spacer grid strengths,

coolability and insertability analyses were performed as alternate justification, as specified in CENPD-178-P, Revision 1-P.

The same conclusions that applied to the PVNGS CE16STD to CE16NGF transition likewise apply to the planned CE16STD to CE16HTP transition, that is:

- Control Element Assembly (CEA) Insertability – Peripheral CE16STD fuel assemblies that are potentially susceptible to grid deformation must be evaluated for CEA insertability because PVNGS reactor cores have CEAs positioned on the core periphery. For the CE16STD to CE16NGF transition, Westinghouse concluded that a **[[ ]]**^{Wa,c} would not alter the assemblies' guide tube patterns, and therefore CEA insertability would not be adversely affected. For the planned CE16STD to CE16HTP transition, the **[[ ]]**^{Wa,c} and thus CEA insertability would also not be adversely affected.
- Deformed Grid Coolability – Peripheral CE16STD fuel assemblies that are potentially susceptible to grid deformation must be evaluated for coolability because grid deformation can result in blockage of the assemblies' channel flow areas. For the CE16STD to CE16NGF transition, Westinghouse concluded that grid deformation could affect the flow areas of deformed channels by values ranging from zero to a maximum of **[[ ]]**^{Wa,c}. However, Westinghouse also concluded that, for this maximum value, there was no PCT penalty effect on Emergency Core Cooling System (ECCS) performance analysis results. This coolability evaluation remains applicable to the planned PVNGS CE16STD to CE16HTP transition, because the evaluation is affected by the type of fuel assembly being evaluated and the amount of grid deformation that is present in that assembly, not by the core composition.

In addition to the conclusions reached in the Westinghouse analysis, APS will operate the PVNGS Units such that fuel bundles along the periphery core are restricted to assembly powers below the core average assembly power.

<u>Part b Response</u>

Although Westinghouse and Framatome have analyzed a variety of mixed core configurations for PVNGS, including similar fuel assembly row configurations, a comparison of their dynamic model predictions is not necessarily straightforward. This is primarily a result of the following significant differences in analytical methodologies:

- The Westinghouse CENPD-178-P, Revision 1-P methodology (Reference 3), described in the supplemental response to SNPB RAI-10 herein, differs from the Framatome methodology (Reference 2) in that Westinghouse utilizes **[[**

]]Wa,c. Framatome, on the other hand, models the lower core plate as a **[[** **]]**F. This difference in models is evidenced, for example, in **[[** **]]**F.

- Whereas the Westinghouse detailed core models for lateral (horizontal) analyses of core row configurations treat all fuel assembly types at BOL with still-water damping, Framatome analyses performed for PVNGS **[[** **]]**F. Framatome **[[** **]]**F, and consequently some variation would naturally be expected to occur with respect to predicted accelerations, deflections, and grid impact loads. Thus a simple, side-by-side comparison of Westinghouse and Framatome dynamic model predictions would not be **[[** **]]**A.
- As noted in the Reference 1 APS response to SNPB RAI-29, the Framatome analytical methodology (Reference 2) **[[** **]]**F. Westinghouse analyses **[[** **]]**Wa,c.

Despite the methodology differences described above, APS observed several similarities between Westinghouse and Framatome analytical results, as follows:

- LOCA-related **[[** **]]**Wa,c. That is, LOCA-induced **[[** **]]**Wa,c.
- Mid-grid impact loads were generally **[[** **]]**A despite differences in fuel assembly types. For example, the Reference 1 response to SNPB RAI-29 stated that the maximum predicted CE16HTP grid impact load for all mixed core configurations was **[[** **]]**F. This maximum impact load corresponds to a **[[** **]]**F CE16HTP fuel assembly in a **[[** **]]**F CE16HTP fuel assemblies. For comparative purposes, the maximum **[[** **]]**Wa,c impact load predicted by Westinghouse for a peripheral BOL CE16STD fuel assembly during a CE16STD to

CE16HTP transition, as noted in the supplemental response above to Part a of this RAI, was **[[ ]]Wa,c**. Thus the vendors' analytical mixed core methodologies are judged to **[[ ]]A**, given the known variations in analytical inputs (for example, natural frequencies and damping values) for the different fuel types.

Part c Response

As noted in the supplemental response to SNPB RAI-10 above, Westinghouse and Framatome shared proprietary information to facilitate analytical evaluations by each vendor for PVNGS. This included information pertaining to grid elevations and heights, so that each vendor could assess **[[ ]]A** on adjacent fuel assemblies for mechanical compatibility and faulted condition analyses. The Reference 1 APS response to TH RAI-01 provides an illustrative comparison of the three reactor fuel designs (CE16STD, CE16NGF, and CE16HTP), including relative grid locations.

The Reference 1 APS response to SNPB RAI-29 describes how Framatome evaluated **[[ ]]F**, and notes that **[[ ]]F**. Westinghouse likewise evaluated grid overlap and concluded that the impact stiffness for dissimilar but adjacent spacer grids could be modeled **[[ ]]Wa,c**.

Treatment of grid locations in the Westinghouse methodology also included the following:

- As noted in the supplemental response to Part b of SNPB RAI-29 above, Westinghouse explicitly **[[ ]]Wa,c**. Framatome **[[ ]]F**.
- As noted in the supplemental response to Part a of SNPB RAI-29 above, Westinghouse CE16NGF fuel assemblies include IFM grids, **[[ ]]Wa,c** IFM grids. The CE16NGF IFM grids **[[ ]]Wa,c** fuel assemblies **[[ ]]Wa,c**. Therefore, the IFM grids will not exceed the CE16NGF assembly envelope. Grid deformation only occurs on the **[[**

]]^{Wa,c}.

SNPB RAI-30

Section 2.4, "End-of-Life Grid Crush Strength for CE 16HTP Fuel," of Attachment 10 of the LAR dated July 6, 2018, describes the Framatome methodology in ANP-10337P-A and how irradiation effects, identified in Information Notice (IN) 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength," have been addressed for CE16 HTP fuel assemblies.

a. Describe how irradiation effects (e.g., grid crush strength, grid stiffness, bundle stiffness) are addressed in the mixed core configuration.

b. Considering this new, and potentially more limiting mixed core utilization of the CE16STD and CE16NGF fuel assemblies, describe the level of confidence in the predicted seismic/LOCA performance using the Westinghouse methodology, which does not address the irradiation effects identified in IN 2012-09.

    i. Provide predicted margin relative to each CE16STD and CE16NGF design criteria.

Supplemental Response to SNPB RAI-30

The APS response to this RAI dated October 4, 2019 (Reference 1), addressed EOL grid crush for Framatome CE16HTP fuel. The following paragraphs supplement the Reference 1 response with additional information about Westinghouse CE16STD and CE16NGF.

Part a Response

The Reference 1 APS response to SNPB RAI-30 described how irradiation effects and EOL grid crush were addressed for Framatome CE16HTP, including for mixed core configurations. Westinghouse CE16STD and CE16NGF are addressed in the supplemental response to Part b of this RAI below.

Part b Response

As noted in Section 3.6 of the January 2018 NRC Safety Evaluation for the PVNGS transition from CE16STD to CE16NGF (Reference 5), the Pressurized Water Reactor Owners Group (PWROG) has been working on a resolution to the issues presented in IN 2012-09 for Westinghouse and Combustion Engineering (CE) fuel designs, including CE16STD and CE16NGF, but no resolution had been approved by the NRC staff as of the issuance of the PVNGS Safety Evaluation. A draft Safety Evaluation for PWROG Topical Report PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs," was subsequently released by NRC staff in August 2018 (Reference 7).

For the proposed PVNGS transition from CE16STD to CE16HTP, which APS plans to implement in the Spring of 2020, the proposed methodology of PWROG-16043-P could not be adopted as a new PVNGS licensing basis because final approval by NRC staff remained pending during execution of the APS project. APS therefore contracted with Westinghouse to perform mixed core analyses with CE16STD, CE16NGF, and CE16HTP fuel types, using the existing PVNGS licensing basis methodology of CENPD-178-P, Revision 1-P (Reference

3). This is the same methodology as was used to support the Reference 5 PVNGS Safety Evaluation for the transition from CE16STD to CE16NGF.

Analytical results of the mixed core analyses with CE16HTP fuel, using the existing PVNGS licensing basis methodology, are summarized in the supplemental response to SNPB RAI-29. As noted therein, the Westinghouse fuel types (CE16STD and CE16NGF) maintained positive margins to grid strength design criteria for a postulated OBE. However, for a postulated combination of an SSE and LOCA, peripheral CE16STD fuel assemblies in a mixed core with CE16HTP fuel were susceptible to mid-grid deformation, although the impact loads on the peripheral CE16STD grids were less than those previously calculated for the NRC-approved CE16STD to CE16NGF transition. APS therefore concludes that peripheral CE16STD mid-grid deformation may begin to occur for seismic excitation in excess of an OBE, for the planned CE16STD to CE16HTP transition.

Consequently, APS concludes that NRC staff approval of the PVNGS transition to CE16HTP fuel should proceed in a manner similar to that previously documented in the Reference 5 Safety Evaluation for the PVNGS transition to CE16NGF. That is, 10 CFR Part 100, *Reactor Site Criteria*, Appendix A, *Seismic and Geologic Siting Criteria for Nuclear Power Plants*, paragraph V.(a)(2), *Determination of Operating Basis Earthquake*, includes the following regulatory requirement:

> "If vibratory ground motion exceeding that of the Operating Basis Earthquake occurs, shutdown of the nuclear power plant will be required. Prior to resuming operations, the licensee will be required to demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public."

Should PVNGS experience vibratory ground motion in excess of the OBE, the demonstration that no functional damage has occurred would include consideration of possible grid deformation.

SRXB RAI-3

In Section 6.2 of Attachment 10 of the LAR (Reference 1), the licensee explains that the maximum cladding strain value cited in the NGF application for CE 14x14 is the limiting and bounding case for HTP fuel. The LAR also claims that **[[**
]]^{A} therefore **[[**

]]^{A}. The LAR includes Figures 6-1 and 6-2, which provide **[[**
]]^{A} and **[[** **]]**^{A}
respectively. Because the criteria for DNB propagation is presented in terms of cladding strain, it is necessary to examine a comparison between Zr-4 and M5 strain values. Please provide data that shows strain is bounded for Framatome CE16HTP fuel.

Supplemental Response to SRXB RAI-3

This response modifies the Reference 1 requested time-in-DNB criterion from less than 5 seconds to less than 4.5 seconds.

Summary

- The EDGAR test simulation presented in the APS letter of October 4, 2019 (Reference 1), based on an ASTM publication by Forgeron et al (Reference 8), shows that the strain rate (slope of the strain curve) of M5® cladding increases as the cladding deforms. This is a result of a Norton-style creep equation in which hoop stress is an important parameter – as the cladding deforms outward, the cladding wall thickness is reduced and the hoop stress rises, which in turn causes the strain rate to increase as a function of time.
- Based on a limited number of EDGAR tests, it is estimated that the total strain over a 4.5-second period for PVNGS would be less than **[[**

  ]]^{A}. Clad burst is not anticipated to occur during this 4.5-second period. The PVNGS design, including the Core Protection Calculator (CPC) trips, limits the amount of time that fuel rods can be in Departure from Nucleate Boiling (DNB).
- The existing 4.5-second time-in-DNB criterion for Westinghouse fuel is used as a "go / no go" test to determine whether a more detailed cladding deformation and DNB propagation analysis needs to be performed. The licensing limit for cladding deformation is 29.3% strain as stated in CEN-372-P-A (Reference 9), a topical report referenced in PVNGS Technical Specification 5.6.5. Analysis with a limiting combination of conditions – **[[**

]]^A – has shown that it would take a minimum of 4.5 seconds to reach the 29.3% strain limit.

- For Westinghouse fuel, sensitivity studies show that variations from the limiting combination of conditions corresponding to the 4.5-second criterion result either in longer times to reach the 29.3% strain limit, or strain rates in excess of **[[ ]]^A**. The minimum time predicted to reach the 29.3% strain limit is sensitive to the **[[ ]]^A**.
- The limiting combination of conditions that resulted in the 4.5-second time-in-DNB criterion are not likely to occur simultaneously for the PVNGS safety analyses that consider DNB propagation.

<u>Basis for Existing 4.5-second Time-in-DNB Criterion (Westinghouse Fuel)</u>

- PVNGS Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," invokes CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990 (Reference 9), as the methodology for Technical Specification 3.2.1, "Linear Heat Rate (LHR)."
- The response to NRC Question 3 in CEN-372-P-A, Appendix A, "Response to NRC Questions on CEN-372-P, 'Fuel Rod Maximum Allowable Gas Pressure,'" discusses the mechanistic assessment of clad ballooning performed by Combustion Engineering (CE). **[[ ]]^A** was used for sensitivity studies on DNB propagation for CE 16x16 fuel.
- The ranges of user-specified inputs, used in the sensitivity studies, are as follows:
  - Rod internal pressure – 2350 psia to 3000 psia
  - Reactor Coolant System (RCS) pressure – 1800 psia to 2300 psia
  - Coolant quality – -0.1 to 0.4
  - Coolant mass flux – $1.4x10^6$ to $2.5x10^6$ lbm/hr-ft^2
  - Cladding heat flux – $250x10^3$ to $900x10^3$ Btu/hr-ft^2 **[[ ]]^A**

- CEN-372-P-A describes a Norton-type strain rate equation, where the strain rate is driven primarily by the hoop stress in the cladding and the cladding temperature. For the range of applicability identified above, **[[**

  ]][A].

- The CE time-in-DNB criterion of 4.5 seconds was based on the worst postulated combination of inputs that would minimize the time it would take to reach the CEN-372-P-A strain limit of 29.3%, without **[[**

  ]][A].

- Sensitivity studies showed that for each cladding ΔP analyzed, the time to reach the 29.3% strain limit was **[[**

]]^A. See Figure 1.

- With regard to the statement above that the time to reach the 29.3% strain limit was **[[**

  ]]^A.

Conservatism in the 4.5-second Time-in-DNB Criterion (Westinghouse Fuel)

- The limiting combination of conditions that resulted in the 4.5-second time-in-DNB criterion are not likely to occur simultaneously for the PVNGS safety analyses that consider DNB propagation. For example:
  - Achieving **[[**

    ]]^A But the CEA Ejection analysis credits a CPC Variable Overpower Trip (VOPT), the Loss of Flow from the Specified Acceptable Fuel Design Limit (SAFDL) analysis credits a CPC Departure from Nucleate Boiling Ratio (DNBR) trip, and the Seized Rotor/Sheared Shaft analysis credits a low flow signal from the CPCs or a low Steam Generator (SG) ΔP trip, and none of these events are expected to approach a RCS low pressure trip of 1800 psia. Steam Line Break analyses, which credit either a Reactor Coolant Pump (RCP) low shaft speed trip or a VOPT, are discussed below with respect to time in DNB. The lower end of the normal RCS operating pressure allowed by Technical Specification LCO 3.4.1 is 2130 psia, not 1800 psia. For comparative purposes, it is noted that the evaluation of a CEA Ejection transient for CE16HTP yielded a **[[** **]]**^{A}.
  - The CE algorithm does not **[[**

]]^{A}.

PVNGS Safety Analysis Time-in-DNB

- Traditionally, PVNGS safety analyses have not challenged the 4.5-second time-in-DNB criterion for Westinghouse fuel. Analyses that consider fuel failure at PVNGS are CEA Ejection, Loss of Flow from SAFDL, Seized Rotor/Sheared Shaft, and Steam Line Break.
    - The CEA Ejection is a power excursion event in which the power rise is turned around by Doppler reactivity. The rapid power rise, however, also activates the CPC VOPT in less than 0.5 second. Accounting for trip delay and CEA drop time, the CEAs are dropping into the core within about 2 seconds after the initiation of the transient, and by 4 seconds the system is out of DNB.
    - For Loss of Flow from SAFDL, the transient starts at time zero. At time zero a CPC DNBR trip is also initiated because the initial conditions are at the SAFDL. By about 1.5 seconds the CEAs are inserting into the core and by 4 seconds the system is out of DNB.
    - Sheared Shaft/Seized Rotor triggers either a low flow signal from CPCs or a low SG ΔP trip. Similar to the Loss of Flow from SAFDL event, the system is out of DNB in approximately 4 seconds because of the early trip at the onset of the transient.
    - Steam Line Break pre-trip analyses maximize the potential for a short-term power excursion, a decrease in the hot channel minimum DNBR, and radiological consequences. For loss of offsite power cases, the CPCs will generate a trip almost immediately as a result of RCP low shaft speeds. These cases do not violate the DNBR SAFDL. For cases with offsite power available, the CPCs will generate a VOPT. These VOPT cases have the potential to violate the DNB SAFDL and result in fuel failure; however, DNBR would turn around in a time frame similar to that of Loss of Flow from SAFDL.

Proposed 4.5-second Time-in-DNB Criterion (Framatome CE16HTP Fuel)

- PVNGS CE16STD and CE16HTP both have a fuel rod outside diameter of 0.382 inch and a rod pitch of 0.506 inch.
- CENPD-372-P-A notes that the contact strain required for a ballooning rod to contact a non-ballooning rod in a 16x16 fuel assembly is approximately **[[** **]]**^{A}, where strain is an engineering strain defined as the change in the clad radius divided by the original radius). That is, the clad radius must change by **[[**

  ]]^{A}.
- CENPD-372-P-A reported a calculated clad strain of 29.3% for a ballooned 14x14 fuel rod that was **[[** **]]**^{A} during a steam line rupture event, with a rod ΔP of **[[** **]]**^{A}. The topical report stated that **[[**

  ]]^{A} The NRC Safety Evaluation for CENPD-372-P-A stated ". . . We have evaluated C-E's claim and agree that DNB propagation will not occur based on their maximum calculated uniform cladding strains and, therefore, their maximum calculated flow blockage into surrounding subchannels of the 14x14 and 16x16 ballooned rods. We have also evaluated the minimum calculated distance from the ballooned rods to adjacent rods and concluded that sufficient distance remains to prevent DNB propagation to these adjacent rods for the limiting postulated accidents. . . . Therefore, if future fuel reload applications have calculated uniform cladding strains or values of percent flow blockage greater than those calculated in this analysis . . . further justification will be required from the licensee on DNB propagation in postulated accidents. . . ."
- Framatome EDGAR test data includes a parameter called Uniform Elongation or Ar (%), which is **[[**

  ]]^{A}.

- For a fuel assembly channel between 4 fuel rods, the cross-sectional flow area without any deformed pins is (0.506 inch)^2 – (π (0.382 inch)^2/4), or 0.14143 inch^2. With a single deformed pin at an engineering strain of 29.3% **[[**

  ]]^A. For comparative purposes, it is noted that the Westinghouse deformed grid coolability study showed that **[[**

  ]]^A.
- The EDGAR test simulation presented in the APS letter of October 4, 2019 (Reference 1), based on an ASTM publication by Forgeron et al (Reference 8), shows that the strain rate (slope of the strain curve) of M5® cladding increases as the cladding deforms. This is the same cladding creep behavior exhibited by Zircaloy-4 cladding as modeled in the CE algorithm, and arises from the use of a Norton-type equation both by Forgeron and by CE. Specifically, as the cladding begins to deform outward, the cladding wall thickness is reduced and the hoop stress increases. The result is approximately an exponential curve of strain as a function of time, starting with a relatively low strain rate and ending with a relatively large strain rate and clad burst. Indeed, Figure 11 of the Forgeron paper shows that if one knows the initial hoop stress of the cladding (for example, approximately 35 MPa for a cladding ΔP of 50 bar or 5 MPa), whether Zircaloy-4 or M5® cladding and over a wide range of temperatures, then one can make a reasonable estimate as to how long it will take the cladding to rupture.
- Because the creep behavior of M5® cladding can be described with a Norton-type exponential equation, with the strain rate increasing as a function of time, it follows that the strain rate early in an EDGAR experiment would be less than the average (linear) strain rate defined by zero strain at time zero and the measured Uniform Elongation strain (Ar) at the time of cladding burst. Table 1 summarizes the average strain rates for ten EDGAR experiments with M5® cladding for which APS has obtained data directly from Framatome, as well as the EDGAR experiment described in the Forgeron et al paper (summarized in the APS letter of October 4, 2019). Note that Table 1 reflects a maximum average strain rate of **[[** **]]^A**.
- For the EDGAR tests summarized in Table 1, **[[**

]][F].

- For a 4.5-second time period in DNB, the cladding strain can be conservatively estimated as **[[**

  ]][A]. For comparative purposes, it is noted that the total strain calculated for CE16STD and CE16NGF for **[[**
  ]][A] (see, for example, CENPD-372-P-A (Reference 9, Table 3-3)). Cladding burst for CE16HTP fuel is not anticipated to occur during a 4.5-second period because the strain rate and accumulated strain is so low.

Figure 1

[[

]]A

Table 1

[[

]]F

SRXB RAI-5

The regulatory basis for this RAI is 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The approved method for utilizing statistical convolution to predict DNBR assumes a single fuel type. It is not clear how multiple fuel types with separate CHF correlations and probability density functions are considered in this analysis. Please provide additional information to clarify how mixed cores will be analyzed to determine DNBR. Discuss the methods as applied to a mixed core with both Framatome and Westinghouse fresh fuel bundles.

Supplemental Response to SRXB RAI-5

As stated in the response to SNPB RAI-1, APS is not requesting authorization to operate PVNGS with mixed batches of fresh fuel in the reactor core of any unit. Lead Test Assemblies per Technical Specification 4.2.1.b are not, however, considered mixed fresh fuel.

Assemblies of different types have different DNBR probability distribution functions (pdfs), which requires an examination of each fuel type. The process of statistical convolution for cycle-specific fuel failure calculations remains unchanged whether an assembly of interest is in a transition core or a uniform core, but now proceeds in a branched fashion analogous to the branching in the CETOP-D benchmarking process (as described in the response to SNPB RAI-7 Part "b"). Fuel failure calculations are performed for assemblies identified as potentially limiting per the logic described in the response to SNPB RAI-7 Part "a". Thermal-hydraulic VIPRE models are created for each candidate in each branch, using the CHF correlation and modeling options appropriate for that fuel type. These models are used to generate radial peaking (Fr) versus DNBR pairs for each candidate. The Fr versus DNBR pairs are combined with the pin census and the respective DNBR probability distribution to perform the fuel failure calculations themselves. By evaluating the mixed core pin census with all fuel type DNBR pdfs, the branch that produces the most conservative results can be identified. Failed fuel pin results from the conservative branch will be compared to the fuel failure percentage limits to ensure they are satisfied, thus ensuring acceptable offsite dose consequences.

A demonstration fuel failure calculation for a mixed core with three different fuel types has been included in the UFSAR Chapter 15E analysis of record for the Framatome CE16HTP fuel design that was reviewed during the June 2019 audit.

For the CEA Ejection transient, it is possible for assemblies not identified in the limiting assembly screening to enter DNB during this event. Therefore, the fuel failure evaluation for CEA Ejection will assess all resident fuel types in the core.

CHF RAI-04

Application of Other CHF Model into Other Computer Codes	
When the NRC approves a CHF model, it is approved as a CHF model/subchannel code combination. The NRC staff recognizes that the behavior of the CHF model is dependent on the subchannel code's performance and limits the approval to the combination for which validation has been presented to the NRC. Typically, an applicant who desired to use a different CHF model in the same subchannel code, or the same CHF model in a different subchannel code will then submit an application to the NRC to change its approved methods. APS has made such a submittal in this LAR. However, APS also stated in this LAR that they would like the ability to use other CHF model/subchannel code combinations, which have not been previously reviewed by the NRC staff. APS should provide further details on the request of application of CHF models into subchannel codes without a submission to the NRC. Specifically, APS should list the complete set of combinations they may wish to use in the future, how they will demonstrate adequate validation to the NRC, and how they will maintain compliance with 10 CFR 50.36.	
Associated Section	Use of WLOP CHF Correlation
Associated Regulations and Guidance	10 CFR Part 50, Appendix A, GDC 10; 10 CFR 50.36; 10 CFR 50.34; and 10 CFR Part 50, Appendix B

Response to CHF RAI-04

Approval of CETOP-D with BHTP CHF Correlation

Addition of BHTP into the CETOP-D code (CEN-160(S)-P as listed in TS 5.6.5.b.16) **[[** **]]^A** as addressed in the response to SET RAI-01. CETOP-D with BHTP may be used in transient analyses as an alternative to VIPRE, with the reload process incorporating the **[[** **]]^A**. The detailed implementation documentation consists of:

- Code Software Quality Assurance (SQA) documentation
- Basedeck analysis
- **[[** **]]^A**
- **[[** **]]^A**
- Setpoint analyses

This material was provided for NRC review during the June 2019 audit. It is APS's position that issuance of the license amendment will constitute NRC approval of the use of the BHTP CHF correlation in the CETOP-D code.

Approach to Implementation of Other CHF Correlations into Other Computer Codes

APS hereby withdraws its request for NRC staff approval of a proposed process for implementing new combinations of CHF correlations and thermal-hydraulic codes. APS is withdrawing its request to support issuance of the license amendment in a timely manner. As such, the associated regulatory commitment (see Enclosure Attachment 1 to the APS letter of October 4, 2019) is also withdrawn.

References

1. Letter 102-07986-MLL/MDD from M. L. Lacal (APS) to Document Control Desk (NRC), *Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, Docket Nos. STN 50-528, 50-529, and 50-530, Renewed Operating License Nos. NPF-41, NPF-51, and NPF-74, Response to NRC Request for Additional Information Regarding License Amendment and Exemption Requests Related to the Implementation of Framatome CE16HTP Fuel*, October 4, 2019 [NRC ADAMS Accession Nos. ML19277J457 (non-proprietary) and ML19277J458 (proprietary)]
2. ANP-10337P-A, Revision 0, *PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations*, April 2018 [NRC ADAMS Accession Nos. ML18144A821 and ML18128A242]
3. CENPD-178-P, Revision 1-P, *Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading*, August 1981 [NRC ADAMS Public Legacy Library Accession No. 8108270312]
4. NUREG-0857, Supplement No. 6, *Safety Evaluation Report Related to the Operation of Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Arizona Public Service Company, et al*, October 1984 [NRC ADAMS Accession No. ML091330139]
5. Letter from S. P. Lingam (NRC) to R. S. Bement (APS), *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)*, January 23, 2018 [NRC ADAMS Accession Nos. ML17319A103 (proprietary) and ML17319A107 (non-proprietary)]
6. Letter 102-06022-TNW/RKR from T. N. Weber (APS) to Document Control Desk (NRC), *Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, Docket Nos. STN 50-528/50-529/50-530, Emergency Core Cooling System (ECCS) Performance Evaluation Models, 10 CFR 50.46(a)(3)(ii) Annual Report for Calendar Year 2008*, June 18, 2009 [NRC ADAMS Accession No. ML091810703]
7. *Draft Safety Evaluation by the Office of Nuclear Reactor Regulation for Topical Report PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs," Pressurized Water Reactor Owners Group (PWROG)*, August 22, 2018 [NRC ADAMS Accession No. ML19071A240]
8. Forgeron, T., Brachet, J. C., Barcelo, F., Castaing , A., Hivroz, J., Mardon, J. P., and Bernaudat, C., *"Experiment and Modeling of Advanced Fuel Rod Cladding Behavior Under LOCA Conditions: Alpha-Beta Phase Transformation Kinetics and EDGAR Methodology,"* Zirconium in the Nuclear Industry: Twelfth International Symposium, ASTM STP I 354, G. P. Sabol and G. D. Moan, Eds., American Society for Testing and Materials, West Conshohocken, PA, 2000, pp. 256-278

9. CEN-372-P-A, *Fuel Rod Maximum Allowable Gas Pressure*, May 1990 [NRC ADAMS Public Legacy Library Accession Nos. 90062600756 and 9006260076]

ENCLOSURE
ATTACHMENT 9

Framatome ANP-3640Q2NP, Revision 0
Palo Verde Units 1, 2 and 3
Small Break LOCA Summary Report
NRC RAI Responses

[NON-PROPRIETARY VERSION]

framatome

Palo Verde Units 1, 2 and 3 Small Break LOCA Summary Report NRC RAI Responses

ANP-3640Q2NP
Revision 0

Licensing Report

November 2019

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

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Nomenclature

Acronym	Definition
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
HHSI	High Head Safety Injection
HPSI	High Pressure Safety Injection
LHSI	Low Head Safety Injection
LPSI	Low Pressure Safety Injection
LOCA	Loss of Coolant Accident
LSC	Loop Seal Clearing
NRC	Nuclear Regulatory Commission
PCT	Peak Cladding Temperature
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
SBLOCA	Small Break Loss of Coolant Accident
SC	Screening Criterion
SI	Safety Injection Actuation Signal

1.0 SUMMARY

This report contains the response to a follow-on question to RAI 19 as asked by the NRC for the Palo Verde Small Break LOCA Analyses.

2.0 FOLLOW ON QUESTION TO SNPB-RAI-19

Request:

The response omitted part of the requested information necessary for the NRC staff to complete its review. In particular, the NRC staff requested that the licensee estimate the observed change in peak cladding temperature (PCT) associated with an S-RELAP5 code modification autonomously implemented by Framatome following the NRC staff's review and approval of the small-break LOCA evaluation model described in EMF-2328(P)(A).

Both conservative requirements in Appendix K to 10 CFR Part 50 and guidance in Regulatory Guide 1.203, "Transient and Accident Analysis Methods," December 2005 (ADAMS Accession No. ML053500170), reflect the importance of performing comparisons of evaluation model predictions against relevant test data. The assessment of the EMF-2328 evaluation model against test data, which constitutes part of the NRC staff's basis for finding the evaluation model acceptable, is specifically discussed in Section 4.5 of the NRC staff's safety evaluation on Revision 0 and Section 5.3 of the NRC staff's safety evaluation on Supplement 1. The validation of the evaluation model is further discussed in the submitted topical reports and a number of RAIs and responses concerning the EMF-2328 methodology.

Confirmation of the impact of the autonomously implemented code modification on the calculated PCT and other relevant figures of merit specified in 10 CFR 50.46(b) is necessary to confirm whether (1) the existing evaluation model assessment remains valid or (2) a new assessment is necessary with the modified evaluation model the licensee proposes to apply to Palo Verde.

Response:

The identified issue, [

] Benchmarks of separate effects and integral tests were performed with S-RELAP5 as part of EMF-2328, Rev. 0 (Reference 1) and EMF-2328, Supplement 1 (Reference 2) to demonstrate the adequacy of various aspects of the code capabilities and modeling. The following RAI response demonstrates that the benchmarks are not affected [ ] and therefore there is no impact on the calculated PCTs used in support of the evaluation model approval.

Test facilities are designed to be representative of a PWR, [

] Furthermore, the tests themselves are designed to focus on specific aspects and phenomena within a portion of the event. This leads to test setups and simplified benchmark models that are not fully characteristic of a plant and the conditions during an SBLOCA. [

]

[

]

[

] provide conclusive evidence that none of the EMF-2328, Rev. 0 and EMF-2328, Supplement 1 benchmarks are impacted [

] Therefore, there is no change in any cladding temperature calculations in the associated benchmarks and the existing NRC evaluation model approval remains valid.

Table 2-1: [ ]

[ ]

Figure 2-1: [ ]

[ ]

3.0 REFERENCES

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