Enclosure Attachments 11, 12, 13, and 14 contain PROPRIETARY information to be withheld under 10 CFR 2.390

When separated from Attachments 11, 12, 13, and 14, this transmittal is decontrolled.

10 CFR 50.90
10 CFR 50.12

MARIA L. LACAL
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102-07986-MLL/MDD
October 4, 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, *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)*, dated August 29, 2019, [ADAMS Accession Numbers ML19234A320 (non-proprietary) and ML19234A321 (proprietary)].

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
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 regulatory audit at the PVNGS site in June 2019 and subsequently issued a request for additional information (RAI) to APS on August 29, 2019 (Reference 2). Reference 2 requested that APS provide a response within thirty (30) days. APS contacted the NRC staff on September 3, 2019, to request an extension to October 4, 2019. The NRC staff authorized the extension.

The enclosure to this letter, which includes fourteen (14) attachments, provides the APS responses to the NRC staff individual RAIs. Attachment 1 includes three new regulatory commitments (as defined by NEI 99-04, Revision 0, *Guidelines for Managing NRC Commitment Changes*) that 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). The modified TS, and the APS 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 and Framatome affidavits have been attached to the enclosure to support withholding of proprietary information from public disclosure pursuant to 10 CFR 2.390. Attachments 11, 12, 13, and 14 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: October 4, 2019

Sincerely,

Maria Lacal

MLL/MDD/mg

Enclosure: Response to NRC Staff 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

Response to NRC Staff 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 several 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]. This Enclosure and its Attachments provide the APS responses to the RAIs.

APS Response to NRC RAIs

The following sections provide a high level summary of the APS responses to the 64 RAIs that the NRC staff issued on August 29, 2019 [Reference 15]. Matters that may be of particular interest to the NRC staff (e.g., the status of condition reports that were in a discovery or evaluation phase at the time of the June 2019 regulatory audit) are briefly described below and addressed further in the attachments to this Enclosure. The following sections are 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], and May 17, 2019 [Reference 12], and are supplemented further by these APS responses.
The APS responses do not affect the conclusions of the 10 CFR 50.91(a) no significant hazards consideration determination provided in the LAR.

Regulatory Commitments (Enclosure Attachment 1)

Attachment 1 includes three new regulatory commitments (as defined by NEI 99-04, Revision 0, *Guidelines for Managing NRC Commitment Changes*) that support NRC staff approval of the license amendment request.

Technical Specifications (Enclosure Attachments 2 and 3)

The LAR [Reference 4] requested NRC approval of 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 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 regulatory audit, 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. APS decided to modify the TS proposed in the LAR to better clarify their intent and reduce the potential for future misinterpretation or misunderstanding. APS notes that, upon approval of the LAR, there would be three NRC-approved fuel types for the PVNGS units (i.e., Westinghouse

CE16STD, Westinghouse CE16NGF, and Framatome CE16HTP). The attached APS responses affirm that, with the exception of lead test assemblies (LTAs) allowed by TS 4.2.1, mixed core designs would include only one fresh fuel type. The APS responses also further modify the proposed TS 5.6.5 to add a requirement for the unit-specific Core Operating Limits Reports (COLRs) to identify the fuel manufacturers, cladding materials, and associated NRC-approved methods for non-LTA fuel used in the reactor cores.
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. Consistent with the PVNGS *Quality Assurance Program Description*, the TS mark-ups have been reviewed and approved by the PVNGS Plant Review Board (PRB).

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.

Affidavits (Enclosure Attachments 5 and 6)

Enclosure Attachments 5 and 6 provide, respectively, APS 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 responses to the NRC RAIs, and is contained in Enclosure Attachments 11, 12, 13, and 14. Non-proprietary versions of those Enclosure Attachments are provided as Enclosure Attachments 7, 8, 9, and 10.

APS Responses (Enclosure Attachments 7, 8, 9, 10, 11, 12, 13, and 14)

The APS responses to the 64 NRC individual RAIs are provided in Enclosure Attachments 7 (non-proprietary) and 11 (proprietary). APS responses to NRC RAIs related to large break loss of coolant accidents are cross-referenced to Framatome material in Enclosure Attachments 8 (non-proprietary) and 12 (proprietary). Likewise, APS responses related to small break loss of coolant accidents are cross-referenced to Framatome material in Enclosure Attachments 9 (non-proprietary) and 13 (proprietary).

During the June 2019 NRC regulatory audit, NRC staff noted that the proposed APS analysis methodology for control element assembly (CEA) ejection accidents did not have the capability to model certain phenomena, such as melting at the outer rim of fuel pellets following a prompt critical excursion. It is the APS position that the proposed analysis methodology provides conservative results that may be used to assess compliance with NRC interim acceptance criteria. In support of this position, APS is providing a Framatome three dimensional (3D) CEA ejection summary report in Enclosure Attachments 10 (non-proprietary) and 14 (proprietary). APS is not seeking NRC review and approval of the 3D methodology or analytical results described in Enclosure Attachments 10 and 14 as part of the PVNGS licensing basis. APS is providing a summary of the 3D analysis only for the

purpose of informing the NRC decision-making process regarding the level of conservatism inherent in the proposed APS analysis methodology, which utilizes the Framatome COPERNIC code to evaluate CE16HTP fuel performance.

Condition Reports

During the June 2019 regulatory audit at the Palo Verde site, APS and Framatome personnel notified NRC staff that three condition reports (CRs) had been initiated shortly before the audit began, and were then in the discovery and evaluation phases. The NRC staff was also told that these CRs would be addressed with the responses to any RAIs that were issued following the audit.

Two of the CRs, Framatome 2019-840 and 2019-1130, were associated with the realistic large break loss of coolant accident (RLBLOCA) calculations performed for APS. Framatome evaluated both CRs against the PVNGS RLBLOCA analysis, and confirmed that the figures of merit reported in the July 2018 LAR would remain applicable and valid as new licensing bases for PVNGS. Further information regarding these CRs is provided in Enclosure Attachments 8 (non-proprietary) and 12 (proprietary).

The third CR, APS 19-09079, was associated with differences identified by Framatome between how Framatome originally implemented the BHTP CHF correlation in the LYNXT thermal-hydraulic code, and how APS and its contractors implemented the same correlation in the VIPRE-01 and VIPRE-W codes. APS decided to modify its implementation and re-perform the thermal-hydraulic analyses that had originally supported the July 2018 LAR. The APS responses to the NRC RAIs in Enclosure Attachments 7 (non-proprietary) and 11 (proprietary) reflect the results of the revised analyses. Although some calculated values changed, the PVNGS departure from nucleate boiling ratio (DNBR) analytical limit of 1.27 for Framatome CE16HTP fuel did not change from the LAR.

Westinghouse Analyses

This document does not include a complete response to some NRC individual RAIs (i.e., SNPB RAIs-10, 17, 28, 29 and 30) related to faulted condition analyses (i.e., seismic and loss of coolant accident loads) and 10 CFR 50.46 emergency core cooling system (ECCS) performance. These responses are pending completion of supporting analyses for Westinghouse fuel that will be co-resident with Framatome fuel in transition cores. Westinghouse is working toward completion of those analyses and APS will make them available for NRC audit later in 2019. The scope, schedule, and location for this audit will be coordinated with the NRC staff prior to November 1, 2019, as described in Attachment 1.

Enclosure Attachments

Attachment 1	–	Regulatory Commitments
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 11 as a Proprietary Document
Attachment 6	–	Affidavits from Framatome Submitted in Accordance with 10 CFR 2.390 to Consider Enclosure Attachments 11, 12, 13, and 14 as Proprietary Documents
Attachment 7	–	APS Responses to NRC Requests for Information [NON-PROPRIETARY VERSION]
Attachment 8	–	Framatome ANP-3639P, Revision 1Q1NP, Revision 0, Palo Verde Units 1, 2 and 3 Realistic Large Break LOCA Summary Report NRC RAI Responses [NON-PROPRIETARY VERSION]
Attachment 9	–	Framatome ANP-3640Q1NP, Revision 0, Palo Verde Units 1, 2 and 3 Small Break LOCA Summary Report NRC RAI Responses [NON-PROPRIETARY VERSION]
Attachment 10	–	Framatome ANP-3785NP, Revision 0, Rod Ejection Accident (AREA) Analysis for Palo Verde [NON-PROPRIETARY VERSION]
Attachment 11	–	APS Responses to NRC Requests for Information [PROPRIETARY VERSION]
Attachment 12	–	Framatome ANP-3639P, Revision 1Q1P, Revision 0, Palo Verde Units 1, 2 and 3 Realistic Large Break LOCA Summary Report NRC RAI Responses [PROPRIETARY VERSION]
Attachment 13	–	Framatome ANP-3640Q1P, Revision 0, Palo Verde Units 1, 2 and 3 Small Break LOCA Summary Report NRC RAI Responses [PROPRIETARY VERSION]
Attachment 14	–	Framatome ANP-3785P, Revision 0, Rod Ejection Accident (AREA) Analysis for Palo Verde [PROPRIETARY VERSION]

<u>References</u>

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)]

ENCLOSURE
ATTACHMENT 1

Regulatory Commitments

ENCLOSURE ATTACHMENT 1
REGULATORY COMMITMENTS

The following table identifies the regulatory commitments in this document. Any other statements in this submittal represent intended or planned actions. They are provided for information purposes and are not considered to be regulatory commitments.

REGULATORY COMMITMENT	TYPE		SCHEDULED COMPLETION DATE (if applicable)
	one-time	continuing compliance	
APS shall apply a radial power fall off (RFO) curve penalty that accommodates the anticipated impacts of thermal conductivity degradation (TCD), to any future Westinghouse-supplied fuel design introduced at PVNGS to which the FATES3B fuel performance code would be applied.		X	N/A
APS shall institute administrative controls for implementation of approved critical heat flux (CHF) correlations into approved codes. This process shall be as described in the APS response to NRC Request for Additional Information CHF RAI-04 (APS letter 102-07986), and shall be limited to the following CHF and code combinations: a. ABB-NV into VIPRE-01 / VIPRE-APS b. WSSV into VIPRE-01 / VIPRE-APS c. CE-1 into VIPRE-01 / VIPRE-APS (or qualification of the existing VIPRE-01 / VIPRE-APS implementation) d. CE-1 into VIPRE-W (or qualification of the existing VIPRE-W implementation) e. BHTP into TORC	X		6/30/2020
APS cannot completely respond at this time to some RAIs (i.e., SNPB RAIs-10, 17, 28, 29 and 30). Westinghouse is working toward completion of the analyses and APS will make them available for NRC audit. The scope, schedule, and location for the audit will be coordinated with the NRC staff prior to November 1, 2019.	X		11/1/2019

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 approved coated or uncoated zirconium-alloy clad 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.

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 approved coated or uncoated zirconium-alloy clad 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]

5.6 Reporting Requirements

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]

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.

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.~~

~~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 "B" AS A NEW PARAGRAPH

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.

~~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.

INSERT "D" AS A NEW PARAGRAPH

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 11
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 11 to the enclosure for APS Correspondence 102-07986, "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 4TH day of October, 2019.

Donna Norman
NOTARY PUBLIC, STATE OF ARIZONA
MY COMMISSION EXPIRES: 8-9-2021
Reg. #:

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

ENCLOSURE
ATTACHMENT 6

Affidavits from Framatome
Submitted in Accordance with 10 CFR 2.390
to Consider Enclosure Attachments 11, 12, 13, and 14
as Proprietary Documents

AFFIDAVIT

1. My name is Gayle Elliott. I am Deputy Director, Licensing & Regulatory Affairs, 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 Licensing Report ANP-3639P, Revision 1Q1P, Revision 0, entitled, "Palo Verde Units 1, 2 and 3 Realistic Large Break LOCA Summary Report NRC RAI Responses," dated September 2019, Licensing Report ANP-3640Q1P, Revision 0, entitled "Palo Verde Units 1, 2 and 3 Small Break LOCA Summary Report NRC RAI Responses," dated September 2019, and the Framatome information provided in Enclosure Attachment 11 to APS Letter Number 102-07986, entitled, "Response to NRC Request for Additional Information Regarding License Amendment and Exemption Requests Related to the Implementation of Framatome CE16HTP Fuel," and 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.

Gayle Elliott

Commonwealth of Virginia)
) ss.
City of Lynchburg)

SUBSCRIBED before me this 25 day of September, 2019.

Heidi Hamilton Elder

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

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

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Nathan E. Hottle. I am Manager, Product Licensing, 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 following document: ANP-3785P Revision 0, "Rod Ejection Accident (AREA) Analysis for Palo Verde," referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome Inc. for the control and protection of proprietary and confidential information.

4. This Document contains 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 this 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 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 this Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(c) and 6(d) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document 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.

SUBSCRIBED before me this 29th

day of May, 2019.

Heidi Hamilton Elder

Heidi Hamilton Elder
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/2022
Reg. # 7777873

HEIDI HAMILTON ELDER
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NOTARY PUBLIC

ENCLOSURE
ATTACHMENT 7

APS Responses to NRC Requests for Information

[NON-PROPRIETARY VERSION]

SNPB RAI-1

TS 5.6.5.b states that "The analytical limits used to determine the core operating limits shall be those previously reviewed and approved by NRC." In the Enclosure, "Description and Assessment of Proposed License Amendment," of the LAR dated July 6, 2018, the licensee stated, in part:

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

a. The licensee states that ". . . the approved PVNGS reload analysis methodology to address both Westinghouse and Framatome fuel, including the implementation of Framatome methodologies. . . ." This statement implies that there is an approved Palo Verde methodology for Framatome fuel designs. The NRC staff is not aware of an approved Palo Verde methodology for Framatome fuel designs. Please clarify this statement to remove the ambiguity with respect to approved methodology for Framatome fuel designs.

b. Please elaborate on your plan to use both Framatome and Westinghouse fuel designs and their respective methodologies in the Palo Verde units.

Response to SNPB RAI-1

Part a Response

The intent of this statement was to say that the approved Palo Verde reload methods, which are for Westinghouse fuel, would be adapted to apply to Framatome fuel as described in the License Amendment Request (LAR) as supplemented by subsequent APS submittals. APS does not have an approved reload methodology for Framatome fuel.

Part b Response

The LAR, and the responses to Requests for Additional Information, address the detailed information regarding the proposed implementation of the Framatome fuel into the Palo Verde reload methods. As discussed in the response to SNPB RAI-25, 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 (TS) 4.2.1.b are not considered mixed fresh fuel.

This response will address the overall administrative controls and how APS plans to use both Framatome and Westinghouse fuel.

The normal reload process is, and will continue to be, based on using a full reload batch from a single fuel supplier. The normal APS 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. Our general approach is to have no more than two fuel types in any core, excluding any lead

test assemblies. Unexpected external drivers, such as fuel design or performance issues, may require a modification to this strategy.

As described in the response to SNPB RAI-22, APS proposes to modify TS 4.2.1 and TS 5.6.5 to add a requirement for the unit-specific Core Operating Limits Report (COLR) to identify the fuel manufacturer, cladding material, and associated methods for the non-Lead Test Assembly (LTA) fuel in use in the reactor.

The preceding process will ensure that the methodologies are used in accordance with their NRC approval, and will provide an easily accessible method of documenting which fuel types and cladding are in the reactor, and which methodologies are being used with each fuel type. By maintaining this information in the unit-specific COLRs, this information will be routinely sent to the NRC as a part of the COLR submissions required by TS 5.6.5.d.

For each reload, the reload analyses are reviewed against the Updated Final Safety Analysis Report (UFSAR) and TS per 10 CFR Part 50.59. This review will verify that conformance with the approved methods for each fuel type has been maintained.

SNPB RAI-2

Criterion 10 of Appendix A to 10 CFR Part 50 requires "that specified acceptable fuel design limits [SAFDLs] are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences." Section 4 of Attachment 5, "Assessment of Topical Report Limitations and Conditions," of the LAR dated July 6, 2018, briefly discusses the BHTP (designation for a Framatome Critical Heat Flux (CHF) Correlation) correlation verification for VIPRE-01 (Versatile Internals and Component Program for Reactors; Electric Power Research Institute (EPRI)) and VIPRE-W (Versatile Internals and Component Program for Reactors; Westinghouse) codes based on the same CHF test points used in the development of HTP™ CHF correlation for use with the thermal-hydraulic subchannel code XCOBRA-IIIC.

a. Provide details on how a CHF design limit was determined from the CHF test points.

b. Provide details on how the statistical design limit for departure from nucleate boiling ratio (DNBR) was calculated as per Section 4.2 of Attachment 5 and Section 5.4.1 of Attachment 10.

Response to SNPB RAI-2

Part a Response

The VIPRE-APS sub-channel code was used to simulate the experimental tests, which are documented in Reference 1, and compute a BHTP predicted CHF for each of the **[[ ]]** test points. For each test, the predicted CHF was compared to the measured local heat flux as discussed in the response to BHTP RAI-01 and a predicted to measured (P/M) ratio was recorded. A mean P/M and standard deviation were then calculated based on all test points. The standard deviation of the data for each test falls between **[[ ]]** across all data. The average ratio of predicted to measured data ranges between **[[ ]]**.

The BHTP CHF correlation design limit was derived using all **[[ ]]** P/M values from the VIPRE-APS executions. The correlation design limit is the value of the P/M ratio below which, with 95% confidence, 95% of the population of P/M values will fall. The safety limit is derived using **[[ ]]**. This method has been previously used in Reference 2 to define a design limit for BHTP using the LYNXT code. **[[ ]]**.

As discussed in Reference 2, **[[ ]]** for the BHTP test database. For g of 95% at P of 95% the beta function evaluation results in a value of **[[ ]]**. That is, for the BHTP database there is a 95% probability with 95% confidence that P/M values chosen at random from the (sorted in descending order) population will fall below the **[[ ]]** largest P/M value.

For the **[[** **]]**.

With 95% confidence, at least 95% of the population of P/M ratios will be less than this value.

The same approach was used to calculate the BHTP correlation limit based on VIPRE-W subchannel code predictions. VIPRE-W and VIPRE-APS are equivalent as discussed in the responses to BHTP RAI-7 and BHTP RAI-8.

<u>Part b Response</u>

The CHF correlation design limit of **[[** **]]** is combined with other uncertainties identified in Table 5-8 of Attachment 10 of the LAR. **[[** **]]**

Additional discretionary margin was used to **[[** **]]** to determine the analytical DNBR design limit of 1.27 for CE16HTP fuel. The analytical DNBR limit is documented in the Technical Specification Bases Section B 2.1.1.

Corrections to the LAR dated July 6, 2018:

- The P/M mean of **[[** **]]** and standard deviation of **[[** **]]** documented in this response supersede the VIPRE values in Table 5-4 of Attachment 10 of the LAR.
- The P/M mean of **[[** **]]** and standard deviation of **[[** **]]** documented in this response supersede the "BHTP CHF Correlation (P/M)" values in Table 5-8 of Attachment 10 of the LAR.
- The BHTP correlation statistics discussion in Section 5.4.1.2 of Attachment 10 of the LAR is superseded with "The standard deviation of the data for each test falls between **[[** **]]** across all data. The average ratio of predicted to measured data ranges between **[[** **]]**."
- The BHTP correlation design limit of **[[** **]]** supersedes the value of **[[** **]]** documented in Section 5.4.1.2 of Attachment 10 of the LAR.
- The DNBR pdf mean of **[[** **]]** and DNBR pdf standard deviation of **[[** **]]** documented in this response supersede the values documented in Table 5-8 of Attachment 10 of the LAR.

SNPB RAI-3

The regulatory basis for this RAI includes Criterion 20, Criterion 24 and Criterion 26 of Appendix A to 10 CFR Part 50.

CEN-356(V)-P-A, Revision 01-P-A, "Modified Statistical Combination of Uncertainties," dated May 1988 and WCAP-16500-P-A, Supplement 1, Revision 1, "Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)," dated December 2010 are NRC-approved methodologies for core operating limits supervisor system (COLSS) and core protection calculator system (CPCS) setpoints analysis specifically for CE type fuel designs.

a. Please explain how the above methodologies are used for COLSS/CPCS analysis with the Framatome CE 16x16 fuel in the Palo Verde core.

b. Explain the modifications that had to be made to the above methodologies in order to analyze the digital COLSS/CPCS setpoints for the mixed core (three different fuel types; CE 16x16 STD (standard or value-added), CE 16x16 NGF and CE 16x16 HTP™) in the Palo Verde core.

Response to SNPB RAI-3

Background

The Core Operating Limits Supervisory System and Core Protection Calculator (COLSS/CPC) systems are designed to monitor the TS Limiting Conditions for Operation (LCO's) and trip the plant in the event that the linear heat rate (LHR) or DNBR safety limit is violated. The COLSS and CPC Overall Uncertainty Analysis (OUA) is performed to develop cycle specific uncertainty factors to ensure that COLSS and CPC systems perform their function at a 95/95 probability/confidence level.

The original method of calculating the DNBR uncertainty factors for COLSS and CPC was documented in CEN-356(V)-P-A, Revision 01-P-A (Reference 4). Due to the existence of both mixing vane grids and non-mixing vane grids for Westinghouse CE16NGF fuel, Westinghouse developed a more robust approach for generating conservative uncertainty factors for cores with mixing vane grids. This "augmented" OUA method as documented in WCAP-16500-P-A, Supplement 1, Revision 1 (Reference 5): **[[**

]] the plant COLSS and CPCS systems maintain the use of the CE-1 critical heat flux (CHF) correlation coefficients. **[[**

]]

For Framatome CE16HTP fuel, the COLSS and CPC OUA will adapt **[[**

]].

CEN-356(V)-P-A, Revision 01-P-A Section 2.5.2 states "The modified SCU methodology will allow calculation and application of these uncertainty factors over several burnup, power, and ASI ranges. Choice of parameters and ranges will be made on a cycle-by cycle basis in order to optimize the uncertainty factors for nominal full power operation throughout the cycle, while retaining the conservatism at a 95/95 probability/confidence level." This allowance, in particular the burnup or Time-In-Life dependent uncertainty factor option, remains for Framatome CE16HTP fuel. Time-in-life OUA uncertainty factors can assist with Margin management.

Part a Response

[[

]].

Figure 1 shows the general application of the **[[**

]]

Part b Response

[[

]] and the CE-1 CHF correlation within the COLSS/CPC system. **[[**

]]

Figure 1

COLSS/CPC DNBR OUA Process Overview

[[

]]

SNPB RAI-4

The regulatory basis for this RAI is Criterion 20, Criterion 24 and Criterion 26 of Appendix A to 10 CFR Part 50.

In the supplemental letter dated October 18, 2018, the licensee indicated that the COLSS/CPCS setpoint analysis performs two major functions: (1) **[[**

]] for the reload cycle, and (2) perform the COLSS/CPCS overall uncertainty analysis (OUA) to provide the final overall COLSS and CPCS uncertainty factors to ensure that COLSS/CPCS DNBR and linear heat rate/linear power density calculations are conservative with a 95 percent probability and a 95 percent confidence level.

a. Describe in detail how the COLSS and CPCS OUA are determined for the mixed core in the Palo Verde core.

b. Describe the process in which the COLSS/CPCS database constants are determined.

c. Provide a summary of the thermal margin assessment performed as part of the COLSS/CPCS analysis for the Palo Verde mixed core.

Response to SNPB RAI-4

Part a Response

The process of calculating the COLSS and CPCS OUA Uncertainty Factors for mixed cores is described in the responses to SNPB RAI-3, SNPB RAI-6, and SET RAI-01. For a mixed core, **[[**

]]

Part b Response

The process used for determining the COLSS and CPC database constants for mixed cores is **[[**

]]. The impact of the chosen COLSS and CPC data bases constants will be reflected in the Overall Uncertainty Analysis penalty factors developed for each fuel type described in the responses to SNPB RAI-3, SNPB RAI-6, and SET RAI-01 and summarized in the response to part (a) above.

[[

]] The plant COLSS and CPC databases will continue to use the CE-1 CHF correlation values. This process will continue to generate conservative

CPC and COLSS overall uncertainty factors thus assuring conservative calculations of CPC DNBR and linear power density (LPD), and of COLSS DNB and LHR POL Uncertainty Factors at a 95/95 probability/confidence level.

Part c Response

The Thermal Margin Assessment calculation is an operational convenience to ensure the existence of sufficient thermal margin to allow for normal plant maneuvering during the entire operating cycle for plants with a COLSS/CPC digital monitoring and protection system. This analysis is performed with best estimate operating conditions of primary pressure, inlet temperature, flow, planar radial peaking and azimuthal power tilt. It is used to assist with margin management and by itself does not have a safety function in the reload analysis process.

SNPB RAI-5

The regulatory basis for this RAI is Criterion 20, Criterion 24 and Criterion 26 of Appendix A to 10 CFR Part 50.

In the supplemental letter dated October 18, 2018, the licensee stated that the fuel dependent factors such as rod bow factors, core neutronics design, thermal-hydraulics design, and fuel performance analyses are updated to reflect fuel type prior to use in the COLSS/CPCS OUA. Explain in detail how the above update process is performed and implemented to generate the setpoints.

Response to SNPB RAI-5

The core design is performed using an approved for Palo Verde physics/neutronics design code (e.g., CASMO-4/SIMULATE-3) which explicitly models the different fuel types within the cycle specific core pattern. The use of CE16HTP fuel including gadolinia burnable poison is specifically accounted for in the neutronics analysis. **[[**

]]

The description of how these analyses feed into the COLSS and CPC Overall Uncertainty Analysis (OUA) process is shown in the response to SNPB RAI-3.

The OUA mixed core process **[[**

]] The Core Physics is a "Restart File" from the perspective of the Setpoint COLSS and CPC OUA. The fuel specifics are included in the Cycle N Restart File(s). There may be more than one physics code Restart file; for example to support analysis of Cycle N based on the preceding Cycle N-1 ending point variation. The Cycle N restart files produce the **[[**

]] within the OUA runs.

SNPB RAI-6

The regulatory basis for this RAI is Criterion 10 of Appendix A to 10 CFR Part 50.

In response to NRC Question 7, in the licensee's supplemental letter dated October 18, 2018, points to an “augmented process” for the mixed core with Westinghouse NGF and Framatome fuel. During the last regulatory audit, the licensee stated that this “augmented process” is described in the NRC-approved topical report WCAP-16500, Supplement 1, Revision 1. However, the “eight steps” process described in the topical report appears to be for the Westinghouse NGF fuel design. Justify the use of this “augmented process” for the Palo Verde core with both NGF and Framatome fuel designs.

Response to SNPB RAI-6

The term “augmented process” in the October 18, 2018 supplement refers to WCAP-16500-P-A Supplement 1 Revision 1 (Reference 5). This process **[[**

]] as outlined in the responses to SNPB RAI-3, SNPB RAI-4, and SET RAI-01. **[[**

]].

Figure 2

[[

]]

SNPB RAI-7

The regulatory basis for this RAI is Criterion 10 of Appendix A to 10 CFR Part 50.

For mixed cores with Framatome fuel,

a. Explain how the potentially limiting fuel type is identified.

b. Describe how the fuel-specific CETOP (Combustion Engineering Thermal On-Line Program) correction factors are determined for the setpoints analysis.

Response to SNPB RAI-7

Part a Response

The limiting assembly selection process for mixed cores with Framatome CE16HTP fuel **[[**

]]. Candidates are selected through a two-phase screening process: a neutronics screening, followed by a thermal-hydraulic screening.

The primary criterion applied in the neutronics screening is **[[**

]]. A set of secondary criteria that are considered at this phase address: **[[**

]]

Candidate assemblies that pass the neutronics screening are then modeled with a thermal-hydraulic code to perform a screening **[[**

]]. The result is a short-list of candidates to evaluate within the CETOP-D benchmarking process. The candidate that produces **[[**
]] is thus considered the limiting assembly. The selection of a short-list of candidates to run through the full CETOP-D benchmark process is an intentional action that serves to **[[**
]].

Part b Response

As summarized in Section 5.5 of Attachment 10 of the LAR for mixed core cycles, the mixed core is modeled **[[**

]]
as summarized in Section 5.5.2 of Attachment 10 of the LAR. The APS response to NRC Question 3 in Reference 8 summarizes the “parallel” or “branched path” analysis in which **[[**

]]. A fuel type is not considered limiting if it does not pass the neutronics screening discussed in the preceding response to Part a).

The CETOP-D benchmark process performs heat flux calculations targeting a specific DNBR with both the VIPRE and CETOP-D codes for a specific cycle core design and in the case of mixed cores for each limiting fuel type. The resultant heat flux values are compared via a ratio called a Correction Factor (CF). Mathematically, the correction factor at a certain set of operating conditions can be expressed as follows:

$$CF = \frac{q''_{VIPRE}}{q''_{CETOP}}$$

It can be seen from this equation that the use of a Correction Factor with CETOP-D will ensure that accurate or conservative heat fluxes will be generated relative to the detailed VIPRE code. The benchmarking calculations are performed **[[**

]] as input for a parallel path setpoint analysis and then compares the **[[**
]] the appropriate limiting values which bound all fuel types.

For CE16HTP mixed core designs, as stated in the APS response to Question 3 in Attachment 3 to Reference 8, the more limiting DNB limit of the fuel types present in the core design will be utilized. As an example, a mixed core containing CE16STD (1.34 DNB limit) and CE16HTP (1.27 DNB limit) fuel would implement the limiting of the two DNB limit values; in this example a value of 1.34 corresponding to the CE16STD fuel type would be implemented.

SNPB RAI-8

The regulatory basis for this RAI is Criterion 10 of Appendix A to 10 CFR Part 50.

Explain the procedures and results from the mechanical compatibility analysis of Framatome fuel with the resident CE STD and CE NGF fuel design, as well as all of the core internals.

Response to SNPB RAI-8

There are two aspects to consider:

1) Framatome analyses for mechanical compatibility.
2) Westinghouse analyses for mechanical compatibility.

Framatome Analyses

Framatome mechanical interface guidelines were used to perform the following four analyses. In addition to analytical assessments, positive Lead Fuel Assembly (LFA) operating experience was considered in this assessment. The eight (8) LFAs irradiated at PVNGS were installed in 12-finger and 4-finger CEA locations and in instrumented locations. The LFAs were handled with PVNGS fuel handling equipment.

a) Upper Tie Plate (UTP)
   i) Framatome evaluated the interfaces between the upper tie plate and the core internals, the interfaces with plant handling equipment, and CEAs.
   ii) This assessment demonstrated that the Framatome UTP design for PVNGS is compatible with the core internals, handling equipment, control components, and co-resident fuel.
   iii) Assessment criteria and results are enumerated in Table 1.
b) Lower Tie Plate (LTP)
   i) Framatome evaluated the interfaces between the lower tie plate and the core internals, the interfaces with plant handling equipment, and interfaces with core monitoring instrumentation.
   ii) This assessment demonstrated that the Framatome LTP design for PVNGS is compatible with the core internals, handling equipment, in-core instrumentation (ICI), and co-resident fuel.
   iii) Assessment criteria and results are enumerated in Table 2.
c) Corner Guide Tubes and Center Instrument Tube
   i) Framatome evaluated the interfaces between the guide tubes and the control rods, compared the resident designs to the Framatome design with regard to key attributes (dashpot / weep hole elevations), and interfaces with core monitoring instrumentation.

ii) This assessment demonstrated that the Framatome corner guide tubes and center guide tube (instrument tube) designs for PVNGS are compatible with the control component rods and core monitoring instrumentation.

iii) Assessment criteria and results are enumerated in Table 3.

d) Fuel Assembly

i) Framatome evaluated the remaining critical interfaces of the Framatome CE16HTP fuel (i.e., not previously addressed in the three preceding evaluations) as they relate to core gaps and positioning, control components, and resident fuel assemblies. Furthermore, this evaluation concluded by evaluating the Framatome CE16HTP for PVNGS against a set of functional design considerations based on engineering best practices for fuel design development.

ii) This assessment demonstrated that the Framatome CE16HTP fuel assembly design for PVNGS is compatible with the core interfaces, the control components, and resident fuel assemblies (CE16STD and CE16NGF).

iii) Assessment criteria and results are enumerated in Table 4.

<u>Westinghouse Analyses</u>

a) Fuel Mechanical Design Areas Assessment

i) Westinghouse evaluated whether the insertion of the Framatome CE16HTP fuel assemblies would adversely impact the fuel performance and mechanical integrity of the co-resident Westinghouse CE16STD and CE16NGF fuel designs.

ii) Westinghouse evaluated the general fuel rod configurations, the spacer grid elevations, the end fitting elevations, active core elevations, fuel assembly growth, and the fuel assembly holddown margins.

iii) Assessment criteria are as follows:

1. <u>General Fuel Rod Configuration</u>

The general fuel rod configuration should be such that the fuel rods are captured by the grids and that the full axial length of the fuel rods is maintained between the upper and lower end fittings. In addition, any additional length change of the fuel rod due to irradiation-induced growth and/or thermal expansion should be limited to ensure that there is ample room (shoulder gap) between the upper and lower end fittings to preclude such issues as fuel rod bowing.

2. <u>Spacer Grid Elevations</u>

For a Framatome CE16HTP fuel assembly that is adjacent to a Westinghouse CE16STD and/or CE16NGF fuel assembly, the grids shall be located at elevations where there is a sufficient amount of overlap on the inner straps. This is important for grid to grid impact forces for the

Seismic/LOCA analyses to ensure that the lateral forces are transferred in the evaluation of whether or not grid crush is predicted to occur.

3. End Fitting Elevations

   The end fitting elevations should ensure that 1) for the lower end fitting the height of the Framatome CE16HTP and the Westinghouse CE16STD and Westinghouse CE16NGF fuel designs are comparable to minimize any cross flow concerns which could affect fuel rod vibration and 2) the upper end fittings elevations are comparable to minimize any concerns with interference during operation and during fuel handling.

4. Active Core Elevations

   The relative active core elevations for the Framatome CE16HTP fuel assemblies and for the Westinghouse CE16STD and Westinghouse CE16NGF fuel designs should be aligned consistent with the needs of the nuclear design. The axial misalignment of the active core region could adversely affect the reactivity control of the core, the radiation dose to the structures, and result in inefficient fuel utilization as well affecting other characteristics which are dependent on the actual elevation of the active core region.

5. Fuel Assembly Bundle Growth

   The fuel assembly design must include sufficient allowance for irradiation-induced axial growth such that there is no solid axial interference between the fuel assembly and the core support and fuel alignment plates at any time during the life of the fuel.

iv) This evaluation includes comparison of the dimensions of interfacing features of the Framatome CE16HTP fuel assemblies and the Westinghouse CE16STD and CE16NGF fuel assemblies that will be located in the Palo Verde cores.

b) Selected Non-Seismic/LOCA Fuel and CEA Mechanical Aspects

i) It was verified that the insertion of the Framatome CE16HTP assemblies will not adversely impact the fuel performance and mechanical integrity of the co-resident Westinghouse CE16STD fuel and CE16NGF and assessed the impact of these assemblies on the safety and setpoint analyses for Westinghouse assemblies.

ii) The evaluations addressed fuel assembly lead-in features, spacer grid gaps/envelopes, CEA scram time, CEA impact load (i.e., stress), and CEA spring deflection (i.e., compression).

iii) Assessment criteria are as follows:

1. Fuel Assembly Lead-in Features

   The lead-in features of adjacent Framatome and Westinghouse fuel assemblies must be compatible with respect to prevention of hang-ups during core on-loads and off-loads.

2. Spacer Grid Envelopes

   To limit fuel assembly bow, the Framatome mid grid envelope must be large enough that the space available for bow of Westinghouse fuel assemblies in transition cores with Framatome assemblies is not significantly greater than for a full core of Westinghouse CE16STD fuel assemblies.

   At the end of any cycle of operation, for any row or column that contains both Westinghouse and Framatome CE16HTP fuel assemblies, the gaps between spacer grids in adjacent fuel assemblies and between spacer grids and the core shroud must be large enough to minimize the potential for hang-ups during core off-loads.

3. CEA Scram Times and Stresses

   Scram times for CEAs that reside entirely or partially in Westinghouse fuel assemblies must be bounded by the CEA drop curve utilized for the safety analysis.

   Stresses in CEAs due to a scram must continue to satisfy the applicable limits.

Table 1

Framatome Mechanical Compatibility Evaluation (Upper Tie Plate)

Upper Guide Structure (UGS)
1. Alignment Features: Proper engagement throughout lifetime. Adequate lead-in between corner locking nuts and UGS to accommodate engagement and bundle lean. The corner locking nut and reaction plate (hold-down plate) remains engaged with UGS throughout fuel assembly design lifetime.
2. Holddown Spring Contact: Proper spring (reaction plate) interface. The reaction plate remains in contact with the UGS throughout design life and the hold down springs do not deflect to the Upper Tie Plate unlatched condition throughout the fuel assembly design lifetime.
3. Upper Guide Structure Protrusions: No interference. The Framatome reaction plate is within the footprint of the Westinghouse (WEC) CE16NGF and CE16STD designs and differences have been accepted by APS.
4. Envelope: Consistent with resident fuel design The Framatome UTP lateral envelope is the same as the WEC CE16NGF and CE16STD designs. The Framatome UTP diagonal envelope is slightly larger than the WEC CE16NGF and CE16STD designs, but is less than the remaining components that control the overall fuel assembly envelope (e.g. spacer grids and lower tie plate).
Fuel Handling Equipment
5. Lifting Grapples: Proper Interface and no interferences with UTP internal components Based on review of the design evolution of the Framatome reaction plate drawings and the positive operating experience (OE) of the Framatome LFAs, it is concluded that the design meets the handling requirements at the Palo Verde units.
6. UTP features / components: Positive Clearance The Framatome design was evaluated to assure that the spring cups and reaction springs are always within the shadow of the reaction plate (under worse case misalignment). This precludes interference with handling equipment when inserted beyond the reaction plate.
7. Auxiliary Components: No interference. The only identified component that could interface with the UTP is the CEA. The assessment showed that the Framatome corner locking nut inner diameter (ID) is consistent with the CE16STD design, and the CE16NGF design has a smaller locking nut ID. Therefore, acceptable interface is confirmed.
8. Miscellaneous Components: Consistent with resident fuel This evaluation demonstrates that the Framatome UTP Center Fitting tip position relative to the top of the reaction plate is comparable to the WEC fuel designs. The center fitting is used during pool side reconstitution where the Framatome design has demonstrated positive performance during PIE. The other key features of the Framatome center fitting are compared to the resident designs (radial dimensions and weep holes) and shown to have the same attributes.

Table 2

Framatome Mechanical Compatibility Evaluation (Lower Tie Plate)

Lower Support Structure (LSS)
1. Alignment Features: Proper engagement throughout lifetime The LSS alignment pin position and size relative to the LTP pin hole radius is acceptable. The LSS alignment pin height and lead-in features are acceptable with the LTP mating features. The Framatome LTP bearing surface area is slightly greater than the resident (CE16STD) LTP bearing surface area on the alignment pin flange.
2. LSS protrusions: No interference The Framatome LTP foot overhangs the edge of the core pin flange consistent with the resident design. There are no known protrusions on the LSS that would affect fuel assembly interfaces as confirmed by APS.
3. LSS inlet flow holes: No obstruction of the flow holes with LTP feet / bearing surface Unique to this plant configuration, the inlet flow holes are machined through the bottom plate which is well below the core pin flanges; therefore, the LTP feet cannot obstruct the flow holes in this plant design.
4. Envelope: Consistent with resident fuel design The Framatome LTP envelopes (lateral and diagonal) are less than the WEC designs.
5. Fuel handling auxiliary equipment: No interference APS confirmed acceptance based on the positive handling interface of the Framatome LFAs.
6. Instrumentation lead-in: Consistent with resident fuel design The Framatome instrumentation lead-in angle and major through hole ID are the same as the WEC design. Furthermore, the gap between the bottom of the instrumentation nozzle and the bottom of the Framatome LTP are the same as the WEC designs. This comparison results in a consistent interface with the in-core instrumentation. The Framatome LFA experience further supports this conclusion.
7. Miscellaneous assessments The Framatome CE16HTP fuel assembly will always remain engaged with the core pin taper / lead-in during worst case lift-off conditions. The Framatome LTP was also evaluated for potential off-index conditions with respect to the LSS alignment pins. The results of that assessment provides APS with off-index indicators for inclusion in handling procedures.

Table 3

Framatome Mechanical Compatibility Evaluation (Guide Tubes and Instrument Tube)

Corner Guide Tubes: Control Component Interfaces
1. Radial and Axial Clearance: Consistent with resident design The Framatome guide tube (GT) IDs were developed based on the WEC CE16STD fuel for Palo Verde. The WEC CE16STD design has three (3) IDs over the length of the tube; a short dashpot region at the bottom end, a long mid-region, and short upper region. The Framatome guide tube IDs are the same as the WEC CE16STD fuel. The Framatome top GT ID is slightly larger than the resident CE16NGF design. This is acceptable since the Framatome design was based on the CE16STD design. This provides a larger annulus between the CEA rod OD and the GT ID in the parked elevation. The bottom (dashpot region) GT ID in the Framatome design is the same as the WEC designs. The clearance between the CEA rod OD and GT ID is sufficient. Lastly, sufficient clearance is present between the CEA rod tip and the Framatome GT lower end fitting of the GT assembly.
2. Elevation of Weep Holes: Consistent with resident design The WEC and Framatome designs have the same quantity and size of weep holes in the guide tubes. The Lower Weep hole in the Framatome design is slightly higher than the CE16STD design and slightly lower than the CE16NGF design. This is acceptable since the weep holes are in the dashpot region and within 1 inch of each other. The Framatome upper weep hole is slightly higher than the WEC designs. This is acceptable since the weep holes are in the non-dashpot region and are within 1 inch of each other.
3. Dashpot locations: Consistent with resident fuel design The Framatome dashpot start elevation is less than 0.05 inches lower than the WEC designs. This is acceptable since a lower dashpot elevation will achieve similar or better control rod drop times.
Instrument Tube: In-core Detector Interfaces
4. Radial and Axial Dimensions: Consistent with resident fuel design The dimple depression parameters on the Framatome design (minimum ID, OD, and dimple radii) are consistent with the WEC designs. This will ensure consistent positioning of the in-core instrumentation within the center guide tube. Furthermore, there is adequate insertion depth for the in-core instrumentation for the Framatome design.
5. Material and water around in-core: Consistent with resident fuel design The Framatome center guide tube OD, ID, and minimum wall thickness is the same as the WEC designs. Furthermore, the Framatome center guide tube upper weep hole diameter is the same as the resident designs.

Table 4

Framatome Mechanical Compatibility Evaluation (Fuel Assembly)

Core Interfaces
1. Gaps between assemblies: Positive gap throughout life for all components The Framatome CE16HTP fuel assembly envelopes are consistent with the WEC fuel designs, except for the top spacer grid. Since the mid-spacer grids on the Framatome design have a slightly smaller envelope than the WEC designs, the top spacer grid envelope was increased to prevent excessive bundle lean that could cause issues during upper guide structure installation. Framatome CE16HTP fuel assembly lateral gaps are sufficient throughout the lifetime of the fuel.
2. Positioning: Consistent and compatible with core internals The Framatome UTP and LTP alignment features were assessed in the discrete evaluations discussed above. No issues with positioning of the Framatome CE16HTP fuel within the core.
Control Component Interfaces
3. Full insertion distances This evaluation was performed in the GT / IT mechanical compatibility evaluation demonstrating that a gap between the CEA rod tip and the top of the GT Lower end plug is present at operating conditions.
4. Relative insertion elevations of the top and bottom of the fuel column The fuel stack lengths are the same between the Framatome design and the WEC designs. There is a small difference in fuel stack elevation that has been accepted by APS.
Resident Fuel Assembly Interfaces
5. Spacer grid elevations: Maintains an overlap throughout the design life At Beginning-of-Life (BOL) conditions, the Framatome spacer grids adequately overlap with the CE16STD and CE16NGF structural spacer grids. At End-of-Life (EOL) compared to BOL conditions, the Framatome spacer grids adequately overlap with the CE16STD and CE16NGF structural spacer grids.
6. Miscellaneous component elevations: Similar to the resident fuel design The Framatome and WEC fuel designs have the same LTP elevations. The Framatome UTP flow plate bottom is slightly higher than the CE16STD design and overall fuel assembly height is equivalent. The Framatome flow plate bottom is slightly lower than the CE16NGF design and overall fuel assembly height is slightly shorter. These differences are shown to be acceptable. Framatome CE16HTP fuel assembly engagement with the UGS over the design life has been demonstrated in the growth evaluation. The Framatome growth evaluation demonstrates sufficient fuel rod shoulder gap while accounting for thermal and irradiation effects.
7. Instrumentation insertion distance This assessment was performed in the guide tube / instrument tube mechanical compatibility evaluation. The results of the assessment showed adequate insertion depth for the in-core detectors for the Framatome CE16HTP fuel assembly.
8. Fuel weight and center of gravity: Consistent with resident fuel design The Framatome CE16HTP fuel assembly is slightly heavier than the CE16STD fuel. This is acceptable and would fall within typical production variations. The Framatome CE16HTP fuel assembly is much heavier than the CE16NGF design. This difference will be accounted for in the Palo Verde site handling procedures. The Framatome CE16HTP fuel design fuel stack is slightly higher than the CE16STD design and slightly lower than the CE16NGF design. Since the fuel stack lengths are the same for all of the designs, these differences are insignificant. The Framatome CE16HTP fuel has successfully been handled at the Palo Verde site during the LFA program; therefore experience is established with this fuel at this site.

SNPB RAI-9

The regulatory basis for this RAI is RG 1.203 and Criterion 10 of Appendix A to 10 CFR Part 50.

Provide details of the following analyses that are summarized in the licensee's supplemental letter dated October 18, 2018 (Response to NRC Question 3):

a. Mixed core compatibility evaluations.

b. CE 16 HTP™ Thermal-hydraulic Characterization.

c. Mixed core thermal margin performance specifically explains how the **[[ ]]**.

d. Explain the **[[ ]]**.

Response to SNPB RAI-9

Parts a and b Response

Framatome and Westinghouse performed mixed core compatibility and thermal-hydraulic characterization analyses for mixed core configurations with the competitor's fuel. Section 5.7 of Attachment 10 of the License Amendment Request details each thermal-hydraulic compatibility/characterization items the fuel vendors identified by phenomenon. This response provides supplemental information on the methodologies used, codes applied, and numerical results to support the statements in Section 5.7 of Attachment 10 of the LAR and in the APS response to NRC Acceptance Question #3 (Reference 8) regarding thermal-hydraulic compatibility/characterization of the CE16HTP fuel product.

Core Pressure Drop (Section 5.7.1 of Attachment 10 of the LAR) was evaluated by Framatome by generating pressure drop profiles using the thermal-hydraulic code COBRA-FLX for the following configurations: full core CE16STD, full core CE16NGF, full core CE16HTP, eight CE16HTP lead test assemblies, one batch of CE16HTP assemblies, and two batches of CE16HTP assemblies. A graphical representation of the resultant pressure drop profiles is provided in Figure 3.

Figure 3 is the basis for the statements made in Section 5.7.1 of Attachment 10 of the LAR. It can be seen that the three mixed core configurations considered lie within the bounds of the full core configuration pressure drop profiles. The difference between the CE16STD bundle core average pressure drop and the CE16HTP bundle average core pressure drop is **[[ ]]** and the pressure drop ratio is approximately **[[ ]]**, where the CE16HTP fuel has the larger pressure drop. As the core design transitions from a full core of CE16STD fuel to a full core of CE16HTP fuel the overall core pressure drop will increase.

The difference between the CE16NGF and the CE16HTP bundle average core pressure drop is **[[ ]]** and the pressure drop ratio is **[[ ]]**, where the CE16HTP fuel has the lower pressure drop. As the core transitions from a full core of CE16NGF fuel to a full core of CE16HTP fuel the core pressure drop will decrease.

As indicated by the pressure drop profiles the CE16HTP fuel in transition or full core load is bounded by the CE16NGF and the CE16STD fuel types.

Total Bypass Flow (Section 5.7.2 of Attachment 10 of the LAR) associated with a full core of Framatome CE16HTP fuel was examined to determine if the active heat transfer coolant flow would be adversely impacted. The flows that contribute to the total bypass include guide tubes, vessel upper head, inlet-to-exit nozzle, and core barrel/baffle flow. The change in total bypass flow is determined by examining the changes in bypass flow from the guide tubes and the change in bypass due to non-guide tube paths. Bypass flow for the non-guide tube paths is affected by changes in core pressure drop, while guide tube bypass flow is dependent on both core pressure drop and assembly geometry. The following relationship was utilized to approximate the change in core bypass flow due to the changes in the core pressure drop:

[[

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The PVNGS design limit for total bypass flow is 3%. Relative to the CE16STD fuel, the total bypass flow would **[[** **]]** for a full core of CE16HTP fuel. Relative to the CE16NGF fuel, the total bypass flow would **[[** **]]**. As the CE16HTP fuel assembly pressure drop is between that of CE16STD and CE16NGF fuel, the design limit of 3% will continue to be met.

Crossflow Velocity (Section 5.7.3 of Attachment 10 of the LAR) was analyzed by Framatome using COBRA-FLX and the pressure drop profiles discussed above. Representative neutronics data and a uniform inlet mass flux distribution were used as input. The maximum crossflow velocities were extracted at each axial elevation for the configurations analyzed. The peak velocities were found to occur at approximately **[[** **]]** with the maximum assembly-to-assembly crossflow velocity at **[[** **]]**. The peaks at **[[** **]]** are the result of the inclusion of **[[** **]]**. The peak at the assembly outlet is likely a result of the

modeling of **[[** **]]** and is consistent with prior analyses.

Westinghouse performed a similar analysis using the TORC thermal-hydraulic code and the BOWFOR code. The TORC and BOWFOR codes computed hydraulic lateral velocity and loads on the fuel assemblies. The resultant cumulative forces in the x and y directions were vector summed to determine the total force and direction. The Westinghouse analysis modeled eight CE16HTP assemblies and the maximum assembly-to-assembly crossflow velocity calculated was **[[** **]]**, similar to the Framatome calculated value of **[[** **]]** for the **[[** **]]** transition core. There is no defined thermal-hydraulic analysis acceptance criterion for crossflow velocities. This data, however, is provided for downstream mechanical assessments (e.g., flow induced vibration analysis).

Reactor Coolant System Flow Rate (Section 5.7.4 of Attachment 10 of the LAR) analyses were performed by Framatome to compare relative flow rates for transitioning from a full core of CE16STD fuel, and from transitioning from a full core of CE16NGF. By assuming that core and loop pressure drops are proportional to flow squared, a flat pump head curve (i.e., conservatively not crediting the decreasing in pump head that would occur if the system flow resistance decreased), and ignoring bypass flow paths, the following equation was developed:

[[

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Using COBRA-FLX models it was determined that a **[[** **]]** and a **[[** **]]** was observed. When applied to the current nominal flow rate, the change resulting from the fuel transition will not affect the current Technical Specification minimum loop flow rate that is the basis for the safety analyses.

CEA Drop (Scram) Time (Section 5.7.5 of Attachment 10 of the LAR) analyses were performed by both Framatome and Westinghouse. Framatome outlined the primary characteristics of guide tube design that can impact control rod drop time as:

- Internal diameter of each guide tube region
- Number and diameter of flow holes
- Elevation of each transition region

Framatome previously performed a CEA drop time analysis for eight (8) LTAs that were installed in PVNGS Unit 1 during operating Cycles 15, 16, and 17. Framatome compared the LTA guide tube design with the guide tube design for Framatome fuel assemblies to be supplied for future PVNGS reload batches. Two noteworthy differences were identified.

First, a slight reduction in guide tube dashpot elevation **[[ ]]** could result in a slight reduction in CEA drop time. Second, the revised guide tube design has a longer dashpot transition region, which will have a negligible effect on overall velocity or drop time. Because one of two differences noted was determined to be negligible, and the other resulted in slightly decreased CEA drop times, Framatome concluded that the previous CEA drop time analysis for the LTAs remained bounding for the new design, that is, CEA drop times would remain acceptable relative to the requirements of Technical Specification Surveillance Requirement 3.1.5.5 (see also the response to TH RAI-02).

The Westinghouse analysis of the impact of the Framatome guide tube design and its potential impact on the CEA drop time concluded that the two guide tube designs were similar. Based on assembly pressure drops, the CE16HTP CEA drop time in the mixed core would lie between the CEA drop times associated with a full core of CE16STD and a full core of CE16NGF. CEA Drop times are validated prior to reactor criticality and after each removal of the reactor head to ensure compliance with the plant's maximum allowed CEA drop time per Technical Specification Surveillance Requirement 3.1.5.5.

Fuel Rod Bow (Section 5.7.6 of Attachment 10 of the LAR) analyses were performed by Framatome focusing on rod bow penalty quantification based on the Framatome's XN-75-32(P)(A) methodology, and by Westinghouse focusing on rod bow force/rod fretting. These analyses were an extension of the crossflow velocity analysis described above with both analyses utilizing the RODBOW computer code. As stated in Attachment 10 of the LAR, Framatome determined that no rod bow penalty was required **[[ ]]**.

Similarly, the Westinghouse results indicated the bow force impact on Westinghouse assemblies adjacent to Framatome assemblies was an **[[ ]]** relative to a uniform CE16NGF core maximum net bow force of **[[ ]]**. Though the rod bow force was **[[ ]]**, the results were found to be within the uniform experience base and were judged acceptable by Westinghouse.

Guide Tube Heating (Section 5.7.7 of Attachment 10 of the LAR). Guide tube heating is addressed in the response to TH RAI-03.

<u>Part c Response</u>

An explanation of the CETOP-D "adjustment factors" development and the measures taken to ensure they are conservative is provided in the response to SNPB RAI-07 that describes the two portions of the CETOP-D benchmarking process that ensure the resultant mixed core CETOP-D "adjustment or penalty factors" are conservative. In addition, a discussion of the thermal margin assessment using CETOP-D is provided in the response to SNPB RAI-05.

<u>Part d Response</u>

Section 5.6 of Attachment 10 of the LAR provides details on the processes and components modeled in the development of the DNBR probability distribution function and the validation of the DNBR Specified Acceptable Fuel Design Limit (SAFDL) on a cycle-to-cycle basis. As stated, the methods used to develop the DNBR safety limit for a full core of CE16HTP were

based on CEN-356(V)-P-A, Revision 01-P-A (Reference 4) and WCAP-16500-P-A Supplement 1, Revision 1 (Reference 5).

As summarized in the APS response to NRC Question 3 in Reference 8, in cycles where multiple fuel types including CE16HTP might be limiting, APS is introducing a "branched" or parallel process that contains thermal-hydraulic as well as setpoints components. This branched process requires that the analyses be performed for each identified limiting fuel type, thereby requiring the analyses to be performed multiple times.

The analysis process detailed in Section 5.6 of Attachment 10 of the LAR is summarized here, but now through the lens of a branched cycle. The first step is to identify potential limiting assemblies based on neutronic and DNBR criterion using the process outlined in the response to SNPB RAI-07. For each limiting fuel type identified, a parallel "branch" of analysis is created for all candidates of that type. For example, if the limiting assembly selection identifies both CE16HTP and CE16STD as potentially limiting types, two branches would be initiated. Each branch then **[[ ]]** for all candidate assemblies of that fuel type and the process is repeated for the other identified fuel types.

This is followed by the system parameter Statistical Combination of Uncertainties analysis that is conducted at a fixed set of most sensitive operating conditions for each branch. A candidate-specific VIPRE model is constructed and provided as input to the **[[ ]]** to perturb the system parameters and construct a DNBR probability distribution for each branch. As described in the APS response to NRC Question 3 in Reference 8, this distribution is **[[ ]]**. Continuing with the example from above where both CE16STD and CE16HTP are potentially limiting, the **[[ ]]**.

The branches continue through the Overall Uncertainty Analysis setpoints process, where the effects of the uncertainties in state parameters are incorporated. The setpoints analyst then reviews the results of the branches to determine which BERRs, EPOLs, and other addressable constants are most conservative for installation into the plant's protection (CPCs) and monitoring (COLSS) systems to bound all limiting fuel types.

Figure 3

[[

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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.

Response to SNPB RAI-10

The response to SNPB RAI-1 states that APS will use its normal fuel transition process to transition a unit from one type of fuel to another type of fuel, using full reload batches from a single fuel supplier. The planned transition will involve the eventual replacement of Westinghouse CE16STD fuel with Framatome CE16HTP fuel. Westinghouse CE16NGF fuel will not be used in that unit during that transition. Likewise, PVNGS transitions from CE16STD to CE16NGF involve no Framatome CE16HTP fuel. Thus, all transitions involve only two types of fuel in a reactor core.

For the planned fuel transition, faulted condition analyses for mixed core configurations are performed by both fuel vendors, Framatome and Westinghouse, using their respective methodologies. Each fuel vendor is responsible for evaluating the effects of seismic and LOCA loads on its own fuel type. Westinghouse and Framatome have shared their proprietary information to facilitate the analytical evaluations**.**

The following paragraphs provide further information on each fuel vendor's scope of work.

Framatome

The Framatome seismic and LOCA core row analysis methodology (Reference 32) is based on the essential premise that all assemblies in the core have similar dynamic characteristics, and that the grids' impact behavior is dynamically equivalent to a **[[** **]]** at all times. Under these conditions, the similar but different neighboring grids can be **[[** **]]** grid elements. This method is detailed in Section 5 of Reference 32. For two neighboring heterogeneous assemblies, A and B, the inter-assembly **[[** **]]** given by the following equations: **[[**

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The terms **[[ ]]** assemblies' external stiffness and damping coefficients, respectively. The internal stiffness of each assembly is unchanged. The method is appropriate for the case of the Palo Verde cores because both CE designs and the Framatome design represent assemblies with **[[**

]] physically meaningful.

The analysis of the mixed core rows is then carried out in the same manner as for the Framatome homogenous core rows. The post-processed fuel assembly deflections are subsequently used for the component stress calculations, and the grid node impact loads are used directly for spacer grid margin assessments. The specific mixed core configurations are described in the response to SNPB RAI-29.

<u>Westinghouse</u>

Westinghouse analysis work related to a Palo Verde CE16STD to CE16HTP fuel transition is underway. Westinghouse is using the same methodologies as were used in the NRC approved transition from CE16STD to CE16NGF. APS anticipates that the NRC will audit Westinghouse work products later in 2019.

Westinghouse also serves as APS's Owner's Designee for the purpose of evaluating certain RCS components and reactor vessel internals with respect to compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

SNPB RAI-11

The regulatory basis for this RAI is 10 CFR 50.46(b)(5), "Long term cooling," that states, "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

Please clarify the performance of the Framatome HTP™ fuel design relative to the fuel currently loaded at Palo Verde with respect to the concerns identified in Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," associated with debris blockage following a loss-of-coolant accident (LOCA). The NRC staff seeks to assure that plant modifications during the interim period prior to the plant-specific resolution of all concerns identified in GL 2004-02 adequately consider impacts of post-LOCA debris blockage to avoid unwarranted increases in the potential for inadequate long-term core cooling. Hence, this information request is necessary to assure that the proposed transition to HTP™ fuel adequately accounts for 10 CFR 50.46(b)(5) requirements pending the plant-specific resolution of GL 2004-02 concerns.

Response to SNPB RAI-11

The sump strainers at PVNGS have a maximum hole size of 0.083 inches. The Framatome CE16HTP fuel has a minimum inlet clearance size that is smaller than the sump strainers, between 0.064 and 0.067 inches. This is contrary to the statement in section 6.2.2.2.1.J of the UFSAR:

> *"The holes in the perforated plate of the strainers are 0.083 inches in diameter, which satisfies original C-E requirements on sump screen design (0.09 inches maximum). No flow blockage will occur beyond the screen as all openings are larger than the minimum screen size."*

Based on analysis performed, although the proposed fuel inlet filter minimum restriction is smaller than the sump strainer maximum hole size, blockage will not occur at fiber loadings less than 15 grams per fuel assembly (g/FA) downstream of the strainer because:

- All downstream openings, with the exception of the proposed Framatome CE16HTP fuel inlet filter, have openings that are larger than the minimum sump strainer screen size
- Debris capture using the current basis WCAP-16793 methodology with 15 g/FA fiber limit already accounts for complete debris capture at the fuel inlet
- Because complete debris capture is already evaluated in the WCAP methodology, particle sizes from coatings debris, that could exceed the fuel inlet filter minimum restriction size would only tend to decrease head loss across the debris bed, and
- More complete capture of particulate, with sizes that are larger than the fuel inlet filter minimum restriction, would result in a higher Particulate/Fiber ratio which has been shown to decrease head loss.

The different debris types were considered and concluded that the new fuel type does not impact the current PVNGS fiber loading value of 13.8 g/FA. This is bounded by the 15 g/FA limit specified in WCAP-16793.

The debris types were also evaluated for effects in the heated core region. None of the debris will build up or adhere to the fuel clad.

Based on these conclusions, as the requirements of WCAP-16793 are met, the emergency core cooling systems and containment spray system performance is not adversely impacted by the CE16HTP fuel and will perform their design functions.

SNPB RAI-12

The NRC staff's review of the calculated results in ANP-3639P, "Palo Verde Units 1, 2, and 3 Realistic Large Break LOCA Summary Report," identified certain datapoints that appear to be outliers with respect to the main body of data. Information available in the submittal does not provide a reasonable physical explanation for the predicted behavior in these outlying cases. To provide reasonable assurance that implementation of the Realistic Large-Break LOCA methodology at Palo Verde produces expected results when computing the figures of merit required for comparison against the acceptance criteria in 10 CFR 50.46(b), please

a. Provide a physical explanation for the following outlying predictions:

   - Figure 3-2, "PCT [Peak Cladding Temperature] versus PCT Time Scatter Plot," of ANP-3639 shows that the case that sets the statistical limit for PCT is **[[ ]]**.
   - Figure 3-4, "Maximum Local Oxidation versus PCT Scatter Plot," of ANP-3639 shows that **[[ ]]**

b. Provide the maximum local oxidation for each case shown in Table A-1, "Summary of Key Input and Output Parameters, Part 1," of ANP-3639P.

Response to SNPB RAI-12

The response to this RAI is provided in Enclosure Attachments 8 (non-proprietary) and 12 (Proprietary).

SNPB RAI-13

Section 4.3 of ANP-3640P, "Palo Verde Units 1, 2, and 3 Small Break LOCA [SBLOCA] Summary Report," describes delayed reactor coolant pump trip sensitivity studies. To ensure that these sensitivity studies are sufficient to provide confidence that the most severe postulated conditions consistent with 10 CFR 50.46 have been calculated, please provide the following information:

a. The results of the break sizes that were analyzed for the hot leg and cold leg sensitivity studies in tables similar to those provided in Table 4-1, "Summary of SBLOCA Break Spectrum Transient Results," of ANP-3640P.

b. The results for the limiting break sizes for the cold leg and hot leg cases in a table similar to those provided in Table 4-2, "Sequence of Events for Break Spectrum (seconds)," of ANP-3640P.

c. Discuss the modeling used for loop seal biasing in the hot leg and cold leg sensitivity studies and discuss if it was necessary to **[[** **]]** for the studies.

Response to SNPB RAI-13

The response to this RAI is provided in Enclosure Attachments 9 (non-proprietary) and 13 (Proprietary).

SNPB RAI-14

To assure the conservatism of the SBLOCA analysis used to demonstrate compliance with the limits of 10 CFR 50.46(b), please provide justification that the 5-minute reactor coolant pump trip delay time assumed in ANP-3640P considers the limiting condition with respect to reactor coolant pump operation for the full range of postulated small breaks on both the cold and hot legs. Please address in particular the range of larger breaks in the SBLOCA spectrum. For such breaks, a 5-minute delay time is essentially equivalent to running the reactor coolant pumps throughout the event, which has long been known to result in reduced PCTs. In responding, please identify the most likely range of times over which plant operators will manually trip the reactor coolant pumps and explain why there is confidence that a reactor coolant pump trip in this time range would be bounded by the existing analyses.

Alternatively, please perform additional sensitivity studies that consider reduced reactor coolant pump trip time delays for break sizes 5 inches and larger to ensure satisfaction of the requirement in 10 CFR 50.46(a)(1)(i) that there is assurance that the most severe postulated LOCAs are calculated. Please further provide a basis for considering any revised sensitivity studies as covering the potential range of times when operators would be expected to complete actions to manually trip the reactor coolant pumps.

Response to SNPB RAI-14

Please note that supporting information related to this RAI response and provided by Framatome is included in Enclosure Attachments 9 and 13.

The Framatome Small Break Loss of Coolant Accident (SBLOCA) analysis summarized in the LAR (Reference 9) reported the following limiting results relative to the acceptance criteria of 10 CFR 50.46(b), for a cold leg break spectrum in which the analytical methodology included an assumed Loss of Offsite Power (LOOP) and Reactor Coolant Pump (RCP) trip at the time of reactor trip:

- Peak Cladding Temperature (PCT), 9.10-inch diameter cold leg pump discharge break = 1620°F.
- Maximum Local Oxidation (MLO), 8.80-inch diameter cold leg pump discharge break = 2.96% **[[** **]]**.
- Core Wide Oxidation (CWO), 8.80-inch diameter cold leg pump discharge break = 0.006%.

The Framatome SBLOCA analysis also included a delayed RCP trip study that considered a spectrum of cold leg and hot leg breaks without an assumed LOOP, as described in the response to SNPB RAI-13. This trip study was performed to address the NRC Safety Evaluation for Framatome topical report EMF-2328(P)(A), Revision 0, Supplement 1(P)(A), Revision 0 (Reference 10), and concluded that although PVNGS does not have an automatic RCP trip, 10 CFR 50.46(b)(1-4) criteria would still be met if plant operators tripped all four RCPs after net positive suction head (NPSH) trip criteria were met. Consequently, APS

proposed in the LAR to adopt 5 minutes as a new time critical operator action in the PVNGS Emergency Operating Procedures (EOPs).

Analytical results of the 5-minute RCP trip study demonstrated that, when RCPs are run for that length of time, postulated hot leg breaks could yield higher figures of merit than cold leg breaks, but would remain bounded by the limiting cold leg break spectrum analysis with an assumed LOOP described above. As indicated in Enclosure Attachments 9 (non-proprietary) and 13 (Proprietary), the 5-minute RCP trip delay study **[[**

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APS stated in its response to IRAB RAI-1 (Reference 11), however, that time critical operator action pilot testing was performed to assess how quickly personnel would trip the RCPs following a postulated SBLOCA. Reference 11 notes that pilot testing in the PVNGS control room simulators utilized five different crews of licensed operators, and involved a SBLOCA scenario of sufficient break size to cause a loss of subcooled margin while the RCPs were operating. APS concluded that, on average, pilot testing revealed that operators would trip the RCPs approximately 30 seconds following a loss of subcooled margin.

Framatome reanalyzed the spectrum of cold leg breaks and hot leg breaks on APS's behalf, using the same approach as the 5-minute RCP trip delay study, but with a 30-second trip time that is representative of expected actual operator performance. Analytical results for the shorter RCP trip time were **[[ ]]** less limiting than, the cold leg break spectrum analysis with a LOOP described above. **[[**

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Based on the above, APS concluded that even short periods of continued RCP operation following a postulated PVNGS SBLOCA, such as may occur while operators assess plant conditions and then take action to trip the RCPs following a loss of subcooling, can have a beneficial effect on the figures of merit relative to the acceptance criteria of 10 CFR 50.46(b). **[[**

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SNPB RAI-15

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Response to SNPB RAI-15

The response to this RAI is provided in Enclosure Attachments 9 (non-proprietary) and 13 (Proprietary).

SNPB RAI-16

The NRC staff's safety evaluation for EMF-2328(P)(A), Revision 0, Supplement 1, states that **[[**

]] ANP-3640P does not discuss whether switchover to the containment recirculation sump could occur prior to core quench for some SBLOCA events, or whether this behavior was explicitly modeled. Therefore, to ensure a conservative calculation of the figures of merit required to satisfy the acceptance criteria in 10 CFR 50.46(b),

a. Please identify whether the switchover to sump recirculation could occur prior to core quench for the spectrum of break sizes considered in ANP-3640P. If switchover prior to core quench is not possible, please provide justification.

b. If switchover to sump recirculation could occur prior to core quench for breaks in the size range considered in ANP-3640P, then please either (1) demonstrate that the reported figures of merit remain adequately conservative or (2) reanalyze the affected cases with an explicit modeling of sump recirculation.

Response to SNPB RAI-16

Please note that supporting information related to this RAI response and provided by Framatome is included in Enclosure Attachments 9 and 13.

Parts a and b Response

Switchover of safety injection (SI) pump suction and containment spray (CS) pump suction, from the refueling water tank (RWT) to the containment recirculation sump, will not occur prior to core quench for the spectrum of break sizes considered in ANP-3640P. Therefore, reanalysis of the SBLOCA break spectrum cases presented in ANP-3640P is not required to explicitly model the effects of sump recirculation.

The conclusion that switchover will not occur prior to core quench is based on the following:

- PVNGS Technical Specification (TS) 3.5.5, Surveillance Requirement (SR) 3.5.5.2, requires the RWT borated water volume be greater than or equal to that specified in TS Figure 3.5.5-1, that is, 634,000 gallons or 84.3% RWT level, at reactor coolant system (RCS) temperatures typical of full power operating conditions.
- PVNGS design analyses that support the 634,000-gallon TS value include an allowance of 541,000 gallons to fulfill ECCS design functions, as well as additional amounts to account for the RWT transfer volume between the time of recirculation actuation and isolation of the RWT from the pumps' suction, possible diversion of RWT inventory to boric acid makeup pumps, and instrument uncertainties and analytical margins. The 541,000-gallon value ensures that, following a LOCA, the

containment sump strainers will be covered with water to prevent air entrainment into the suction piping for SI and CS pumps, as well as to provide adequate net positive suction head (NPSH) for the pumps during the recirculation phase post-LOCA.

- The Framatome SBLOCA analyses summarized in ANP-3640P addressed a cold leg break spectrum ranging in size from 1.0-inch diameter to 9.49-inch diameter. For those cases **[[ ]]**, the quench time occurred after the time of PCT. The longest quench time of **[[ ]]** occurred for the **[[ ]]**, and for this case PCT occurred at **[[ ]]**. Inspection of SBLOCA analysis output data revealed that, for this case as well as for all other break sizes in the spectrum, **[[ ]]** before quench was achieved.
- **[[ ]]**, the 541,000-gallon value above needs to be adjusted downward to establish the RWT volume (and mass) that would be available to support SI delivery to the RCS. Assuming a single failure involving the loss of an emergency diesel generator, consistent with the sequences of events presented in ANP-3640P, as well as a CS pump runout flow rate of 5,200 gallons per minute throughout the longest quench time of **[[ ]]**, yields a total CS demand of approximately **[[ ]]**. Consequently, the RWT volume available for SI delivery would be approximately **[[ ]]**. The corresponding mass of RWT inventory available for SI delivery, assuming a density of approximately 61.7 pounds per cubic foot at representative temperature (120°F) and pressure (14.2 psia) conditions, would be **[[ ]]**.

Thus, the RWT inventory available for SI delivery to the RCS **[[ ]]** is approximately 9 times that required **[[ ]]** to achieve core quench following any SBLOCA in the spectrum of break sizes considered in ANP-3640P. Based on this, APS concludes that core quench following a SBLOCA will occur during the injection phase while SI is delivered from the RWT, rather than during the recirculation phase when the pumps will take suction from the containment sump.

<u>SNPB RAI-17</u>

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.

<u>Response to SNPB RAI-17</u>

<u>Part a Response</u>

As noted in the response to SNPB RAI-1, the potential suite of mixed core configurations that could ensue following implementation of the proposed license amendment would be based on the normal PVNGS reload process. That is, a unit that transitions from a full core of one type of fuel to a full core of another type of fuel would utilize full reload batches from a single fuel supplier. This is how PVNGS is transitioning from CE16STD to CE16NGF, and how PVNGS will transition from CE16STD to CE16HTP.

PVNGS fuel management practices typically result in reactor cores comprised of approximately 100 fresh fuel assemblies, approximately 100 once-burned fuel assemblies, and approximately 41 twice-burned assemblies, with a transition between two types of fuel completed over approximately three years during consecutive refueling outages in the selected unit.

When a PVNGS unit transitions from a full core of one fuel type to a full core of another fuel type, each fuel vendor’s large break and small break LOCA analyses consider bounding core configurations for the transition that address the impacts of co-resident fuel types on predicted LOCA figures of merit (e.g., peak cladding temperature). Bounding core configurations are determined by each responsible fuel vendor and take into consideration the PVNGS practice of utilizing full reload batches of fuel from a single vendor, differences in fuel assembly mechanical design and hydraulic characteristics, and the methodologies employed in each vendor’s NRC-approved LOCA Evaluation Models (EMs).

When a PVNGS reactor core transitions from one fuel type to another fuel type, the potential effects of co-resident fuel assemblies on LOCA analyses are evaluated by each responsible fuel vendor to ensure that calculated figures of merit are bounding for all anticipated core configurations throughout the transition. Co-resident fuel effects typically arise from differences in fuel assembly mechanical design, including but not limited to fuel

pellet and fuel rod cladding radial dimensions; guide tube and instrument tube radial dimensions; and elevations and pressure loss coefficients associated with upper and lower end fittings, spacer grids, and mixing vane grids. These characteristics can affect, for example, LOCA analysis calculated values for engineering hoop stress across fuel rod cladding and consequently cladding deformation predictions; the core cross-sectional area for coolant flow; core flow redistribution due to hydraulically dissimilar fuel assemblies; heat transfer coefficients between fuel rods and the coolant during core uncovery; reflood rates; and transient mixture levels in the core region.

The effects of co-resident fuel types are evaluated regardless of the number of fuel vendors involved in a transition. For example:

- The transition from CE16STD to CE16NGF involves only a single fuel vendor, Westinghouse. The APS LAR for the transition to CE16NGF, submitted to NRC in July 2016 (Reference 6) and subsequent approved with an NRC Safety Evaluation (Reference 7), addressed the effects of the two co-resident fuel types on both large break and small break LOCA analyses. Section 8.4.4 of Attachment 7 of the LAR summarizes the Westinghouse analyses as follows:

  > Multiple core configurations were examined and bound the likely transition core scenarios and address core loading differences that are expected in the coming PVNGS cycles of operation. The transition mixed core ECCS performance assessment determined that the results were bounded by the results of the full core CE16NGF implementation analysis. The evaluation bounds the likely transition core scenarios.

- The transition from CE16STD to CE16HTP involves two fuel vendors, Westinghouse and Framatome. To ensure that both vendors can effectively assess the effects of co-resident fuel during the transition, Framatome and Westinghouse released proprietary fuel mechanical design and hydraulic performance data to each other. The Framatome large break and small break LOCA analyses described in the CE16HTP LAR (Reference 9) thus account for anticipated core configurations during the transition from CE16STD to CE16HTP. The Westinghouse large break and small break LOCA analyses **[[**

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As noted above, the methodologies employed in the vendors' NRC-approved LOCA EMs can also be a factor in establishing how bounding transition core configurations are analyzed. For example, **[[**

]] of fuel mechanical design parameters to assess the impact of co-resident Framatome CE16HTP fuel on Westinghouse small break LOCA analysis results. **[[**

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For 10 CFR 50.46 reporting purposes, the transition involving Westinghouse and Framatome fuel types will require four licensing basis analyses of record – two large break LOCA analyses and two small break LOCA analyses – which upon NRC approval will serve as baselines against which future changes will be measured in accordance with 10 CFR 50.46(a)(3).

Part b Response

As indicated in the response to Part a of this RAI, the large break and small break LOCA analyses prepared by APS's fuel vendors for a transition from one fuel type to another would bound anticipated mixed core configurations throughout that transition. It is unlikely that explicit reanalysis of a particular mixed core configuration would be required over the three year period that a fuel transition would take place.

As a part of the normal APS reload process, each vendor's LOCA analyses are explicitly evaluated each cycle to verify applicability. If applicability is not verified, then appropriate reanalysis is performed. This verification process includes mandatory checks of numerous analytical inputs (e.g., Technical Specification requirements), to affirm that the analyses will remain bounding for the anticipated plant configuration. Some LOCA analytical inputs (e.g., containment heat sinks) have been conservatively biased in both Westinghouse and Framatome LOCA analyses at APS's request, so that minor changes in plant configuration (e.g., removal of coatings from metal components in containment) would not invalidate the LOCA analyses.

SNPB RAI-18

Several figures included in ANP-3640P that plot key parameters predicted for the limiting SBLOCA event display behavior that may appear non-physical. To assure that the predicted figures of merit for this event are calculated correctly such that reasonable assurance exists that the acceptance criteria in 10 CFR 50.46(b) have been satisfied, please provide justification that the following calculated behavior is reasonable:

a. Prior to 200 seconds into the event, the steam generator total mass in Figure 4-14, "Steam Generator Total Mass – 9.10 Inch Break," of ANP-3640P begins to vary as a function of time, despite Figures 4-11 (main feedwater flow), 4-12 (steam generator safety valve mass flow) and 4-13 (auxiliary feedwater flow) showing no appreciable changes in the mass flow into or out of the steam generators.

b. After about 200 seconds into the event, the hot assembly mixture level in Figure 4-22, "Hot Assembly Mixture Level – 9.10 inch Break," of ANP-3640P appears to take several different, approximately constant, values across several different periods, most prominently between approximately 355 and 620 seconds; whereas it is not obvious why the hot assembly mixture level should remain approximately constant during this period at a number of distinct values, all of which are below the top of active fuel.

Response to SNPB RAI-18

The response to this RAI is provided in Enclosure Attachments 9 (non-proprietary) and 13 (Proprietary).

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. **[[**

]] 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. **[[** **]]**

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

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

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

Response to SNPB RAI-19

The response to this RAI is provided in Enclosure Attachments 9 (non-proprietary) and 13 (Proprietary).

SNPB RAI-20

Please clarify and justify the modeling of non-Framatome fuel in the large-break LOCA analysis in ANP-3639P.

In particular, ANP-3639P states that

> In addition to the Framatome HTP™ fuel, the hydraulic characteristics of other fuel types that could be present in the core were considered. **[[**
>
> **]]**

However, Limitation 4.3 associated with EMF-2103P-A states that the methodology is applicable to fuel with M5® cladding, and that application of the evaluation model to fuel with other types of cladding has not been reviewed. Furthermore, it is not clear how the fuel pellet and cladding thermal-mechanical behavior is modeled for non-Framatome fuel (1) as a function of burnup to support initialization of the LOCA calculation and (2) during the LOCA transient calculation.

Clarification and justification regarding the modeling of non-Framatome fuel in the large-break LOCA analysis is necessary to ensure that figures of merit are calculated in a representative or conservative manner in order to satisfy the acceptance criteria in 10 CFR 50.46(b) during operating cycles with multiple fuel types present in the reactor core.

Response to SNPB RAI-20

The response to this RAI is provided in Enclosure Attachments 8 (non-proprietary) and 12 (Proprietary).

<u>SNPB RAI-21</u>

Proposed TS SL 2.1.1.2 would specify separate fuel centerline melt limits for "Westinghouse supplied fuel" and "Framatome supplied fuel." However, it is not clear that the proposed terminology is sufficiently precise, since certain types of fuel supplied by Westinghouse or Framatome may be incompatible with the proposed limits. To assure satisfaction of 10 CFR 50.36(c)(1), please clarify whether additional description is necessary in proposed TS SL 2.1.1.2 (e.g., fuel design, applicable topical report methodology) to avoid unsupportable broad interpretations of the proposed wording:

a. Regarding proposed TS SL 2.1.1.2.1, please clarify any applicability criteria (e.g., burnable absorber material) that must be satisfied to apply the fuel centerline melt methodology specified in CENPD-382-P-A to Westinghouse-supplied fuel.

b. Regarding proposed TS SL 2.1.1.2.2, please identify any applicability criteria that must be satisfied to apply the proposed fuel centerline melt limit to Framatome-supplied fuel. Please further identify whether the wording of proposed TS SL 2.1.1.2.2 should be modified to reflect the source of the methodology, such that the limits of applicability are unequivocally defined within the safety limit.

<u>Response to SNPB RAI-21</u>

<u>Part a Response</u>

The proposed Technical Specification 4.2.1.a refers to approved fuel designs. Technical Specification 2.1.1.2 addresses the approved and proposed fuel designs. CENPD-382-P-A only applies to Westinghouse fuel with erbium as a burnable poison. For clarity, the proposed wording for Technical Specification 2.1.1.2 is being rewritten as indicated to clearly define the requirements for each approved and proposed fuel design (see Enclosure Attachment 2 and Enclosure Attachment 3).

With regard to peak centerline temperature, there are no applicability criteria that must be satisfied for approved Westinghouse fuel designs beyond the presence or absence of erbium as a burnable poison, which is accounted for in the proposed wording. The proposed Technical Specification 4.2.1.a, per the response to SNPB RAI-22, limits APS to using approved fuel designs. The proposed Technical Specification 2.1.1.2.1 and 2.1.1.2.2 wording is appropriate for the Westinghouse fuel currently approved for APS. APS adoption of any proposed new Westinghouse approved fuel design would include a review per 10 CFR 50.59 as a minimum. The 10 CFR 50.59 review includes determination as to whether a Technical Specification change is required. As such, if the existing Technical Specification 2.1.1.2 limit did not apply to the proposed new fuel, a license amendment request to modify Technical Specification 2.1.1.2 would be required to be submitted.

<u>Part b Response</u>

There are no applicability criteria beyond those discussed in the proposed Technical Specification and its associated Bases (see Enclosure Attachments 2, 3, and 4) that must be satisfied for Framatome fuel. The proposed Technical Specification 4.2.1.a, per the

response to SNPB RAI-22, limits APS to using approved fuel designs. The proposed Technical Specification 2.1.1.2.3 wording for Framatome fuel is appropriate for the Framatome fuel for which approval is being requested. If the proposed Technical Specification 2.1.1.2 limit did not apply to a future fuel design, a license amendment request to modify Technical Specification 2.1.1.2 would be required to be submitted.

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.

Response to SNPB RAI-22

Part a Response

APS has updated the proposed Technical Specification 4.2.1 wording to clarify that the use of the term "zirconium-alloy clad" refers to cladding that has been approved by the NRC. Additional clarification has been added to explicitly address whether a coated clad falls under this definition. A corresponding change to Technical Specification 5.6.5 is being proposed to require identification of the cladding material in use in each reactor in the unit-specific Core Operating Limits Reports (see Enclosure Attachment 2 and Enclosure Attachment 3).

Part b Response

The revised markup corrects the editorial error on marked up TS page 4.0-1.

SNPB RAI-23

Please clarify the intended meaning of proposed TS 4.2.1 with respect to the materials allowed for lead test assemblies and discuss whether the intended meaning is consistent with a plain-language, literal interpretation. In particular, proposed TS 4.2.1 states that

> The reactor shall contain 241 fuel assemblies. Each assembly shall consist of a matrix of zirconium-alloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material.

The proposed change to TS 4.2.1 would further delete an exception to allow lead test assemblies to use cladding types other than those described in TS 4.2.1. Consequently, it appears that proposed TS 4.2.1 would require lead test assemblies (i.e., which number among the 241 fuel assemblies in the reactor) to have the properties described in the passage quoted above. Clarification of proposed TS 4.2.1 is necessary for the NRC staff to confirm that reactor design features have been adequately specified in TSs, in satisfaction of 10 CFR 50.36(c)(4).

Response to SNPB RAI-23

APS has updated the proposed Technical Specification 4.2.1 wording to clearly identify the requirements applicable to lead test assemblies (see Enclosure Attachment 2 and Enclosure Attachment 3). Refer to the response to SNPB RAI-22 for the details of the proposed Specification 4.2.1 wording.

SNPB RAI-24

The proposed license amendments would add a suite of Framatome methodologies to Palo Verde TS 5.6.5 (Core Operating Limits Report), without deleting corresponding Westinghouse methodologies. To assure that the administrative controls in proposed Palo Verde TS 5.6.5 are sufficient to support operation of the facility in a safe manner in accordance with 10 CFR 50.36(c)(5), please provide the following information:

a. For each analytical methodology in TS 5.6.5, please identify whether the methodology is restricted in its application to fuel assemblies supplied by a particular vendor, or whether it would be applicable to both Framatome and Westinghouse fuel designs.

b. Please identify and provide justification for any instances where topical report methodologies supplied by one fuel vendor would be used to generate core operating limits for fuel supplied by a different vendor.

Response to SNPB RAI-24

Part a Response

As shown in the response to SNPB RAI-22, APS has updated the proposed Technical Specification 5.6.5 wording to require the unit-specific Core Operating Limits Reports (COLRs) to identify which topical report methodologies apply to which fuel type. Additionally, as discussed in the response SNPB RAI-22, the specific fuel type and cladding material present in the reactor core will be identified in each unit's COLR.

Part b Response

The justification for instances where topical report methodologies supplied by one fuel vendor would be used to generate core operating limits for fuel supplied by a different fuel vendor are as described in the License Amendment Request (Reference 9), as addressed by the responses to the RAIs. Upon LAR approval, any future changes will be done in accordance with 10 CFR 50.90 or 10 CFR 50.59, as appropriate.

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.

Response to SNPB RAI-25

Part a Response

As discussed in the 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

As APS is no longer requesting authorization for mixed fresh fuel core designs with this license amendment request, a response is not required.

Part c Response

While the implications of mixed fresh fuel in the areas of core design, thermal-hydraulics, transient analysis, and COLSS/CPC setpoints were addressed during the June 2019 regulatory audit, further justification would be needed in areas such as LOCA (10 CFR 50.34 and 10 CFR 50.46) and fuel seismic response (10 CFR 50.34, 10 CFR Part 50 Appendix A General Design Criterion 2, and 10 CFR Part 100 Appendix A Section VI) to support the use of mixed fresh fuel in future core designs. Implementation of mixed fresh fuel would require a separate prior NRC review and approval pursuant to 10 CFR 50.90, perhaps on an emergency or exigent basis, should there be a security of supply issue arise that would drive the need for the use of mixed fresh fuel in a future PVNGS unit core design.

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.

Response to SNPB RAI-26

Amendment 205 to the Palo Verde Nuclear Generating Station Renewed Operating License added the following License Condition to Appendix D of Facility Operating License Nos. NPF-41, NPF-51, and NPF-74.

> "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 predictions of FATES3B at high burnup for Westinghouse Next Generation Fuel.
>
> To ensure the adequacy of this RFO curve penalty, as part of its normal reload process for each cycle that analysis using FATES3B is credited, APS shall verify that the FATES3B analysis is conservative with respect to an applicable confirmatory analysis using an acceptable fuel performance methodology that explicitly accounts for the effects of 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."

The License Condition in question was added to the PVNGS Renewed Operating License as part of the NRC approval for use of Westinghouse Next Generation Fuel. The changes that Palo Verde proposed in this LAR are not related to the licensing of Westinghouse fuel.

However, to address the issue identified in this RAI, APS will adopt a Regulatory Commitment as shown in Enclosure Attachment 1.

SNPB RAI-27

In order for the NRC staff to confirm the licensee's determination that the existing evaluation of long-term core cooling remains applicable to Palo Verde and continues to satisfy 10 CFR 50.46(b)(5), please either (1) confirm that the values in Tables 8-13 and 8-15 of Attachment 8 to the LAR dated July 6, 2018, to implement Westinghouse NGF dated July 1, 2016, remain valid, or (2) identify any changes and justify that the revised values do not adversely affect the calculated results.

Response to SNPB RAI-27

As part of the Loss of Coolant Accident (LOCA) work performed to support the utilization of Framatome CE16HTP fuel in Palo Verde cores, the Long-Term Core Cooling (LTCC) analysis that were presented to the NRC in the Westinghouse CE16NGF license amendment request (Reference 6) and Approved by the NRC staff in the Safety Evaluation dated January 23, 2018 (Reference 7) were evaluated by Framatome for applicability to CE16HTP fuel.

Table 8-13 of Attachment 7 of Reference 6 was reviewed by Framatome and it was determined that the only parameters that could be affected by the fuel are the RCS parameters: RCS water volume and maximum RCS boron concentration. The RCS boron concentration is a function of the fuel cycle design and will vary even when only considering one fuel type. The Palo Verde reload process contains a verification that this maximum boron concentration bounding value is preserved on a cycle-specific basis. Therefore, the only impact to be addressed for Framatome CE16HTP fuel is the RCS water volume which is a function of the fuel geometry. The differences between Westinghouse CE16STD and Framatome CE16HTP fuel assembly components is negligible relative to the core volume and therefore can be ignored for this evaluation. Since those parameters won't change with a change in fuel design and the core volume at steady-state conditions is unchanged relative to CE16STD, the core mixing volume will be unchanged for CE16HTP fuel relative to CE16STD fuel, it can be concluded that the Boric Acid Precipitation analyses performed for the CE16STD fuel (as presented in Reference 6) can be applied to Framatome CE16HTP fuel in either a mixed core or full core capacity.

Table 8-15 of Attachment 7 of Reference 6 was reviewed by Framatome and it was determined that the items in this table are not affected by the implementation of Framatome CE16HTP fuel. It was concluded that the results presented in the license amendment request for CE16STD fuel are applicable when Framatome CE16HTP fuel is inserted into the core in either a mixed core or full core capacity.

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.

Response to SNPB RAI-28

Parts a and b Response

As noted in the response to SNPB RAI-10, Westinghouse analysis work related to a Palo Verde CE16STD to CE16HTP fuel transition is underway. Westinghouse is using the same methodologies as were used in the NRC approved transition from CE16STD to CE16NGF. APS anticipates that NRC will audit Westinghouse work products later in 2019.

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.

Response to SNPB RAI-29

Parts a and b Response

The Framatome seismic and LOCA analysis for PVNGS involved a number of representative mixed core configurations as follows:

1. A core comprised of Westinghouse CE16STD fuel assemblies with eight co-resident Framatome CE16HTP fuel assemblies (e.g., a lead test assembly configuration). In this analysis, the Framatome CE16HTP assemblies were placed **[[**

   ]]. These configurations are shown in Figures 4 and 5.

2. **[[**

]] are shown in Figures 6 and 7.

3. **[[** **]]** are shown in Figures 8 and 9.

The mixed core configurations were analyzed for all cases of frequency shifted time histories **[[** **]]**. For Framatome CE16HTP fuel, all mixed core scenarios analyzed to this point are **[[** **]]**. The maximum CE16HTP grid impact load for all mixed core configurations is **[[** **]]**.

As noted in the response to SNPB RAI-10, Westinghouse analysis work related to a Palo Verde CE16STD to CE16HTP fuel transition is underway. APS anticipates that NRC will audit Westinghouse work products later in 2019.

<u>Part c Response</u>

The response to TH RAI-01 provides an illustrative comparison of the three reactor fuel designs (CE16STD, CE16NGF, and CE16HTP). The CE16NGF fuel assemblies have IFM grids **[[** **]]** Axial offsets between spacer grids can vary due to differences in fuel assembly design, as well as differences in operating conditions (e.g., temperature) and irradiated state (e.g., BOL vs. EOL). **[[**

]]. For the EOL condition several cases were considered: **[[**

]]

All of the EOL cases **[[**

]].

Therefore, Framatome assessed the minimum axial grid overlap for both the BOL and various EOL conditions **[[**

]].

As noted in the response to SNPB RAI-10, Westinghouse analysis work related to a Palo Verde CE16STD to CE16HTP fuel transition is underway. APS anticipates that NRC will audit Westinghouse work products later in 2019.

Figure 4

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Figure 5

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Figure 6

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Figure 7

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Figure 8

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Figure 9

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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.

Response to SNPB RAI-30

Part a and b Response

The Framatome seismic and LOCA analysis methodology (Reference 32) accounts for the irradiated condition of the fuel assembly, for both the assembly stiffness and grid stiffness, damping, and strength. The process is as follows:

1. For the effects of irradiation on the fuel assembly stiffness, Framatome conducts dynamic tests (free vibration, baffle impact, and vertical drop test) on fuel assemblies equipped with spacer grids conditioned for EOL. The spacer grid conditioning consists of **[[**

   ]].

2. For spacer grid dynamic characterization (stiffness, damping, and buckling strength), Framatome conducts impact tests on EOL conditioned grids. The test results are **[[**

   ]].

3. With the external grid stiffness and damping, it possible to **[[**

   ]].

4. For mixed cores, the grid external stiffness and damping are used to **[[**
   ]] impact elements, as discussed in the response to SNPB RAI-10. This method of **[[**

]]. The Framatome CE16HTP and the Westinghouse CE16STD and CE16NGF grids are in this category.

5. After the dynamic transient analysis is complete, the spacer grid margins are calculated directly from the post-processed impact loads.

In conclusion, the Framatome analysis for mixed cores followed the approved process described in (Reference 32).

As noted in the response to SNPB RAI-10, Westinghouse analysis work related to a Palo Verde CE16STD to CE16HTP fuel transition is underway. APS anticipates that NRC will audit Westinghouse work products later in 2019.

SRXB RAI-1

The regulatory basis for this RAI is 10 CFR 50.36 and Criterion 10 of Appendix A to 10 CFR Part 50.

For multiple non-LOCA transients, the LAR dated July 6, 2018, states that "The transient Linear Heat Rate (LHR) will not exceed 21.0 kW/ft [kilowatt per foot]. Therefore, the fuel centerline melt temperature will not be exceeded."

However, Table 2-3, "CFM Rod Local LHGR Limits," of Attachment 10, of the LAR indicates that 21.0 kW/ft will not preclude centerline melt for **[[**

]]. Provide additional information to demonstrate that fuel centerline melt temperature will not be exceeded for Updated Final Safety Analysis Report (UFSAR) events 15.1.3, 15.1.4, 15.1.5, 15.1.6, and 15.2.3.

Response to SRXB RAI-1

Table 2-3 of Attachment 10 of the LAR (Reference 9) indicates that the limiting UO_2 fuel rod at low burnups has a power-to-centerline melt that is always above 21 kW/ft. Radial fall-off credit will be employed to ensure that the "effective" power-to-centerline melt for the gadolinia fuel rods and the high burnup UO_2 fuel rods is also above 21 kW/ft. Since the transient LHR does not exceed 21 kW/ft for Updated Final Safety Analysis Report (UFSAR) events 15.1.3, 15.1.4, 15.1.5, 15.1.6, and 15.2.3, the fuel centerline melt temperature will not be exceeded.

Power-to-centerline melt (PTM) is shown as a function of rod average burnup and gadolinia content in Table 2-3 of Attachment 10 of the LAR. The power which will result in just reaching fuel centerline melt decreases as rod average burnup increases and the gadolinia content increases. The application of this behavior in a safety analysis will be to design the core to ensure that the peaking factors in the gadolinia fuel rod and the high burnup UO_2 fuel rod are proportionately lower as required to ensure that the "effective" power-to-centerline melt limit is not less than the local power density trip setpoint of 21 kW/ft. The use of gadolinia as a burnable absorber requires that different levels of U-235 enrichments be used in the gadolinia bearing fuel rods. As the gadolinia loading is increased, the U-235 enrichments will decrease such that the performance of the gadolinia bearing fuel rod will be similar to a UO_2 fuel rod.

For example, the power-to-centerline melt in a gadolinia fuel rod at 4 wt% gadolinia does not drop below 21 kw/ft until burnups exceed 20 GWd/mtU. Such a fuel rod will be in its second cycle of operation and will be significantly lower in power peaking than the fresh fuel. Thus, such a fuel rod cannot reach the 21 kW/ft centerline melt limit.

The radial fall-off normalized to the peak at each time-point in the cycle as a function of burnup will be performed for each core design. A cycle-specific radial fall-off check is already part of the cycle-specific reload physics design assessment process to ensure that "effective" power-to-centerline melt margin exists for a given core design.

The following is a demonstration that the "effective" power-to-centerline melt is > 21 kW/ft for the N+2 full core of Framatome CE16HTP fuel. The N+2 core design is described in Attachment 10 of the LAR.

Gadolinia Fuel Rod Evaluation

The LHR Ratio is defined as the PTM of Table 2-3 of Attachment 10 of the LAR divided by 21 kW/ft. The LHR Ratio is tabulated in Table 5 as a function of gadolinia wt% and rod average burnup.

The cycle-specific radial fall-off curve for each gadolinia wt% fuel rod type is then compared to the gadolinia wt% LHR Ratio in Figures 10 through 13. It is shown in Figures 10 through 13 that the gadolinia bearing fuel rods for the N+2 core design do not reach the high powers as indicated by the margin between the radial fall-off curve and the LHR Ratio.

UO_2 Fuel Rod Evaluation

The LHR Ratio is defined as the PTM of Table 2-3 of Attachment 10 of the LAR divided by 21 kW/ft. The LHR Ratio is tabulated in Table 6 for UO_2 fuel rods as a function of burnup.

The cycle-specific radial fall-off curve for all fuel rod types is then compared to the UO_2 specific LHR Ratio in Figure 14. It is shown in Figure 14 that the UO_2 fuel rods for the N+2 core design do not reach the high powers as indicated by the margin between the radial fall-off curve and the LHR Ratio. Since the UO_2 fuel rods cannot reach such high powers, "effective" power-to-centerline melt margin exists for the N+2 core design.

"Effective" Power-to-Centerline Melt Evaluation

For the N+2 core design, the following is an additional demonstration that the "effective" power-to-centerline melt is greater than the 21 kW/ft limit for all fuel rod types.

The "effective" power-to-centerline melt is defined to be the PTM of Table 2-3 of Attachment 10 of the LAR divided by the radial fall-off relative power. Figures 10 through 13 (gadolinia rods) and Figure 14 (all fuel rods) present the radial fall-off relative power as a function of fuel rod type at each burnup point for the N+2 core design.

Figure 15 shows that the gadolinia and UO_2 fuel rod "effective" power-to-centerline melt is in excess of the minimum local power density trip setpoint of 21 kW/ft.

Table 5

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Table 6

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Figure 10

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Figure 11

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Figure 12

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Figure 13

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Figure 14

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Figure 15

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SRXB RAI-2

The regulatory basis for this RAI is 10 CFR 50.36 and Criterion 10 of Appendix A to 10 CFR Part 50.

For multiple transients, the LAR dated July 6, 2018, states that "The maximum LHGR [linear heat generation rate] will remain below the value that causes peak centerline melt temperature (TS 2.1.1.2 limit)."

However, the TS 2.1.1.2 limit is provided in terms of peak fuel centerline temperature. Further, there are separate limits for Westinghouse Supplied Fuel and for Framatome Supplied Fuel, which vary as a function of burnup as plotted below.

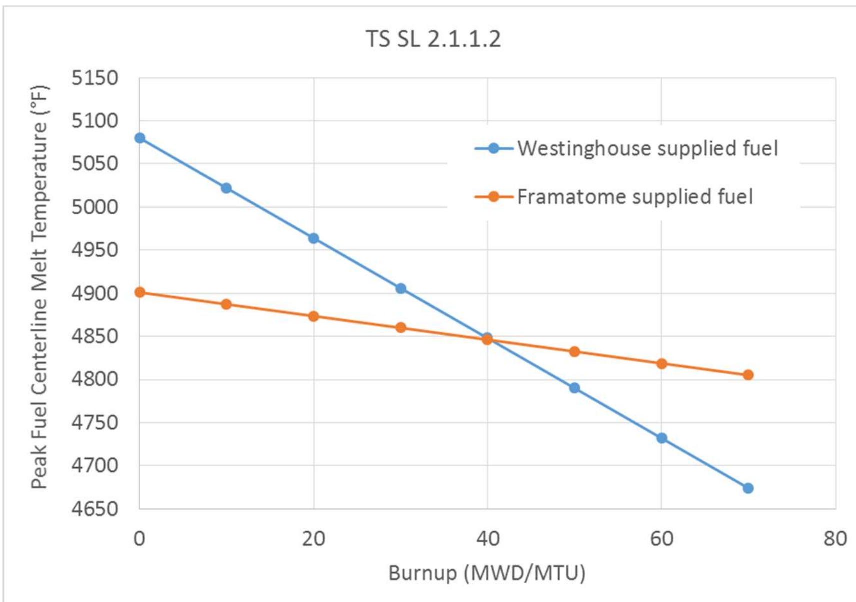


Table 2-3 of Attachment 10 of the LAR dated July 6, 2018, indicates that **[[ ]]**. Additional information is needed to demonstrate how LHGR values are controlled to ensure that peak centerline melt is precluded and that variations in fuel type and **[[ ]]** are accounted for in the calculations for UFSAR events 15.4.2 and 15.4.3.

Response to SRXB RAI-2

Table 2-3 of Attachment 10 of the LAR (Reference 9) indicates that the limiting UO_2 fuel rod at low burnups has a power-to-centerline melt that is always above 21 kW/ft. Radial fall-off credit will be employed to ensure that the "effective" power-to-centerline melt for the gadolinia fuel rods and the high burnup UO_2 fuel rods is also above 21 kW/ft. Since the transient Linear Heat Rate (LHR) does not exceed 21 kW/ft for Updated Final Safety

Analysis Report (UFSAR) events 15.4.2 and 15.4.3, the fuel centerline melt temperature will not be exceeded.

To determine the minimum power-to-centerline melt, each specific fuel rod type resident in the Palo Verde units has been analyzed with the appropriate NRC approved fuel performance code. The COPERNIC code was employed to predict the linear heat rates at which the onset of fuel centerline melting occurs for the Framatome CE16HTP fuel rods. FATES3B calculations have been performed for Westinghouse CE16NGF UO_2, CE16NGF ZrB_2, CE16STD UO_2, and CE16STD erbia fuel rods.

21 kW/ft is the steady state limit that provides protection against exceeding the peak fuel centerline melt temperature. For each "hot rod" fuel rod type at each burnup step, the minimum power-to-centerline melt is calculated. The minimum power-to-centerline melt is less than 21 kW/ft at high burnups and high burnable absorber concentrations but the normalized power-to-centerline melt is maintained above the bounding radial fall-off curve. As the gadolinia loading is increased, the U-235 enrichments will decrease such that the performance of the gadolinia bearing fuel rod will be similar to a UO_2 fuel rod. At high rod average burnups, the burned fuel can no longer achieve the high powers achieved by fresher fuel. Thus, the higher power rods in the adjacent lower burnup assemblies are the limiting rods. Since the normalized power-to-centerline melt always exceeds the bounding value of the radial fall-off curve, the power-to-centerline melt of the lower burnup rods is limiting. Consequently, the "effective" power-to-centerline melt of the limiting fuel rod is always greater than 21 kW/ft.

The radial fall-off normalized to the peak power at each time-point in the cycle as a function of burnup is used to provide a conservative power profile for the pseudo hot pin model. Each fuel rod type has a specific radial fall-off curve that ensures the "effective" power-to-centerline melt of the limiting fuel rod is always greater than 21 kW/ft. A cycle-specific radial fall-off curve validation for each fuel rod type already is incorporated into the cycle-specific reload physics design assessment process to ensure that "effective" power-to-centerline melt margin exists for a given core design. The philosophy behind this method is that the fuel rod cycle specific radial fall-off curves can be compared to the limiting curves to show that the "effective" power-to-centerline melt analysis is applicable and conservative for a particular reload.

<u>SRXB RAI-3</u>

The regulatory basis for this RAI is 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

In Section 6.2, "DNB Propagation," of Attachment 10 of the LAR dated July 6, 2018, 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 **[[** **]]** therefore **[[** **]]**. **[[** **]]**. The LAR includes Figures 6-1 and 6-2, which provide **[[** **]]** and **[[** **]]** 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.

a. Please provide data and an explanation that shows strain is bounded for Framatome CE16 HTP™ fuel.

b. If sufficient dynamic strain data under representative cladding temperature ramp rate conditions does not exist to demonstrate that existing DNB propagation methods may be applied to Framatome HTP™ fuel, then please provide an alternate means of addressing DNB propagation for Framatome HTP™ fuel (e.g., consideration favorable pressure gradient across the cladding wall for the relevant set of events and exposure histories).

<u>Response to SRXB RAI-3</u>

<u>Parts a and b Response</u>

During certain Non-LOCA UFSAR Chapter 15 events, the event DNBR falls below the DNBR SAFDL. When this occurs, heat transfer from the fuel rods in DNB decreases and circumferential fuel rod clad strain increases, which can then lead to ballooning of the fuel rod cladding if the rod internal pressure is greater than the RCS pressure. Deformation (ballooning) of these fuel rods can lead to channel blockage which would then reduce the coolant flow past adjacent rods leading to a reduction in the DNBR in an adjacent rod. Should an adjacent rod then experience DNB, similar ballooning due to high internal pressure could result in DNB propagation continuing to additional fuel rods. APS currently uses NRC-approved ABB-CE Topical Report CEN-372-P-A (Reference 12) methodology to assess DNB propagation for Westinghouse fuel.

During the discussion of this RAI at the June 2019 on-site audit for the LAR, the NRC Staff requested clarification as to whether APS is requesting NRC approval to use existing methodology for evaluating Framatome fuel with M5 cladding or a time in DNB criterion to ensure that DNB propagation of Framatome fuel with M5 cladding will not occur. APS is requesting NRC approval to use a time in DNB of less than 5 seconds as a criterion to ensure that DNB propagation of Framatome fuel with M5 cladding will not occur. The

acceptability of this criterion is addressed in this RAI response. The use of this criterion negates the need for a comparison between Zr-4 and M5 strain values and related responses to items (a) and (b).

The PVNGS Reactor Protection System (RPS) design with its digital CPC system features a DNB trip that ensures that no fuel rod will be in DNB for a significant period of time. For example, as described in Section 6.6 of Attachment 10 of the LAR, the analysis of an Anticipated Operational Occurrence (AOO) from SAFDL with Framatome fuel was in DNB for 3.8 seconds.

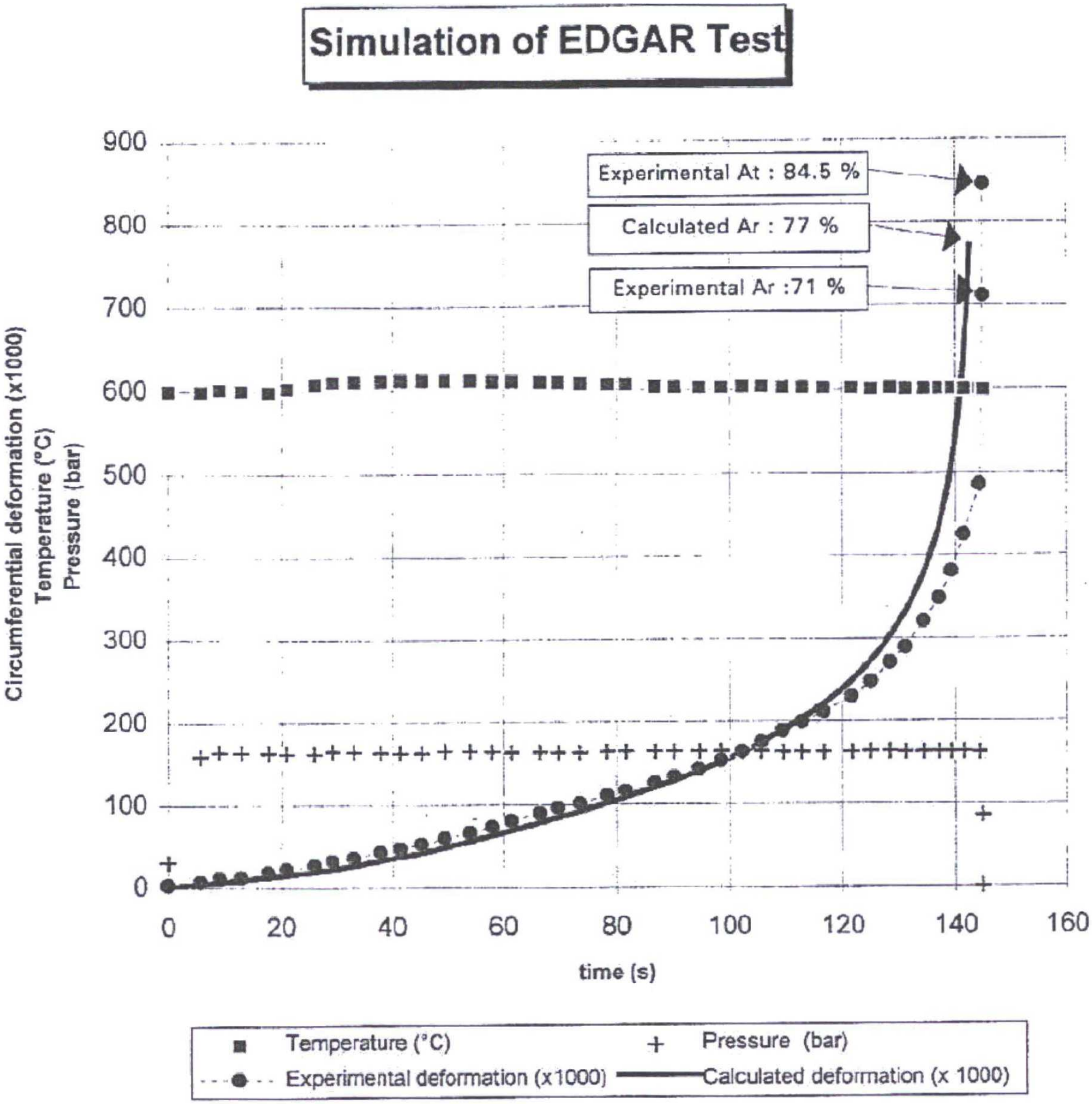
ABB-CE Topical Report CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure" (Reference 12), provides technical justification for rod internal pressure exceeding RCS pressure, and performed analyses for 14x14 and 16x16 fuel designs to assess the likelihood and potential consequences of DNB propagation. Per the topical report, **[[**
]] resulted in insignificant Zircaloy-4 clad strain that would not cause clad ballooning sufficient to initiate DNB in adjacent fuel rods.

Comparisons of circumferential fuel rod clad strain buildup in both Zircaloy-4 and M5 alloy clad fuel rods are documented in published studies evaluating LOCA event conditions. For example, Figure 16 shows a typical M5 alloy clad fuel rod creep rate test simulation documented in an American Society for Testing and Materials (ASTM) publication (Reference 13). Figure 16 shows that the cumulative strain was insignificant at 5 seconds for an M5 alloy clad fuel rod. This test simulation was performed in the EDGAR-2 test facility for a pressure of approximately 175 bar (approximately 2500 psi) and a temperature of 600°C (approximately 1100°F). The plotted empirical cumulative strain associated with this LOCA event is more severe than the cumulative strain associated with non-LOCA events, in that the LOCA event features a depressurized reactor coolant system which results in higher pressure gradients across the cladding, and higher fuel rod temperatures and pressures.

Based on the preceding, it is conservative to use a time in DNB of less than 5 seconds as a criterion to ensure that DNB propagation of Framatome fuel with M5 cladding will not occur. The methodology and criteria for evaluating Westinghouse fuel remains unchanged.

Figure 16

Example EDGAR Test Simulation



SRXB RAI-4

The regulatory basis for this RAI is 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

The LAR dated July 6, 2018, presented information to demonstrate that the impact of M5® cladding thermal conductivity and cladding specific heat were appropriately considered in the non-LOCA transient analysis system response computer codes CENTS and HERMITE. However, the HERMITE code needs the heat conduction equation in the fuel pellet, gap and clad. Please clarify how the fuel specific input for the Framatome CE16 HTP™ fuel was generated such that the fuel-to-clad gap coefficient of conductance was appropriately modeled, including any effects of gadolinium on the fuel-to-clad gap coefficient of conductance.

Response to SRXB RAI-4

The fuel performance specific inputs to either the HERMITE or CENTS code for CE16HTP fuel are generated by the Framatome fuel performance code COPERNIC. In addition, COPERNIC includes any effects of the gadolinium.

The comparison of the CE16STD "generic" minimum and maximum gap conductance values to the COPERNIC calculated gap conductance values has shown that the CE16STD "generic" minimum value is conservative (closer to zero) than the minimum value that is calculated by COPERNIC. Thus, other than the value utilized for minimum gap conductance (CE16STD "generic" value), when modeling CE16HTP fuel, COPERNIC generated fuel performance specific inputs are utilized.

The HERMITE and CENTS codes used for CE16HTP evaluation **[[**

]].

With regard to the effect of M5 on non-LOCA licensing basis analyses, as stated above, and as discussed during the June 2019 site audit, **[[**

]].

Based on a review of typical non-LOCA licensing basis analyses, it has been determined that the CEA Ejection event is the only non-LOCA event which could result in clad temperatures reaching or exceeding the M5 phase transition temperature. For the CEA Ejection event, the COPERNIC based M5 cladding properties **[[**
]] (refer to the response to SRXB RAI-6). For other non-LOCA analyses, clad temperatures remain below the M5 phase transition temperature. The results of non-LOCA transients are mainly driven by the transient dynamics and the system

responses. The clad properties are of secondary importance when compared to the aforementioned parameters. Thus, the overall trend of transients is not impacted by clad material properties.

In conclusion, the use of CENTS and HERMITE codes **[[ ]]**, produces acceptable results when modeling non-LOCA events at Palo Verde for the CE16HTP fuel with M5 cladding.

CENPD-404-P-A (Reference 14) and WCAP-12610-P-A & CENPD-404-P-A Addendum 1-A (Reference 15) are examples of previous NRC approval of the use of **[[ ]]** in approved transient analyses computer codes.

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.

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.

This process is modified for CEA ejection events and is described in the response to SRXB RAI-6.

SRXB RAI-6

The regulatory basis for this RAI is 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

a. The licensee's supplemental letter dated October 18, 2018, states, in part, that "for the Reference (3) application, APS will use interim criteria published by NRC staff in Standard Review Plan, Section 4.2, in 2007 . . . with plant specific adjustments." It is not clear how this position will be implemented for mixed core scenarios. Please describe the intended application of the interim criteria on mixed core scenarios.

b. For UFSAR event 15.4.8, the LAR dated July 6, 2018, states that, "One (1) change to methodology required: The methodology in CENPD-190-A is required to be modified to allow the use of COPERNIC fuel performance code instead of FATES." Section 6.5.1, "M5® Cladding Impact on CEA Ejection Analysis," states that **[[ ]]**. Confirm these are the only inputs and parameters that COPERNIC will be used for. Additionally, the interim criteria take into consideration the effect of burnup and are presented as a function of oxide/wall thickness for PWRs. Please describe (1) how burnup effects will be considered for both the Framatome and co-resident fuel and (2) which fuel performance codes will be used to determine the oxide/wall thickness and how the limiting fuel enthalpy rise is determined in a mixed core.

c. In Section 6.5.4.2, "Fuel Failure Evaluation Results," in Attachment 8 of the LAR, the licensee states the following, in part:

> A comparison of the fuel failure percentage to the fuel failure percentage in the current UFSAR demonstrated that the fuel failure percentage for the CEA Ejection are bounded by the fuel failure percentages in the current PVNGS UFSAR. Consequently, the offsite and control room dose consequences meet the acceptance criteria.

Please provide the results of this comparison and demonstrate that the current UFSAR results are bounding.

Response to SRXB RAI-6

Part a Response

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.

APS will apply the interim criteria published by NRC staff in the 2007 edition of Section 4.2 of the Standard Review Plan to Framatome CE16HTP fuel, as indicated in the APS letter of October 18, 2018 (Reference 8). For transition cores that utilize Framatome and Westinghouse fuel products, each fuel type will be analyzed independently against the interim criteria using NRC-approved methodologies (e.g., fuel performance codes, corrosion models).

APS intended to clarify the phrase "plant specific adjustments" in Reference 8 with the bulleted paragraphs that appeared immediately below that phrase. For example, Reference 8 stated that APS uses radial fall-off curves in core reload design to address certain burnup effects, and stated that APS uses statistical convolution to determine fuel failure. With regard to core coolability, the statements made in Reference 8 about DNB propagation are modified by the APS response to SRXB RAI-3 herein.

Part b Response

The APS LAR (Reference 9) and the supplemental letter of October 18, 2018 (Reference 8) were intended to modify the CENPD-190-A CEA ejection analysis methodology by replacing the Westinghouse FATES3B fuel performance code with the Framatome COPERNIC code, for the purpose of analyzing Framatome CE16HTP fuel. The PVNGS CEA ejection rod heatup calculations are performed with the Westinghouse STRIKIN-II code, **[[**

]]

Note that 5 percent power is used as a proxy for hot zero power in the evaluation. Per the 2007 edition of Section 4.2 of the Standard Review Plan, 5% power is upper end of the hot zero power range. For confirmation, a 1 MW (i.e., 0.025% power) case was evaluated to confirm that the 5% power case provides more limiting results.

For a mixed core design, APS would perform the hot rod heatup analysis for each fuel type with the STRIKIN-II code, **[[**

]] its associated fuel vendor, not by another vendor.

The thermal conductivity and heat capacity of M5 cladding were specifically called out by APS in the LAR (Reference 9). **[[**

]]

With regard to oxide/wall thickness, the APS letter of October 18, 2018, states that the COPERNIC cladding corrosion model of BAW-10231P-A (Reference 33) would be adopted for CE16HTP fuel at PVNGS. The COPERNIC best estimate peak cladding oxide thickness **[[**

]] Thus the APS methodology includes discretionary conservatism and will bound transition cycles as well as a full core of CE16HTP. Power history envelopes and cycle lengths **[[**

]]

For Westinghouse fuel that is co-resident with Framatome CE16HTP fuel, the applicable cladding corrosion model is described in WCAP-12610-P-A & CENPD-404-P-A, Addendum 2-A (Reference 34), which the NRC approved as part of the PVNGS licensing basis in January 2018 (Reference 7). Specifically, Table 4-1 of that topical report addendum summarizes plant assessments that provided maximum predicted **[[**
]] given the plants' fuel designs and operating conditions. For example, **[[**

]] a burnup-dependent enthalpy rise limit for CE16STD fuel in support of the CE16HTP fuel transition.

In this manner, APS will specify appropriate enthalpy rise limits for both Westinghouse and Framatome fuel types during a mixed core transition. The amount of discretionary conservatism (margin) applied to the specified limits for each fuel type may change in the future.

Finally, during the June 2018 regulatory audit, the NRC staff noted that **[[**

]] and that, during some postulated CEA ejection events, fuel melt can occur near the fuel pellet rim rather than the fuel pellet centerline. It is APS's position that its CEA ejection methodology proposal provides conservative results relative to newer methods that **[[** **]]**. In support of this position, Enclosure Attachments 10 (non-proprietary) and 14 (proprietary) provide the results of a demonstration Framatome 3D CEA ejection analysis for PVNGS. The demonstration analysis models CE16STD-to-CE16HTP transition cycles as well as a full core of CE16HTP fuel, and shows that there are no prompt critical ejected rod worths and that no fuel melt will occur. The demonstration analysis predicts that the highest fuel temperature **[[**
]]. By comparison, the highest fuel temperature predicted **[[**

]]. The Framatome 3D CEA ejection analysis is being provided to support the APS position as stated above, and not as a licensing application request for review of a new methodology.

<u>Part c Response</u>

When APS performs a PVNGS licensing basis radiological dose consequence analysis for a Condition III (Infrequent) or Condition IV (Limiting Fault) design basis event, a bounding or limiting fuel failure value is typically assumed and the analysis results are compared to the applicable acceptance criteria for that event. APS, however, also performs cycle-specific

reload design analyses to ensure that these bounding or limiting values are not exceeded, taking into consideration specific characteristics related to the plant configuration, plant operating conditions, and the selected core reload design. This process is followed for full cores of one fuel type as well as for mixed transition cores.

For example, for postulated Main Steam Line Breaks (MSLBs), PVNGS UFSAR Section 15.1.5.5 assumes 1% fuel failure due to DNB in the radiological dose analysis, although cycle-specific analyses may show that no fuel failure is predicted to occur. For the CE16STD to CE16NGF transition at PVNGS, the associated NRC Safety Evaluation (Reference 7) concluded that "the post-trip phase of the MSLB events has been adequately analyzed and the results conservatively demonstrate that SAFDLs are not violated" because, in part, APS stated that it had options in reload design space to prevent fuel failure, including but not limited to adjusting the initial thermal margin of the plant. That NRC Safety Evaluation also concluded, however, that although "the MSLB event is classified as a limiting fault (i.e., Condition IV event), the licensee is ***typically*** able to demonstrate that SAFDLs will not be violated and no fuel damage will occur" (emphasis added). In this particular case, the NRC staff's use of the word "typically" does not invalidate the assumption of some allowable fuel failure for postulated MSLBs, it merely acknowledges that "a core design can be selected that ***conservatively precludes*** fuel failures for this Condition IV event" (emphasis added). Consistent with the preceding discussion, the MSLB event reported in Section 6.1 of Attachment 10 of the LAR assumes a fuel failure of 1 percent.

With regard to postulated PVNGS CEA ejection events, the licensing basis radiological dose consequence safety analysis is likewise driven by the amount of fuel that is assumed to fail due to DNB, as well as the amount that is assumed to experience incipient fuel melt. PVNGS UFSAR Table 15.4.8-6 assumes values of 19% fuel failure due to DNB and 0% incipient melt for the radiological dose analysis.

PVNGS reload core design analyses, however, for both CE16STD to CE16NGF and CE16STD to CE16HTP transitions, predict that a CEA ejection event during a first transition core would result in approximately 15% to 16% fuel failure due to DNB, and 0% fuel failure due to incipient melt, which are bounded by the fuel failure assumed in UFSAR Section 15.4.8. Likewise, a full core of CE16NGF or CE16HTP would yield approximately 10% to 11% fuel failure due to DNB and 0% due to incipient melt.

Again, it is APS's position that its CEA ejection methodology provides conservative results relative to newer analytical methodologies. In support of this position, Enclosure Attachments 10 (non-proprietary) and 14 (proprietary) provide a demonstration Framatome 3D CEA ejection analysis. The report shows that no fuel melt is predicted to occur, and that DNB-related fuel failure would be **[[ ]]**. This demonstration analysis models CE16STD-to-CE16HTP transition cycles as well as a full core of CE16HTP fuel. This analysis is being provided as supporting the APS position stated above, and not as a licensing application request for review of a new methodology.

SRXB RAI-7

The regulatory basis for this RAI is 10 CFR Part 50, Appendix A, GDC 16, “Containment design”; GDC 38, “Containment heat removal”; and GDC 50, “Containment design basis.”

Section 8, “Containment Response Analysis,” of Attachment 8, of the LAR dated July 6, 2018, discusses the parameters which are not affected by a change to Framatome CE16 HTP™ fuel; however, there is no discussion regarding what has changed as a result of the fuel transition. Describe any input changes to the mass and energy release analyses and describe how the analysis of record continues to remain valid for the main steam line break accident and LOCA.

Response to SRXB RAI-7

Mass and Energy (M&E) release Analyses of Record (AORs) were not explicitly performed for the Framatome CE16HTP fuel transition. Rather, Westinghouse identified parameters of interest and their values, and then Framatome confirmed the applicability of those parameter values for Framatome CE16HTP fuel. The following provides a summary of the effort.

Mass and Energy Releases

The reflood/post-reflood analysis uses a resistance network to model the core and RCS loops. These resistances are calculated based on the pressure drops from the blowdown analysis. The evaluation and comparison performed for the blowdown analysis showed that the change in pressure losses were insignificant to the result of the transient and therefore will have no impact on the reflood/post-reflood analysis as well.

Evaluated fuel parameters included the following:

- Core average linear heat rate – Scaling to a core power of 4070 MWt, the maximum Framatome core average linear heat rate is 5.735 kW/ft. The AOR blowdown analysis calculated a core average linear heat rate value is 5.925 kW/ft. The AOR blowdown value is larger and results in more energy being transferred to the coolant, which is conservative for containment mass and energy releases.
- Pellet and cladding geometry – Three primary dimensions that are compared for the CE16STD and Framatome CE16HTP fuel assemblies are the pellet outside diameter, cladding inside diameter, and cladding outside diameter. All three of these dimensions are the same between CE16STD and the Framatome CE16HTP fuel. Any small differences in the flow area will be insignificant since the flow area has already been included in the core pressure drop that was previously discussed and evaluated to have a negligible impact.
- Centerline temperature – All maximum fuel centerline temperatures remain bounded by the existing CE 16x16 AOR with the exception of the LOCA peaking case where the Framatome CE16HTP fuel centerline temperature of 3014°F slightly exceeds the AOR value of 3000°F. The maximum Framatome CE16HTP fuel centerline temperature observed for the LOCA peaking corresponds to axial node

16 at a power of 11.34 kW/ft. The maximum CE 16x16 fuel centerline temperature corresponds to axial node 18. The maximum centerline temperatures appearing at a different axial node for the two analyses can be attributed to the COPERNIC adjustment of the axial power shape. Based on the different computer code use of axial power shape and the modeling and calculation differences between the two analyses, the difference of 14°F is negligible.

- Decay heat – The AOR assumed a 102% thermal rating modifier to set the initial core power level. This is bounding for the Framatome CE16HTP fuel decay heat.
- Metal/water reaction – The Zirconium-water reaction option is enabled for the LBLOCA M&E release analysis. However, the M&E release analysis biases the reactor core parameters to extract as much energy from the fuel and components as possible in order to generate steam. This results in lower fuel clad temperatures such that the metal-water reaction is negligible.

Tributary Line Breaks

The short-term mass and energy releases generated for the tributary line break transient is a short duration event (approximately 1 second). There is not sufficient time for the reactor core, and the primary and secondary sides to interact to significantly affect the mass and energy releases. The initial conditions that have an impact on the analysis such as the RCS pressure and temperature have not changed as a result of the Framatome CE16HTP fuel transition. The tributary lines that are postulated to break remain the same. Therefore, the Framatome CE16HTP fuel will have no impact on this analysis.

- The full-power feedwater temperature used for the AOR is 450°F, and the full-power feedwater temperature for Framatome is 448°F. A higher feedwater temperature results in more energy being transferred to the steam generator secondary side until the feedwater system isolated. This results in higher energy steam being release during the blowdown of the steam generator and therefore a higher pressure and temperature containment response. As such, the Framatome CE16HTP fuel is bounded.
- The main steam line break M&E evaluation calculated the total energy in the fuel region in hand calculations based on average temperatures, UO_2 density, specific heat, and volume. The AOR total fuel region energy is greater than the Framatome fuel region energy (34.67 MBtu vs. 34.47 MBtu). A higher fuel region total energy results in more energy being transferred to the reactor coolant system which will then transfer more energy to the steam generator secondary side. This results in higher energy steam being released during the blowdown of the steam generator and therefore a higher pressure and temperature containment response. As such, the Framatome CE16HTP fuel is bounded.

V-APS RAI-01

VIPRE-01 Version Number	
What is the initial version and MOD of VIPRE-01 that APS is using to create its version of VIPRE-APS?	
Associated Section	Theory Manual
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 SRP 15.0.2 Subsection III.3.a

Response to V-APS RAI-01

VIPRE-01 MOD 2.6 was modified to create VIPRE-01 MOD 2.6 APS, also known as VIPRE-APS.

V-APS RAI-02

Changes to VIPRE-01	
Provide the complete list of changes between VIPRE-01 MOD 2.0 (the version of VIPRE-01 approved by the NRC staff) and the version of VIPRE-01, which was used to generate VIPRE-APS. Additionally, provide the justification for each change.	
Associated Section	Theory Manual
Associated Regulations and Guidance	10 CFR Part 50, Appendix A, GDC 10; 10 CFR 50.36; 10 CFR 50.34, "Contents of applications; technical information"; and 10 CFR Part 50, Appendix B SRP 15.0.2 Subsection III.3.a

Response to V-APS RAI-02

VIPRE-01 MOD 2.0 is the version of VIPRE-01 approved by the NRC staff in October 1993. VIPRE-01 MOD 2.6 is the version of VIPRE-01 used as the basis for VIPRE-01 MOD 2.6 APS also known as VIPRE-APS.

Appendix C of Volume 2 of the VIPRE-01 topical report, EPRI NP-2511-CCM-A (Reference 16), includes an extensive code modification summary that describes every change made to the code between VIPRE-01 MOD 2.0 and VIPRE-01 MOD 2.6. This code modification summary is included herein as Table 7 and clearly ties code corrections to the trouble report numbers. The summary also identifies code changes that were classified as enhancements (e.g., going from single precision to double precision), and those that added new models or options that had not previously been reviewed for acceptance by NRC.

Two such models or options have been introduced since the approval of VIPRE-01 MOD 2.0. Those are the Chexal-Lellouche (C-L) drift flux model and the C-L correlation "algebraic" option. Neither is used by APS. Code corrections and enhancements do not alter the VIPRE-01 methodology originally approved by the NRC.

Table 7

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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Table 7 (continued)

Code Modification Summary (VIPRE-01 MOD 2.0 to VIPRE-01 MOD 2.6)

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V-APS RAI-03

Application of VIPRE-APS	
Confirm that VIPRE-APS will be used in a manner consistent with VIPRE-W as described in WCAP-14565-P-A/WCAP-15306-NP-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-hydraulic Safety Analysis," dated October 1999 (ADAMS Accession No. ML993160158).	
Associated Section	Theory Manual
Associated Regulations and Guidance	10 CFR Part 50, Appendix A, General Design Criterion 10; 10 CFR 50.36; 10 CFR 50.34; and 10 CFR Part 50, Appendix B SRP 15.0.2 Subsection III.3.a

Response to V-APS RAI-03

VIPRE-APS will utilize modeling in a manner similar to that described in WCAP-14565-P-A/WCAP-15306-NP-A (References 17, 19, and 20). The VIPRE-01 MOD 2.6 APS (referred to herein as VIPRE-APS) modeling assumptions supporting the CE16NGF fuel type and the corresponding ABB-NV/WSSV CHF correlations are the same as those utilized and approved in Reference 7. These include the choice of model geometry, two-phase flow models and correlations, heat transfer correlations, and turbulent mixing models. As such, the methodology and guidelines used to create the VIPRE-APS model for the Palo Verde plant cores with APS licensed Westinghouse fuel products will be consistent with WCAP-14565-P-A/WCAP-15306-NP-A (Reference 17) and as modified per the NRC approved Next Generation Fuel LAR (Reference 7). Note the modeling options discussed are independent of the Thermal-Hydraulics code but rather dependent on the particular fuel type being modeled.

APS intends to apply the deviations to WCAP-14565-P-A / WCAP-15306-NP-A previously approved for the CE16STD and CE16NGF fuel type in Reference 7 to model the Westinghouse fuel types with VIPRE-APS consistent with approved NRC safety evaluation for Next Generation Fuel (Reference 7). Note that the deviations to WCAP-14565-P-A / WCAP-15306-NP-A listed below are also applicable to VIPRE-APS in the modeling of the BHTP CHF correlation, which supports CE16HTP thermal-hydraulic safety analyses.

1. Reference 17 Section 3.1 discusses the model geometry and radial nodalization as depicted in Figure 4-2 utilizing **[[ ]]** VIPRE-W model for a 2-loop PWR with a 14x14 lattice. Reference 17 does not discuss or provide a 2-loop PWR 16x16 lattice model. VIPRE-APS will utilize the same, more detailed, 2-loop PWR 16x16 lattice geometry and radial nodalization scheme developed and approved for the CE 16x16 lattice plant by the NRC in Reference 7 in the VIPRE-APS models of the APS licensed fuel products.

2. Reference 17 Section 3.3 discusses the modeling of fuel rods as conduction fuel rods. Note that the VIPRE-APS code will not be utilized for analyses that require model of conduction fuel rods and VIPRE-APS will not be utilized to perform analyses in the post-CHF film-boiling region.

3. Reference 17 Section 3.9.2 states that a **[[ ]]** is assumed. Consistent with the methods approved by the NRC for VIPRE-W in Reference 7, VIPRE-APS will not **[[ ]]**. In addition, as approved by the NRC in Reference 7 for VIPRE-W, VIPRE-APS will incorporate **[[ ]]** to model **[[ ]]** for the licensed fuel products at Palo Verde.

4. Reference 17 Section 4.1.2 discusses the mixed or transition core and the possible introduction of a **[[ ]]** caused by the **[[ ]]**. Mixed or transition core effects on the CE16STD, CE16HTP, or CE16NGF fuels are evaluated with respect to DNB. For each mixed or transition cycle, the entire core is reviewed and the limiting assembly candidates (CE16STD, CE16HTP, and CE16NGF) are then analyzed with their respective CHF correlation. The response to MIX RAI-01 provides additional details regarding APS's approach to modeling mixed or transition core designs. VIPRE-APS will utilize the same modeling as approved by the NRC for VIPRE-W and the Westinghouse fuel products (Section 5.5 of Reference 7). Note VIPRE-APS will utilize the approved VIPRE-W modeling and process that **[[ ]]**.

5. Reference 17 Section 3 states that VIPRE-W is based on the **[[ ]]** approach. Consistent with the approved VIPRE-W modeling, VIPRE-APS will also utilize a similar **[[ ]]** process as outlined in Reference 17, Section 3. Additionally, VIPRE-APS will utilize the **[[ ]]** as cited in Section 5.5 of Reference 7 and as previously approved for use with the VIPRE-W and the TORC thermal-hydraulic code (References 7 and 18). The application of **[[ ]]** with VIPRE-W and extended to VIPRE-APS supports the **[[ ]]**. APS will utilize the **[[ ]]** consistent with the previously approved VIPRE-W uses per Section 3.5.5.5 of Enclosure 4 to the NRC's SE for CE16NGF fuel (Reference 7).

6. Reference 19 Table 3-2 provides the turbulent mixing coefficient ABETA and turbulent momentum factor (FTM) recommended values of **[[ ]]** and **[[ ]]** for VIPRE-W as applied to the APS licensed Westinghouse fuel products. Consistent with the approved VIPRE-W modeling in Reference 20 of the NRC's Safety Evaluation, VIPRE-APS will utilize a value of ABETA of **[[ ]]** and a

value of FTM of **[[ ]]** for the VIPRE-APS modeling of the Palo Verde licensed Westinghouse fuel products. Core designs utilizing the Framatome CE16HTP fuel product will model values of ABETA of **[[ ]]** and FTM of **[[ ]]** when analyzing the CE16HTP fuel type as discussed in the response to V-APS RAI-05.

7. Reference 17 Sections 3.9.1 and 3.9.3 discuss the application of the engineering enthalpy rise hot channel factor and the engineering heat flux hot channel factor. Reference 17 discusses both the deterministic and statistical method to account for the application of the enthalpy rise and heat flux factors. Consistent with current licensing and practice for all APS licensed fuel products, VIPRE-W and VIPRE-APS will continue to **[[ ]]**.

8. Reference 17 Table 3-1 lists the VIPRE model selections; some of these selections have been superseded as discussed in Items 1 through 7 of this RAI response. Additional selections have been superseded for CE16HTP fuel by the closure model discussion presented in the response to V-APS RAI-05.

V-APS RAI-04

<table>
<tr><td colspan="2">Confirmation of Conditions and Limitations of VIPRE-01
Provide confirmation that the following conditions and limitations form VIPRE-01 MOD-1.0 and VIPRE-01 MOD-2.0 will be satisfied:
a. Confirm that the application of VIPRE-APS is limited to PWR licensing calculations with heat transfer regime up to CHF and will not exceed CHF.
b. Confirm that VIPRE-APS will abide by the quality assurance procedures described in Section 2.6 of the Safety Evaluation Report (SER) for VIPRE-01 MOD-1.
c. Confirm that VIPRE-APS will not be used in the following situations:
i. Specific two-phase flow conditions that are characterized by large relative velocity between the phases or radical changes in flow regime, such as low-flow boil-off, annular flow, stratified two phase flow, or countercurrent flow.
ii. Phenomena dominated by local pressure such as flow-down transient, boiling incepted at low pressure, or boiling-water reactor transient flow instability.
iii. Free-field situations not dominated by wall friction.
iv. Situations out of the applicable range of the constitutive correlations.</td></tr>
<tr><td>Associated Section</td><td>Previously Reviewed and Accepted Codes and Methods</td></tr>
<tr><td>Associated Regulations and Guidance</td><td>10 CFR Part 50, Appendix A, GDC 10; 10 CFR 50.36; 10 CFR 50.34; and 10 CFR Part 50, Appendix B</td></tr>
</table>

Response to V-APS RAI-04

Part a Response

Application of VIPRE-APS is limited to PWR licensing calculations and will not exceed the CHF regime as discussed in Condition and Limitation 1 of Section 5.1, Attachment 5 of the LAR (Reference 9).

Part b Response

Applications of VIPRE-APS will abide by the quality assurance procedures described in Section 2.6 of the SER for VIPRE-01 MOD-1 as discussed in Condition and Limitation 5 of Section 5.1, Attachment 5 of the LAR (Reference 9).

Part c Response

VIPRE-APS will not be used in the following situations as discussed in Condition and Limitation 3 of Section 5.2, Attachment 5 of the LAR (Reference 9):

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

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

In addition, VIPRE-APS will not be used in the following situations:

iii. VIPRE does not produce accurate results in free-field situations because of the omission of several cross-coupling terms from the lateral momentum equation. APS only intends to use VIPRE to analyze standard PWR rod/tube bundles geometries, which exhibit large lateral flow resistance relative to axial flow resistance and are therefore dominated by wall friction.

iv. The BHTP CHF will be restricted within its ranges of applicability as described in response to BHTP RAI-04. Other constitutive correlations will be used consistent with previous NRC approvals.

V-APS RAI-05

Benchmark Comparison	
Provide justification for the modeling options chosen for VIPRE-APS, including specific and justifying the two-phase friction multiplier.	
Associated Section	Validation of the Closure Relationships
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 SRP 15.0.2 Subsection III.3.b

Response to V-APS RAI-05

This RAI response confirms compliance with Condition 3 of the VIPRE-01 SER (References 21 and 22), which requires that each organization using VIPRE-01 provide justification for specific modeling assumptions, including the choice of two-phase flow models and correlations, heat transfer correlations and turbulent mixing models. As such, the methodology and guidelines used to create the VIPRE-APS model for the Palo Verde plant cores are described herein.

The VIPRE-APS correlations and models selected and discussed herein were previously approved by the NRC for the VIPRE-01 (References 21 and 22) model. The modeling choices described in Table 5-5 of Attachment 10 of the LAR (Reference 9), which are not plant specific, were developed in a manner consistent with the NRC approval of TORC (Reference 18) and with standard industry practice. These modeling choices were utilized in the qualification or verification of the BHTP CHF correlation and the performance of thermal-hydraulic CE16HTP related analyses. Note that the responses provided below regarding the Table 5-5 of Attachment 10 of the LAR modeling options are also applicable to the modeling options described in Table 5-7 of Attachment 10 of the LAR pertaining to the VIPRE-W codes modeling of the BHTP CHF correlation and the performance of the CE16HTP thermal-hydraulic related analyses.

The following discussions describe the VIPRE-APS and VIPRE-W modeling choices that are applied to the CE16HTP fuel product. With the exception of the turbulent mixing ABETA term, these modeling choices are unchanged from Table 5-5 and Table 5-7 of Attachment 10 of the LAR.

Water Properties

The VIPRE-APS model utilizes the default EPRI curve-fit functions for water properties as described in Reference 16 and found acceptable for licensing calculations based on the NRC safety evaluation of the VIPRE-01 code (Reference 21) provided licensees provide justification for their use. The VIPRE-APS BHTP CHF correlation benchmark analysis

performs the testing and analysis that demonstrates the selection of the EPRI water properties functions is accurate and justified for the CE16HTP fuel type.

Two-Phase Friction, Subcooled Void, and Bulk Boiling Void Correlations

VIPRE-APS has the same empirical correlations as VIPRE-01 to simulate two-phase flow effects (Reference 21). These correlations can be grouped in three major categories: 1) two-phase friction multipliers, 2) subcooled void correlations, and 3) bulk boiling void correlations. In Reference 23, a sensitivity study was performed to assess the differences in the performance of the various correlations and the EPRI models were selected as the default models for VIPRE-01. The NRC, in Reference 21, concluded that the EPRI void models and EPRI correlation for two-phase friction are acceptable for licensing calculations provided licensees provide justification for their use.

A VIPRE-APS BHTP CHF correlation benchmark analysis was performed to verify that this set of two-phase flow correlations provided the closest comparison to the NRC approved Framatome LYNXT code for Framatome BHTP CHF correlation and was determined to be the most suitable for Palo Verde applications.

The VIPRE-APS model selections are:

- Subcooled Void Model: EPRI
- Bulk Boiling Void Model: EPRI
- Two-Phase Friction Multiplier: EPRI
- Hot Wall Friction Correlation : NONE

Axial Friction

Palo Verde will apply the Framatome recommended axial friction factor coefficients of **[[ ]]** for both turbulent and laminar flow. These Framatome recommended friction factors were utilized in the VIPRE BHTP CHF correlation benchmark analysis and shown to provide consistent results with the BHTP Technical Report (Reference 2). APS performed sensitivity studies that indicated there was no sensitivity to the axial friction factors selected. Therefore, a value **[[ ]]** for both turbulent and laminar flow are utilized in the VIPRE-APS and VIPRE-W CE16HTP fuel models.

Heat Transfer

VIPRE-APS requires the user to select the heat transfer correlations that describe the boiling curve. These selections (except the Single Phase Forced Convection Correlation), however, are only applied to the heat transfer solution if the conduction model is used. VIPRE-APS uses the "DUMY" rod model; therefore, the conduction model is ignored. Use of this rod model requires the user to provide the fuel rod surface heat flux as one of the operating input conditions. For steady-state statepoint analyses this value is calculated based on core thermal power. Fuel rod surface heat flux forcing functions for transient calculations are provided by an NRC-approved transient system code (e.g., CENTS). Transient system codes account for fuel conduction, gap conductance, and delayed energy transport effects. Therefore, the use of the DUMY rod model appropriately includes the relevant effects into the analysis.

Forced Convection, Subcooled Boiling, and Film Boiling Correlations

Although wall heat transfer is not used in the VIPRE-APS models by selecting the DUMY rod option, the remaining heat transfer options must be input. The default VIPRE-01 input options were selected. For single-phase forced convection to liquid, the Dittus-Boelter correlation is used. For subcooling and saturated boiling, the combination of Thom and Dittus-Boelter is used, where single-phase correlation heat transfer is corrected by excess wall temperature to provide continuous heat transfer for transition region. The VIPRE-01 default correlation for the boiling curve peak EPRI is selected. The Tong-Young correlation for transition boiling and Bishop-Sandberg-Tong for film boiling are selected for the VIPRE-APS models. The selected heat transfer options were implemented and tested within the VIPRE-APS BHTP CHF correlation benchmark analysis to verify that this set of heat transfer correlations provided results consistent with the results already approved by NRC for the Framatome BHTP CHF correlation.

The Single Phase Forced Convection is modeled with the standard Dittus-Boelter correlation, which is commonly used for this type of configuration (Reference 16):

$$h_{DB} = 0.023 \times Re^{0.8} \times Pr^{0.4} \times \frac{k}{D_e}$$

where Re is the Reynolds number for the liquid, Pr is the Prandtl number, k is the thermal conductivity of the fluid (Btu/sec-ft-°F) and D_e is the hydraulic diameter in ft. This selection is consistent with the NRC approved TORC models (Reference 18) and with standard industry practice.

Turbulent Mixing

The VIPRE-APS turbulent mixing model is identical to the NRC approved VIPRE-01 model that accounts for the exchange of energy and momentum between adjacent subchannels due to turbulence. Note this is not a turbulence model, but rather an attempt to empirically account for the effect of turbulent mixing. The following inputs are required as input to this model:

- Turbulent Momentum Factor (FTM), which can range from 0.0 to 1.0, measures how efficiently the turbulent crossflow mixes momentum. The VIPRE-01 User's Manual (Reference 16) recommends a value of 0.8 for FTM and explains that VIPRE is not very sensitive to the value of FTM. The Palo Verde model follows this recommendation. The FTM value selected was further confirmed in the BHTP CHF benchmark and the sensitivity calculations.

  The model for turbulent mixing chosen for single phase mixing describes the mixing as **[[**

  ]]. ABETA is dependent on the particular fuel type and can range in value from 0.0 to 0.1. Framatome recommends a value of **[[ ]]** for ABETA for use with the BHTP CHF correlation. A sensitivity study evaluating ABETA values from 0.0035 to 0.06 was performed in the BHTP CHF benchmark. An

ABETA value of **[[ ]]** was confirmed to yield results best aligned with the test data. Therefore, a value of **[[ ]]** for ABETA is utilized in the BHTP VIPRE-APS model. The turbulent mixing ABETA value of **[[ ]]** documented in this response supersedes the values in Table 5-5 and Table 5-7 of Attachment 10 in the LAR.

V-APS RAI-06

Single Version of the Evaluation Model (EM)	
Confirm that all runs of VIPRE-APS were performed with a single version of the computer code and were implemented in a consistent manner.	
Associated Section	Single Version of the Evaluation Model
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 SRP 15.0.2 Subsection III.3.d

Response to V-APS RAI-06

The licensing applications (i.e., VIPRE-APS validation to BHTP test data, VIPRE-APS to VIPRE-W benchmark) were performed with a single version of the computer code (VIPRE-01 MOD 2.6 APS, which is also known as VIPRE-APS).

The licensing applications were implemented with modeling options consistent with the descriptions detailed in Section 5.4.1.1 of Attachment 10 of the LAR (Reference 9) as modified by the RAI responses (e.g., V-APS RAI-05).

V-APS RAI-07

Sensitivity Studies	
APS should confirm that the axial and radial noding used in the safety analysis is consistent with the methodology presented in VIPRE-W or perform sensitivity studies, which justify its chosen axial and radial noding.	
Associated Section	Single Version of the Evaluation Model
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 SRP 15.0.2 Subsection III.3.d

Response to V-APS RAI-07

Background

APS has two versions of the VIPRE-01 code that are capable of modeling the BHTP Critical Heat Flux Correlation. VIPRE-01 MOD 2.6 (hereafter referred to as VIPRE-01) added the ability to model the BHTP CHF correlation to the pre-existing VIPRE-01 code version based on EPRI Topical Report NP-2511-CCM. VIPRE-01 MOD 2.6 APS (hereafter referred to as VIPRE-APS) modified VIPRE-01 MOD 2.6 to handle a two-pass process.

Axial Noding

BHTP sensitivity studies were performed to verify the VIPRE code axial noding used in the safety analysis.

The axial noding sensitivity studies were performed with both VIPRE-W and VIPRE-01. The same sensitivity study results would have been obtained with VIPRE-APS, since its coding changes are not relevant to the axial node input changes made in the sensitivity study.

The axial noding sensitivity studies support the modeling decisions made for the development of VIPRE-W and VIPRE-01 base decks that model 79 axial nodes. APS intends to use an axial noding scheme of approximately 2 inches or smaller height corresponding to 79 or more axial nodes. An axial node sensitivity summary figure is provided in Figure 17. This figure shows that the axial noding scheme utilized for both VIPRE-W and VIPRE-01 provides consistent results with more detailed nodalization schemes for a nominal base case, a case at high power conditions, a case at low flow conditions, and a case with intermediate high power and low flow conditions. The axial noding sensitivity studies determined that there is no significant change in calculated minimum DNBR values until axial node size increases to approximately 6 inches.

Radial Noding

APS will utilize the same 2-loop PWR 16x16 lattice geometry and radial nodalization scheme developed for the CE 16x16 lattice plant as approved for APS use in the Next Generation Fuel Safety Evaluation (Reference 7). Selected analyses (such as transition cores) may be analyzed with more detailed models.

Radial noding studies have been previously performed for VIPRE-01 and VIPRE-W. EPRI NP-2511-CCM (Volume 4, Section 7.1.2, Reference 24) performed sensitivity studies on radial noding and recommends that at least one full row of subchannels surround the hot channel, because the hot channel mass velocity and minimum DNBR are relatively insensitive to the channel layout as long as a few subchannels are placed adjacent to the hot channel. APS models are more detailed than the minimum recommendation of the sensitivity studies on radial noding performed in EPRI NP-2511-CCM. The sensitivity study on radial noding is not dependent on the modeling of Framatome CE16HTP fuel design with the BHTP CHF correlation.

Figure 17

[[

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BHTP RAI-01

<table>
<tr><td colspan="2">Measured to Predicted Comparison
For BHTP implemented into both VIPRE-W and VIPRE-APS, describe the method of determining the measured CHF value from the experimental test data and the method of determining the predicted CHF value. Confirm that those methods are consistent with the method used in the initial approval of the BHTP correlation or justify the new method if it is different.</td></tr>
<tr><td>Goal</td><td>G3.1 – Validation Error</td></tr>
<tr><td>Associated Regulations and Guidance</td><td>10 CFR Part 50, Appendix A, GDC 10; 10 CFR 50.36; 10 CFR 50.34; and 10 CFR Part 50, Appendix B</td></tr>
</table>

Response to BHTP RAI-01

The experimental data consists of the test bundle conditions (exit pressure, inlet temperature, inlet mass flux, and bundle power distribution) and the specific heater rod and axial location of the measured thermal excursion. Reference 1 provides detailed information for the experimental configuration and test results, and the APS response to BHTP RAI-02 discusses its usage in this submittal.

To determine each measured CHF data point, the test bundle conditions were input to VIPRE-APS and VIPRE-W simulations of the test. The location of minimum DNBR determined by VIPRE using the BHTP CHF correlation was used to define the hot channel and axial elevation at which to extract the measured and predicted values. Each measured CHF data point is defined as the local heat flux for the rod producing the minimum DNBR. This method of simulating the test using a sub-channel code then extracting the "measured" local heat flux for the rod producing the minimum DNBR is consistent with the method used in BAW-10241(P)(A) (Reference 2).

As with the measured CHF data, the predicted CHF was determined at the location of minimum DNBR predicted by VIPRE using the BHTP CHF correlation. The predicted CHF value is the BHTP predicted critical heat flux for the local coolant conditions at the location of minimum DNBR. This method is consistent with that used by Framatome in the BHTP correlation development in Reference 2.

BHTP RAI-02

Validation Data	
Confirm that the same data that was used to validate BHTP in its initial submittal was also used to validate BHTP's performance in VIPRE-W and VIPRE-APS.	
Goal	G3.2.1 – Validation Data
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 BHTP RAI-02

The experimental test data that APS used to validate BHTP for use in both VIPRE-W and VIPRE-APS were provided by Framatome and are consistent with that used in BAW-10241(P)(A) (Reference 2). The extended range of data presented to the NRC in Appendix A of Reference 2 were not directly used by APS to validate BHTP in VIPRE, but are addressed in the response to BHTP RAI-03.

BHTP RAI-03

Expanded Range of BHTP	
Justify the use of the expanded application domain for the BHTP model for computer codes other than LYNXT.	
Goal	G3.2.2 – Application Domain
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 BHTP RAI-03

The application range of the BHTP correlation for LYNXT was expanded in Revision 1 of BAW-10241(P)(A) (Reference 2). The original application range and the "extended range" described in the topical report are summarized in Table 8.

The extension was performed by assessing the BHTP performance for **[[ ]]** additional statepoints, from the same CHF test campaigns, that were not a part of the original BHTP correlation development. The BHTP correlation is able to very conservatively predict the additional **[[ ]]** data points with an average "predicted/measured values," or P/M, of **[[ ]]** (See Section A.3.2 of Reference 2). The original design limit for the BHTP correlation was conservatively maintained for applying the BHTP correlation to the "extended range" using LYNXT.

The BHTP CHF Design Limit for VIPRE is based on the same **[[ ]]** data points as the original approval of the BHTP correlation for LYNXT (i.e., no extended pressure, mass flux, or quality range as approved in Reference 2).

A comparison of the limiting subchannel exit conditions between VIPRE and LYNXT was performed to quantify the code-to-code differences within the original **[[ ]]** CHF statepoint cases and to determine if any trends or biases of significance exist in the exit condition differences that would indicate a potential significant impact on the conservative nature of the BHTP correlation with VIPRE in the "extended range" that was demonstrated in Appendix A of Reference 2 for LYNXT application. The exit conditions were selected for the comparison **[[ ]]**.

The comparison of exit quality differences between the two codes, (VIPRE quality – LYNXT quality), show that the average difference for all CHF statepoints was **[[ ]]**. The average difference in exit quality predictions **[[ ]]**. The slope of this quality difference decreasing versus pressure was comparable for all the CHF tests.

The comparison of the exit mass flux differences between the two codes, **[[** **]]** was examined for all the original **[[** **]]** CHF test statepoints. Overall, the average exit mass flux difference was approximately **[[**

]].

The examination of the exit local conditions from VIPRE and LYNXT show that local coolant condition predictions do differ between CHF tests and within a given CHF. Had VIPRE been used to analyze the additional CHF test data used by Framatome to defend the application of LYNXT to the "extended range" (as shown in Appendix A of Reference 2), similar local coolant condition differences, as quantified examining the original **[[** **]]** data point data base, would be expected to be observed. Such differences of local mass flux and quality predictions between VIPRE and LYNXT would be small and their impact on the P/M values would be offset by the significant conservatism quantified for LYNXT application in the "extended range." Figures A.2, A.5, and A.7 in Appendix A of Reference 2 show the extent of significant conservatism for the BHTP correlation regarding the quality, local mass flux, and pressure for the "extended range" CHF test data.

Therefore, the application of the BHTP CHF correlation in the "extended range" using the VIPRE code with its respective CHF Design Limit will have a similar magnitude of conservatism as the LYNXT application. VIPRE will be implemented consistent with the extended range limits documented in Section 4.0 Table 1 of the NRC's Safety Evaluation (SE) of Reference 2.

Table 8 provides the extended BHTP CHF ranges as approved by the NRC in the Safety Evaluation of Reference 2. The values cited on Table 8 replace the applicable values cited on Table 4-1 of Attachment 5 and Table 5-1 of Attachment 10 of the LAR (Reference 9).

VIPRE-APS data was used as the basis for the assessment of the applicability of the BHTP extended ranges to VIPRE as compared to LYNXT. The response to BHTP RAI-08 provides justification that VIPRE-APS and VIPRE-W produce equivalent results; therefore, this assessment is also applicable to VIPRE-W.

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]].

Table 8

Original and Extended Range of Independent Variables
for the BHTP CHF Correlation

	Original Application Range		Extended Range	
Independent Variable	Minimum Value	Maximum Value	Minimum Value	Maximum Value
Pressure, psia	1775	2425	1385	2425
Mass Velocity, $Mlbm/hr\text{-}ft^2$	0.897	3.549	0.492	3.549
Thermodynamic Quality at CHF	-0.130	0.344	No Lower Limit	0.512

BHTP RAI-04

Restricted Domain	
Describe the process used to restrict the BHTP CHF model to its application domain in both VIPRE-W and VIPRE-APS.	
Goal	G3.2.6 – Restricted Domain
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 BHTP RAI-04

The parameters in the BHTP CHF correlation application domain for coolant conditions are pressure, local mass flux, inlet enthalpy, and local quality. Consistency with the coolant condition limits for all versions of VIPRE is assured through a combination of controls. Most parameters are controlled automatically within the thermal-hydraulic codes or with automation tools that discard results that do not fall within the application domain. Administrative controls are used for parameters that are not filtered automatically.

BHTP RAI-05

Poolability	
Examine the validation error (e.g., predicted to measured CHF) for potential inconsistences. This should include examining the poolability of the different subgroups, especially the subgroup of data from the BHTP database that would be expected to have the most similar behavior to the CE16 HTP™ fuel. Confirm that all subgroups are poolable for both VIPRE-W and VIPRE-APS.	
Goal	G3.3.1 – Poolability
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 BHTP RAI-05

The implementation of the BHTP CHF correlation into VIPRE has not introduced poolability concerns relative to the prior approval of the BHTP correlation. A detailed review of the data was performed resulting in the following conclusions:

- The raw test data used in the VIPRE implementation is identical to the raw test data used in the LYNXT implementation. Observed differences in the local conditions between the two codes are small and will not significantly impact the poolability.
- Predicted/Measured (P/M) values were analyzed for specific subgroups:
  - Subchannel type – the pooled results in Table 9 **[[ ]]** are comparable between LYNXT and VIPRE implementations of the BHTP correlation.
  - Axial power shape - the pooled results in Table 9 **[[ ]]** are comparable between LYNXT and VIPRE implementations of the BHTP correlation.
- **[[ ]]**. Per the response to SNPB RAI-2, the BHTP CHF correlation design limit is **[[ ]]**. The average P/M for this test is conservative (< 1.0), and the population contains only **[[ ]]**. This supports the claim that the BHTP correlation adequately predicts the specific performance of the fuel product intended for use in Palo Verde.

Table 9 summarizes key VIPRE poolability data.

VIPRE-APS data was used as the basis for this assessment. The response to BHTP RAI-08 provides justification that VIPRE-APS and VIPRE-W produce equivalent results; therefore, this assessment is also applicable to VIPRE-W.

Table 9

VIPRE Poolability Summary

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BHTP RAI-06

Non-Conservative Sub-Regions	
Examine the validation error (e.g., predicted-to-measured CHF) for possible non-conservative sub-regions. This should include examining the validation error expected to have the most similar behavior to the CE16 HTP™ fuel. Confirm that all there are no obvious non-conservative sub-regions with BHTP's application in both VIPRE-W and VIPRE-APS.	
Goal	G3.3.2 – Non-Conservative Sub-Regions
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 BHTP RAI-06

The BHTP CHF correlation was developed for application with the HTP spacer grid with and without the corresponding IFM grids for reactor applications. The **[[ ]]** CHF test configurations were utilized to successfully defend the approval of the HTP CHF correlation (using the XCOBRA-IIIC code) and later the BHTP CHF correlation (using the LYNXT and COBRA-FLX codes) for application to various HTP fuel designs.

The examination of the validation error of the VIPRE-based results was made to see if the VIPRE application of BHTP produced results indicating a significantly less conservative behavior than the original LYNXT development basis. The VIPRE mean predicted-to-measured (P/M) critical heat flux varies, as expected, from test to test within the data base. **[[**

]]. The mean P/M for **[[**
]] which indicates the VIPRE BHTP predictions align well with the test.

The statepoints that exhibited the **[[ ]]** from the **[[ ]]** statepoint database were identified and plotted versus pressure, local mass flux, and local quality for the VIPRE predictions in Figure 18 and for the LYNXT predictions in Figure 19. Both figures show the distribution of these non-conservative predictions to be similar between the two codes. **[[**

]], again indicating the VIPRE BHTP predictions align well with the test.

The examination of the P/M values for both codes, particularly for the statepoints with P/M values greater than the CHF Design Limit, has indicated that VIPRE produces behavior that is consistent with the LYNXT results used to establish the BHTP CHF correlation.

VIPRE-APS data was used as the basis for this assessment. The response to BHTP RAI-08 provides justification that VIPRE-APS and VIPRE-W produce equivalent results; therefore, this assessment is also applicable to VIPRE-W.

Table 10

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Figure 18

[[

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Figure 19

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BHTP RAI-07

Model Trends For both VIPRE-W and VIPRE-APS, provide plots demonstrating that the error in the BHTP (measured-to-predicted or predicted-to-measured) versus its key input parameters (e.g., pressure, local mass flux, local quality, shape factor) is consistent over the entire application domain and is adequately quantified by the 95/95 limit applied to BHTP.	
Goal	G3.3.3 – Model Trends
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 BHTP RAI-07

For VIPRE-W and VIPRE-01, Figures 20 through 24 illustrate that there are no observable trends or biases in the P/M ratios as a function, respectively, of pressure, local mass flux, local quality, inlet enthalpy, and “F-axial” shape factor for the **[[ ]]** data point set. The BHTP CHF 95/95 design limit **[[ ]]** is also displayed in the plots. The lack of observable trends in the plots supports the validity of the 95/95 limit over the entire application domain. **[[ ]]**, which led to the **[[ ]]** data point being selected to provide the 95/95 limit as discussed in Reference 2.

Figure 20

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Figure 21

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Figure 22

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Figure 23

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Figure 24

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BHTP RAI-08

Differences between VIPRE-APS and VIPRE-W	
Provide an explanation for the differences between VIPRE-APS and VIPRE-W predictions of the BHTP CHF data given in Figure 5-3 of the LAR dated July 6, 2018. While most of the predictions are nearly identical, a small number of the data points have differences, which are larger than anticipated and would indicate differences between VIPRE-APS and VIPRE-W.	
Goal	G3.3.3 – Model Trends
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 BHTP RAI-08

The benchmark between VIPRE-W/BHTP and VIPRE-APS/BHTP was re-performed in response to an APS Condition Report (19-09079) described in the Enclosure. The new results are shown in Figure 25, and in Tables 11 and 12. The data points in question now show excellent agreement. Plots of the Predicted VIPRE-APS/BHTP and Predicted VIPRE-W/BHTP DNB heat flux data compared to the Measured DNB heat flux data are shown in Figures 26 and 27, respectively.

The overall average ratio of the predicted VIPRE-W/BHTP to the predicted VIPRE-APS/BHTP is **[[ ]]** and the overall standard deviation of the predicted VIPRE-W/BHTP to the predicted VIPRE-APS/BHTP is **[[ ]]**. Therefore, VIPRE-W and VIPRE-APS produce equivalent results when implementing BHTP.

Plots of the P/M frequency distribution for VIPRE-APS and VIPRE-W are shown in Figures 28 and 29, respectively. The normal distribution curve is superimposed onto the frequency distributions for comparison.

In addition, as addressed in the response to BHTP RAI-07, no significant trends or biases are observed in plots of the P/M ratio as a function of pressure, local mass flux, local quality, inlet enthalpy, and “F-axial” shape factor for the data point set.

The information provided in this RAI response replaces the following items from the License Amendment Request:

- Figure 5-1 of Attachment 10
- Figure 5-2 of Attachment 10
- Figure 5-3 of Attachment 10
- Standard deviation between VIPRE-W/BHTP and VIPRE-01/BHTP in Section 5.4.2 of Attachment 10

Table 11

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Table 12

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Figure 25

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Figure 26

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Figure 27

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Figure 28

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Figure 29

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BHTP RAI-09

VIPRE-W Modeling Options	
Justify the new modeling options given in Table 5-7 of the LAR dated July 6, 2018, when applying BHTP and discuss how the originally approved modeling options will be applied when using another CHF model.	
Goal	G3.3.3 – Model Trends
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 BHTP RAI-09

The response to BHTP RAI-09 is provided in the three following sections:

1. A listing of the deviations from WCAP-14565-P-A/WCAP-15306-NP-A (Reference 17) required to model the CE16HTP fuel type and BHTP CHF correlation in VIPRE-W.

   Modeling of the BHTP CHF correlation and the CE16HTP thermal-hydraulic analyses with VIPRE-W or VIPRE-APS requires deviations from portions of Topical Report WCAP-14565-P-A/WCAP-15306-NP-A. Implementation of these exceptions will ensure modeling of the CE16HTP fuel type and usage of the BHTP CHF correlation are consistent with the methods utilized in the BHTP validation described in the LAR (Reference 9).

   APS intends to take the exceptions to WCAP-14565-P-A/WCAP-15306-NP-A as listed in the response to V-APS RAI-03 and apply them to both the VIPRE-W and VIPRE-APS modeling of the BHTP CHF Correlation and the CE16HTP related thermal-hydraulic safety analyses. The exceptions listed in the response to V-APS RAI-03 were demonstrated to be applicable in the benchmark and sensitivity analyses done to support the validation of the CE16HTP and BHTP CHF correlation with VIPRE-W.

2. A description of the new CE16HTP and BHTP CHF correlation VIPRE-W required modeling options.

   Table 5-7 of Attachment 10 of the LAR (Reference 9) contains modeling and closure options required for the consistent implementation of CE16HTP and the BHTP CHF correlation that are not directly discussed in Reference 17. These modeling and closure options are the same as those listed on Table 5-5 of Attachment 10 of the LAR as applied to the use of VIPRE-APS for the implementation of the CE16HTP fuel type and BHTP CHF correlation. As such, the response to V-APS RAI-05 is also applicable to the VIPRE-W modeling of the CE16HTP fuel product and BHTP CHF correlation and therefore constitutes the response to Part 2 of this RAI. The model and closure options listed in the response to V-APS RAI-05 were demonstrated to be applicable in the benchmark

and sensitivity analyses performed to support the validation of the CE16HTP and BHTP CHF correlation with VIPRE-W.

3. A discussion of how the originally approved modeling options will be utilized with other CHF models.

   Other CHF correlations that have been previously approved for APS will be implemented with the modeling options consistent with the original approvals. For example, the ABB-NV CHF and WSSV correlations for use with Westinghouse CE16STD and CE16NGF fuel will be implemented with modeling options consistent with Reference 17 as modified by the approval of CE16NGF in Reference 7. The sole exception is the use of the CE-1 CHF correlation to evaluate CE16HTP fuel below the first HTP grid as discussed in the response to CHF RAI-01. In this case, the CE-1 CHF correlation will utilize the modeling options consistent with the BHTP correlation as defined in the response to V-APS RAI-05.

   Additionally, Palo Verde intends to maintain the exceptions to WCAP-14565-P-A/WCAP-15306-NP-A discussed in the response to V-APS RAI-03 when evaluating the Framatome CE16HTP and Westinghouse fuel products (CE16STD and CE16NGF) licensed for use at Palo Verde.

   Note that the Macbeth CHF model will be applied to Framatome CE16HTP fuel as discussed in Section 5.3.3 of the License Amendment Request and in the response to CHF RAI-02.

BHTP RAI-10

Same Evaluation Framework	
Confirm that future uses of the BHTP in both VIPRE-W and VIPRE-APS will be performed in a similar manner as was when generating the validation data presented in this LAR.	
Goal	G3.5.2 – Same Evaluation Framework
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 BHTP RAI-10

The Framatome BHTP CHF correlation has been implemented in both the VIPRE-APS and VIPRE-W thermal-hydraulic analysis computer codes. Future uses of the BHTP CHF correlation, independent of the version of the VIPRE code (VIPRE-W or VIPRE-APS), will be performed in a manner similar to that utilized in the preparation of the validation data presented in the LAR (Reference 9) and RAI responses.

CHF RAI-01

CHF Model Below First HTP™ Grid Spacer	
Provide justification for the conclusion that no CHF model is needed to predict the DNBR behavior below the first HTP™ grid spacer.	
Associated Section	Modeling of HTP™ fuel below the first HTP™ grid spacer
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-01

Clarification: The Framatome HTP design contains a single High Mechanical Performance grid just above the lower end fitting. This grid is referred to as Grid 0 in APS nomenclature. This response considers the first HTP grid as "Grid 1" or the bottom grid. In the VIPRE models, once the first HTP grid is reached, the BHTP CHF correlation applies. The question here is what CHF correlation is used before the first HTP grid is reached.

The CE-1 CHF correlation will be conservatively used for DNB analysis in the region below the first HTP spacer grid. Framatome has analyzed the CE-1 CHF correlation and determined that it is conservative compared to the BWU-N CHF correlation licensed for this non-mixing vane region of the CE16HTP fuel design.

Automation of the CETOP/VIPRE Benchmarking process and other thermal-hydraulic analyses would require significant modification to model both the CE-1 CHF correlation below the first HTP spacer grid and the BHTP CHF correlation for all other locations. APS has analyzed past CETOP/TORC and CETOP/VIPRE analyses and determined that the DNBR limiting location has always been well above the first spacer grid in the CE 16x16 fuel design due to Technical Specification 3.2.5 Axial Shape Index restrictions in COLSS/CPC monitoring and protection systems.

For this reason, the APS thermal-hydraulic codes will continue to utilize the BHTP CHF correlation for the CE16HTP fuel type in the span below the first HTP spacer grid. Should cycle-specific analyses determine that the location of the limiting DNBR occurs below the first spacer grid, further analysis will be performed using the conservative CE-1 CHF correlation in the region below the first HTP spacer grid.

<u>Supporting Information</u>

Since the area of concern is the bottom grid span, a bottom peaked shape just below the allowable ASI range is needed. A positive 0.210 ASI power shape, which would be just outside the COLSS Operable limit of positive 0.2 ASI (as specified in the COLR per Technical Specification 3.2.5), was evaluated. A positive 0.601 ASI power shape, which would be

outside the normal operating limit of positive 0.5 ASI (per UFSAR Table 7.2-1A) was evaluated.

The positive 0.210 and positive 0.601 ASI power shapes, along with the location of the first HTP grid **[[ ]]**, are shown in Figure 30. This figure shows that the peak power location occurs above the first HTP grid; therefore, the point of minimum DNBR will occur above the first HTP grid. The point of minimum DNBR cannot occur at a point below the first HTP grid due to the ASI limits in COLSS and CPC.

To specifically examine DNBR performance below the first spacer grid, two test cases were run using the most extreme positive 0.601 axial shape, first using the BHTP CHF correlation, then with the CE-1 CHF Correlation. The results are shown in Figures 31 and 32.

For each of the test cases, the location of minimum DNBR is at **[[**

]].

The test cases show that CE-1 is more conservative compared to BHTP as expected. However, even for the most bottom-peaked axial shape, the point of minimum DNBR occurs **[[**

]]. Test Case 1 is where CE-1 comes closest to the MDNBR point. This case is nominal conditions, but power is driven up to 143% nominal to reach a DNBR point near the design limit DNBR for BHTP. Both the power and axial shape are extreme values that would never be encountered in actual plant operation, and even in those extreme conditions CE-1 below the first HTP spacer grid still has significant margin to the point of minimum DNBR.

<u>Conclusion</u>

The preceding discussion has determined that normal DNB analysis need not consider the point of minimum DNBR occurring in the bottom HTP grid, as ASI limits ensure that the point of minimum DNBR occurs well above the first HTP grid.

Framatome was asked to evaluate the CE-1 CHF correlation should a DNB analysis below the first grid in HTP fuel be desired. Framatome concluded that:

> **[[**
>
> **]]**

Therefore, use of the CE-1 CHF correlation for the bottom grid span (below the first HTP grid) is conservative and acceptable. It is not necessary to install either the BWU-N or XNB CHF correlations in T-H codes for use in PVNGS analyses. Furthermore, ASI limits preclude the point of minimum DNBR from occurring in the region below the first HTP grid.

Figure 30

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Figure 31

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Figure 32

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CHF RAI-02

Use of Macbeth at Low Pressures	
Provide a reference for Macbeth. The NRC staff is aware of an issue where older CHF models may under predict the CHF in the hot channel at low pressures. This is believed to be related to the fact that many of the older CHF correlations were based on tube data, and that at low pressures and low mass fluxes, the coolant is not kept inside the hot channel of an open lattice bundle as it would be in a tube. Justify the continued use of Macbeth given this new information.	
Associated Section	Modeling of HTP™ fuel below the first HTP™ grid spacer
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-02

Reference for Macbeth

PVNGS Units 1, 2, and 3 are approved to perform DNB evaluations of Westinghouse Standard (CE16STD) and Next Generation Fuel (CE16NGF) for low pressure events such as the post-trip steam line break event using the Macbeth CHF correlation. The Macbeth CHF correlation is used when the primary CHF correlation (i.e., CE-1, ABB-NV, or WSSV) is not applicable because the Reactor Coolant System pressure or flow is outside the primary CHF correlation range of applicability. This approval is documented in a series of Safety Evaluations and letters exchanged between the NRC, the System 80 NSSS Vendor (i.e., Combustion Engineering), and APS, as documented below.

The Macbeth CHF correlation coded into the HRISE code is the same Macbeth CHF correlation approved in July 1983 for use in the steam line break analysis for Calvert Cliffs Nuclear Power Plant (CCNPP) Units 1 and 2 (Reference 25).

The Macbeth CHF correlation is derived with rod bundle data with vertical upflow, and is calculated as a function of mass flux, inlet subcooling, system pressure, and heated diameter. The Safety Evaluation for CCNPP acknowledged a May 1983 letter (Reference 26) providing support for the ranges of applicability and a DNBR limit of 1.30. This May 1983 letter cites a 1965-66 Macbeth paper (Reference 27) as the source of the Macbeth CHF correlation, and it cites a 1964 Macbeth paper (Reference 28) as the source of the Macbeth CHF correlation ranges of applicability.

The 1965-66 Macbeth paper presents the Macbeth CHF correlation based on rod bundle data with vertical upflow as its Equation 17.22. The 1964 Macbeth paper presents the same Macbeth CHF correlation based on rod bundle data with vertical upflow as its Equation 15. Of note is that the 1965-66 Macbeth paper is cited in Paragraph I.C.4.b(4) of 10 CFR Part

50 Appendix K, ECCS *Evaluation Models*, as providing an acceptable steady-state CHF correlation for use in LOCA transients.

Justification for Use of Macbeth

As described above, the Macbeth CHF correlation coded into the HRISE code is based on rod bundle data with vertical upflow. As such, the Macbeth CHF correlation coded into the HRISE code is not subject to the issue where older CHF models based on tube data may underpredict the CHF in the hot channel at low pressures. Therefore, the continued use of the Macbeth CHF correlation coded into the HRISE code is justified.

CHF RAI-03

Use of WLOP CHF Correlation	
Justify the use of the Westinghouse Low-Pressure (WLOP) Correlation for use on fuel types of than those fuel types it initially validated, specifically CE16 HTP™ fuel.	
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-03

Palo Verde has decided to withdraw the request that the NRC approve the use of the Westinghouse WLOP CHF correlation with Framatome CE16HTP fuel. This request was made in Section 5.3.3.2 of Attachment 10 of the LAR (Reference 9). Consistent with the APS response to CHF RAI-03, the Macbeth correlation will be used for Framatome fuel.

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) **[[**

]] as addressed in the response to SET RAI-01. The detailed implementation documentation consists of:

- Code Software Quality Assurance (SQA) documentation
- Basedeck analysis
- **[[ ]]**
- **[[ ]]**
- 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

Section 5.4.3 of Attachment 10 of the LAR (Reference 9) provides the details of the planned implementation process, including the NRC notification requirements. The requirements specified in Section 5.4.3 of the LAR are modified herein to address the RAI:

The key elements of the process include the following:

- Notification of Qualification – To document APS's qualification to implement combinations of approved codes and methods to perform safety-related thermal-hydraulic analyses, APS shall docket a notification letter with NRC at least six (6) months before the planned startup of a cycle using the combination for safety related work. Any such notification letter shall explain how APS complied with the elements of this process, and shall make available any pertinent supporting data or information.
- Eligibility – Only those thermal-hydraulic codes and CHF correlations that have been previously approved by NRC for use at Palo Verde are eligible for this process. Eligibility does not extend to codes or correlations that have been accepted as part of another plant's licensing basis, but which have not yet been accepted as part of the Palo Verde licensing basis. No Technical Specification changes will be required, as this process may only be used if both the code and correlation are already specified as COLR methods in Palo Verde Technical Specification 5.6.5.b.
- Software Quality Assurance – Software modification practices including but not limited to verification and validation practices, shall conform to the Palo Verde Quality Assurance Program Description (QAPD), which was approved by NRC in a Safety Evaluation dated July 22, 2016 (Reference 29). With respect to SQA, the Palo Verde QAPD invokes the design control and test control provisions of the ASME NQA-1-2008 standard and NQA-1a-2009 addenda.
- Application Procedures – For each new combination of computer code and CHF correlation, Palo Verde application procedures (for example, Safety Analysis Basis Documents) will address pertinent limitations and constraints associated with the selection of specific modeling assumptions and input values, including but not limited to two-phase flow models and correlations, heat transfer correlations, determination of DNBR limits, turbulent mixing coefficients, hot channel factors, and grid loss coefficients, that are applicable for the fuel types approved for use at PVNGS. Application procedures shall include proper controls to preclude misapplication of code and CHF correlation combinations.
- Training and Qualification – Palo Verde personnel have previously demonstrated the capability to perform non-LOCA safety analyses, as evidenced by the NRC Safety Evaluation dated June 14, 1993 (Reference 30), as well as during subsequent interactions with NRC (e.g., power uprate and steam generator replacement, and this licensing application). Palo Verde personnel shall receive fuel vendor technology transfer training and computer code-related training as needed to acquire and maintain technical competence.

- Benchmark and Comparison Calculations – Benchmark and comparison calculations will be performed to ensure proper implementation of CHF correlations into different computer codes. Such calculations shall verify and validate each new combination in accordance with the method used in either Reference 9 Section 5.4.1 (comparison to CHF test data) or Reference 9 Section 5.4.2 (comparison to another approved code using a CHF correlation, where the NRC specifically reviewed and approved that code and CHF correlation combination for use at PVNGS). Under no circumstances will application ranges for CHF correlations (for example, temperature, pressure, quality, spacer grid span length) be extrapolated beyond the NRC approved range of the code or CHF correlation.
- Approval Process – Approval for the proposed CHF and code combination shall be based on an NRC audit of the implementation, and if acceptable, issuance of an NRC approval letter.
- The PVNGS UFSAR will be updated to explicitly identify any correlation and code combinations approved under this process.

The following are the specific code and CHF combinations for which this process may be used:

a. ABB-NV into VIPRE-01 / VIPRE-APS

b. WSSV into VIPRE-01 / VIPRE-APS

c. CE-1 into VIPRE-01 / VIPRE-APS (or qualification of the existing VIPRE-01 / VIPRE-APS implementation)

d. CE-1 into VIPRE-W (or qualification of the existing VIPRE-W implementation)

e. BHTP into TORC

The future addition of ABB-NV and WSSV into VIPRE-01 / VIPRE-APS supports the APS long-term goal for using fuel vendor independent computer codes. As described in Section 5.4.2 of the License Amendment Request (Reference 9) and the response to BHTP RAI-08, VIPRE-01 / VIPRE-APS and VIPRE-W are equivalent codes, giving essentially identical results for identical inputs with the BHTP critical heat flux correlation. Based on the excellent VIPRE-APS/BHTP to VIPRE-W/BHTP agreement, and the common original code basis for both VIPRE-01 / VIPRE-APS and VIPRE-W, similarly consistent results are expected when ABB-NV and WSSV are implemented in VIPRE-01 / VIPRE-APS.

Currently, the CE-1 CHF correlation implementations in VIPRE-01 / VIPRE-APS and VIPRE-W are not NRC approved. As discussed in the response to CHF RAI-01, the VIPRE CE-1 CHF correlation will be conservatively used for DNB analysis in the region below the first HTP spacer grid. If APS moves to a fully VIPRE based setpoint process, there may need to be a need to fully utilize CE-1 with VIPRE, due to the plant Core Operating Limits Supervisory System and Core Protection Calculator System both continuing to use the CE-1 CHF correlation.

For completeness, BHTP may be added to TORC, to support re-performance of any TORC based analyses if required.

The validation process will be as specified in Section 5.4.3, "Process for Implementation of CHF Correlations with Other T-H Codes," of Attachment 10 of the LAR (Reference 9), and as demonstrated for CETOP-D/BHTP. The NRC process will be done as per Section 5.4.3 of the LAR, as modified by the response to this RAI.

10 CFR 50.36 compliance will be maintained as both the CHF correlation and computer codes to be used would be listed in the Administrative Controls section of the Technical Specifications. This is reflected in the proposed Technical Specification changes submitted as Attachments 2 and 3 of the LAR (Reference 9), as modified by the response to RAI SNPB RAI-22. Specifically for each of the proposed future code and critical heat flux combinations, the TS 5.6.5 references are summarized in Table 13.

Table 13

CHF Correlation and Thermal-Hydraulic Computer Code Combinations

Combination	CHF Correlation Topical Report Technical Specification	Computer Code Topical Report Technical Specification
a	ABB-NV CENPD-387-P-A TS 5.6.5.b.24	VIPRE-01 EPRI-NP-2511-CCM-A proposed TS 5.6.5.b.31
b	WSSV WCAP-16523-P-A TS 5.6.5.b.25	VIPRE-01 EPRI-NP-2511-CCM-A proposed TS 5.6.5.b.31
c	CE-1 NUREG-0852 SER (CENPD-162 and CENPD-207 as referenced in 4.4.2.1) and CEN-356(V)-P-A TS 5.6.5.b.3 and 5.6.5.b.4	VIPRE-01 EPRI-NP-2511-CCM-A proposed TS 5.6.5.b.31
d	CE-1 NUREG-0852 SER (CENPD-162 and CENPD-207 as referenced in 4.4.2.1) and CEN-356(V)-P-A TS 5.6.5.b.3 and 5.6.5.b.4	VIPRE-W WCAP-14565-P-A TS 5.6.5.b.23
e	BHTP BAW-10241(P)(A) proposed TS 5.6.5.b.30	TORC CENPD-161-P-A TS 5.6.5.b.15

MIX RAI-01

Mixed Core	
APS has described a mixed core approach, which seems different from that described in the VIPRE-W topical report. APS should provide documentation and justification for its mixed core approach, including specification of any previous approvals. This approach should specify how the impact to DNBR behavior is calculated when a core contains multiple fuel types in a transition core (i.e., not a mixed fresh batch).	
Associated Section	Mixed Core Methodology
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 MIX RAI-01

Section 4.1.2 of WCAP-14565-P-A (Reference 31) discusses a mixed core approach that involves the introduction of a dedicated mixed core penalty, the value of which is calculated in a manner consistent with that used for VANTAGE 5 fuel and is based on the fraction of each fuel type present in the core. The reason cited by WCAP-14565-P-A for the penalty is to account for uncertainties in local flow redistributions.

The proposed APS approach to mixed cores including Framatome CE16HTP fuel differs as it captures the local flow redistribution phenomenon in a more explicit manner than that described in WCAP-14565-P-A, precluding the need for a dedicated mixed core penalty. Additionally, an inherent conservatism in the proposed APS mixed core approach is highlighted in this response. Though APS is no longer requesting mixed fresh cores (as per the response to MIX RAI-02), the same logic outlined in this RAI would apply to that situation. For that reason this response uses the term "mixed core."

The explicit analysis of the mixed core to preclude the need for a mixed core penalty relies on the capability of the code and methodology to capture relevant thermal-hydraulic effects. As discussed in WCAP-14565-P-A, the specific phenomenon of interest is potential local flow redistributions caused by dissimilar fuel types adjacent to one another. Other phenomenon that could affect DNBR are already captured explicitly; assembly-specific radial power distributions are already captured explicitly in the neutronics analysis, differences in grid loses are captured explicitly through application of assembly-specific loss coefficients, etc.

There are two key aspects regarding potential local flow redistribution: effects on flow distribution at the core inlet, and effects along the axial length. As stated in Section 5.5.1 of Attachment 10 of the LAR, mixed core reloads that contain CE16HTP fuel and another licensed fuel type will perform an analysis to calculate a cycle-specific inlet flow distribution.

This inlet flow distribution serves as input to downstream analysis, which uses a detailed VIPRE model to analyze the cycle-specific mixed core.

With regards to crossflow above the inlet, VIPRE models crossflow to support its use in a mixed core reload. Two types of crossflow are of importance: diversion crossflow and turbulent crossflow. Diversion crossflow refers to a laterally convective flow due to the pressure difference between channels. This is essentially treated by the lateral momentum equation and driven by assembly specific grid-loss coefficients. Turbulent crossflow does not result in a net mass transfer between channels but is a medium for energy and momentum transfer. These effects are usually contained in the mixture momentum and energy equations as correction terms with user defined mixing coefficients.

With regards to diversion crossflow, the VIPRE subchannel modeling allows a region of fluid flow to be described by a number of channels of various sizes and shapes, defined by their flow area, wetted perimeter, and gap size. For each axial node, conservation equations of mass, axial, and lateral momentum and energy are solved for the fluid enthalpy, axial flow rate, lateral flow per unit length, and momentum pressure drop. For the lateral momentum component, the code assumes that the flow direction is determined by the gap orientation and that the crossflow loses its identity away from the gap. The equation is therefore integrated only in the region of the influence of each gap. The generic modeling of the lateral momentum equation allows the code to react to differing conditions between adjacent channels driving lateral mass and energy exchange.

Turbulent crossflow mixing is primarily significant at the subchannel level, since turbulence around the hot pin will affect the coolant temperature at that location, and it is less significant at the assembly level. Turbulent crossflow mixing in VIPRE is a function of gap size, (calculated) average axial mass velocity, and a mixing coefficient. The mixing coefficients are discussed in the response to V-APS RAI-05.

There is no empirical data that directly supports the VIPRE crossflow models as applied to mixed CE16HTP/CE16NGF/CE16STD cores, but VIPRE has been tested and benchmarked in situations in which the crossflow model played a major role. For example, Section 2.2 of Volume 4 of the VIPRE manual (Reference 24) describes a 7x7 rod array test at Battelle with flow blockages to simulate clad ballooning. A VIPRE model was created to mimic the tests and the VIPRE-predicted results were compared to the measured data from the Battelle tests. Overall, the predictions were in very good agreement with the data. The code correctly predicted the velocity decrease just before the blockage, the expected acceleration in the blockage throat, the expansion loss at the blockage exit, and the subsequent downstream recovery.

The last component of ensuring that all necessary mixed core thermal-hydraulic effects are captured is the usage of a sufficiently detailed model. The APS mixed core model is a detailed VIPRE model that models the hot assembly in full subchannel detail, and surrounding assemblies in quarter-assembly detail. This model is consistent with historic APS usage of TORC, and VIPRE-W as approved for CE16NGF fuel (Reference 7). The mixed core effects are evaluated on a cycle-specific basis with this detailed VIPRE model.

The conclusion that can be drawn from the above is that APS's explicit calculation of cycle-specific inlet flow distribution, combined with VIPRE's adequate crossflow models with appropriately selected coefficients and model geometry, is sufficient to model mixed core designs without a dedicated mixed core penalty such as that described in WCAP-14565-P-A.

In addition, it is important to highlight an inherent conservatism that exists in the proposed APS mixed core approach. As discussed in the APS response to NRC Question 3 (Reference 8), the highest (i.e., more conservative) DNBR analytical limit for any fuel type identified as potentially limiting will be used. For example, in a core in which both CE16HTP and CE16STD have been identified as limiting candidates, the higher CE16STD limit of 1.34 would be applied rather than the lower CE16HTP limit of 1.27. When in the presence of CE16STD, this has the effect of penalizing the CE16HTP assembly.

MIX RAI-02

Mixed Core with Mixed Fresh Batch	
APS stated its intention to use its mixed core methodology for mixed fresh batches. APS should provide documentation and justification for this seemingly new application of a mixed core approach, including specification of any previous approvals. This approach should specify how the impact to DNBR behavior would be impacted by mixed fresh batches and would be calculated with mixed fresh batches. Additionally, APS should demonstrate that this approach is within the capabilities of the code and methodology.	
Associated Section	Mixed Core Methodology
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 MIX RAI-02

As described in the response to MIX RAI-01, the mixed core reload methodology is valid not only for mixed transition cores but also mixed fresh batches as it explicitly models all fuel types. As described in the response to SNPB RAI-25, APS is no longer requesting approval for mixed fresh fuel core designs; therefore, no additional discussion of this scenario is required.

SET RAI-01

Setpoints Methodology	
Describe how the BHTP CHF model will be implemented in the COLSS/CPCS setpoint methodology. This description should contain a description or reference to the currently approved process, as well as an explanation of the statistical methodology used.	
Associated Section	Application of COLSS/CPCS to Generate Setpoints for CE16 HTP™ fuel
Associated Regulations and Guidance	10 CFR Part 50, Appendix A, GDC 10; 10 CFR 50.36; and 10 CFR 50.34

Response to SET RAI-01

With regards to the Setpoints COLSS\CPC Overall Uncertainty Analysis (OUA), the BHTP CHF model will be implemented in the CETOP-D code. The Setpoints OUA also requires Framatome CE16HTP fuel input from the Thermal-Hydraulic (T-H) design analyses that **[[ ]]**. Those T-H analyses also require use of the BHTP CHF correlation within the T-H design code(s).

The implementation of CE16HTP fuel does not involve any CHF correlation change to the online plant protection (CPC) and monitoring (COLSS) systems. The plant COLSS and CPCs will continue to use their existing CE-1 CHF correlation. This is the same approach used for the implementation of CE16NGF.

The current setpoints methodology is addressed in WCAP-16500-P-A, Supplement 1, Revision 1 (Reference 5). **[[**

]] what COLSS and CPCs use (i.e., the CE-1 CHF correlation) **[[ ]]**.

Modifying the existing CETOP-D code to use the BHTP CHF correlation **[[ ]]**. Implementation of BHTP in the CETOP-D code is discussed in the response to CHF RAI-04.

[[ ]] is discussed in the response to SNPB RAI-3, and depicted in Figure 1.

TH RAI-01

Pressure Drop	
Provide a thermal hydraulic description of CE16STD, CE16NGF, and CE16 HTP™ fuel that contains the number of grid spacers, and the elevation and pressure drop of the spacer, including the bottom grid.	
Associated Section	Thermal-hydraulic Compatibility
Associated Regulations and Guidance	10 CFR Part 50, Appendix A, GDC 10; 10 CFR 50.36; and 10 CFR 50.34

Response to TH RAI-01

Table 14 provides a tabular comparison of the spacer grid loss coefficients associated with each of the three reactor fuel designs.

Figure 33 provides an illustrative comparison of the three reactor fuel designs (CE16STD, CE16NGF, and CE16HTP). The comparison depicts the spacer grid locations along the fuel rod. Note that spacer grid locations are provided based on the mid-grid elevation taken from the beginning of the heated length (BOHL) or bottom of the fuel stack.

Table 14

Comparison of Spacer Grid Loss Coefficients

[[

]]

Figure 33

CE16STD, CE16NGF, and CE16HTP Fuel Types

[[

]]

TH RAI-02

Scram Time Testing	
Confirm that scram time testing will validate the similarity between Westinghouse and Framatome fuel and will demonstrate that the control element assembly drop time will not be challenged due to the fuel transition.	
Associated Section	Thermal-hydraulic Compatibility
Associated Regulations and Guidance	10 CFR Part 50, Appendix A, GDC 10; 10 CFR 50.36; and 10 CFR 50.34

Response to TH RAI-02

The response to SNPB RAI-9 addresses the similarity between Westinghouse and Framatome designs and shows that the Control Element Assembly (CEA) drop time will not be challenged due to the fuel transition. The eight (8) Lead Fuel Assemblies irradiated at Palo Verde were installed in 12-finger and 4-finger CEA locations with no operational issues. During unit shutdowns and scram time testing, the 12-finger CEAs successfully inserted into adjacent Framatome and Westinghouse fuel assemblies.

CEA drop times are validated prior to reactor criticality and after each removal of the reactor head to ensure compliance with the maximum allowed CEA drop time per Technical Specification Surveillance Requirement 3.1.5.5.

TH RAI-03

Guide Tube Heating	
Provide details of the analysis which confirms there will be no boiling in the guide tubes.	
Associated Section	Thermal-hydraulic Compatibility
Associated Regulations and Guidance	10 CFR Part 50, Appendix A, GDC 10; 10 CFR 50.36; and 10 CFR 50.34

Response to TH RAI-03

The Guide Tube and Control Element Assembly heating rates were calculated for the Framatome CE16HTP fuel assembly design in a vendor analysis. The analysis considered potential absorber materials, used the limiting radial power distribution, and limiting axial power shape. The analysis also modeled a partially rodded assembly under the Power Dependent Insertion Limit (PDIL) at hot full power condition, even though during steady-state operation the assemblies with rods inserted to the PDIL will not be the highest power peaked assemblies in the core. To ensure the results remain bounding for future operation, the reactor core models include representative transition cycles and a full-core Framatome CE16HTP fuel cycle. The heating rates at 80 GWd/MTU were conservatively used to bound the maximum pin burnup of 62 GWd/MTU for which Framatome is licensed.

Using these heating rates as input, the Framatome guide tube heating thermal hydraulic analysis determined that a Control Rod LHGR of less than or equal to **[[ ]]** precludes boiling within the CE16HTP guide tube. Based on a Relative Power Density of 1.42, Framatome used COBRA-FLX to calculate a Control Rod LHGR of **[[ ]]**, which precludes boiling within the guide tube during normal, steady-state operation. This analysis represents a bounding case with significant discretionary conservatism.

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33. BAW-10231P-A, Revision 1, *COPERNIC Fuel Rod Design Computer Code*, January 2004

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35. Letter from P. H. Buckberg (NRC) to M. Nazar (NextEra Energy), *St. Lucie Plant, Unit No. 2 – Issuance of Amendment Regarding Transitioning to AREVA Fuel (CAC No. MF5495)*. April 19, 2016. [NRC ADAMS Accession No. ML16063A121 (non-proprietary) and ML16063A120 (proprietary)].

ENCLOSURE
ATTACHMENT 8

Framatome ANP-3639P, Revision 1Q1NP, Revision 0
Palo Verde Units 1, 2 and 3
Realistic Large Break LOCA Summary Report
NRC RAI Responses

[NON-PROPRIETARY VERSION]

framatome

Palo Verde Units 1, 2 and 3 Realistic Large Break LOCA Summary Report NRC RAI responses

ANP-3639P
Revision 1Q1NP
Revision 0

Licensing Report

September 2019

Nature of Changes

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

Contents

Page

1.0 SUMMARY 1-1
2.0 RAI 12 2-1
3.0 RAI 20 3-1
4.0 CONDITION REPORTS ON ANALYTICAL MODEL 4-1
5.0 REFERENCES 5-1

List of Tables

Table 2-1: Maximum Local Oxidation 2-10

List of Figures

Figure 2-1: PCT Trace Comparison (Case 74 vs. Case 131) 2-4
Figure 2-2: Total Reactor Vessel Mass (Case 74 vs. Case 131) 2-5
Figure 2-3: Total SIT Flow (Case 74 vs. Case 131) 2-6
Figure 2-4: Hot Assembly Collapsed Liquid Level (Case 74 vs. Case 131) 2-7
Figure 2-5: Maximum Local Oxidation versus Time of PCT 2-9

Nomenclature

Acronym	**Definition**
CP	Cathcart-Pawel
CR	Condition Report
CWO	Core-Wide Oxidation
ECR	Equivalent Cladding Reacted
LOCA	Loss of Coolant Accident
MLO	Maximum Local Oxidation
NRC	U. S. Nuclear Regulatory Commission
PCT	Peak Clad Temperature
RAI	Request for Additional Information
RLBLOCA	Realistic Large Break LOCA

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1.0 SUMMARY

This report contains the responses to the request for additional information (RAI) numbers 12 and 20 as asked by the NRC for the Palo Verde Realistic Large Break LOCA Analyses. Information regarding two Condition Reports (CRs) that were written after the Palo Verde RLBLOCA analyses were performed is also discussed in this report.

2.0 RAI 12

Request:

The NRC staff's review of the calculated results in ANP-3639P, "Palo Verde Units 1, 2, and 3 Realistic Large Break LOCA Summary Report," identified certain datapoints that appear to be outliers with respect to the main body of data. Information available in the submittal does not provide a reasonable physical explanation for the predicted behavior in these outlying cases. To provide reasonable assurance that implementation of the Realistic Large Break LOCA methodology at Palo Verde produces expected results when computing the figures of merit required for comparison against the acceptance criteria in 10 CFR 50.46(b), please.

(a) *Provide a physical explanation for the following outlying predictions:*

- *Figure 3-2, "PCT [Peak Cladding Temperature] versus PCT Time Scatter Plot," of ANP-3639 shows that the case that sets the statistical limit for peak cladding temperature is* [

  ]

- *Figure 3-4, "Maximum Local Oxidation versus PCT Scatter Plot," of ANP-3639 shows that* [

  ]

(b) *Provide the maximum local oxidation for each case shown in Table A-1, "Summary of Key Input and Output Parameters, Part 1," of ANP-3639P.*

Response (a, Part 1):

The limiting case with respect to PCT, Case 131 from Table 3-5 of ANP-3639, models a [

] These sampled parameters, in general, tend to lead to higher cladding temperatures. However, the sampled parameters listed above are not solely responsible for the PCT magnitude and timing. Rather, the result stems from the combination of all of the parameters and the Palo Verde plant design, which leads to a specific system and cladding temperature response.

As shown in Figure 3-6 of ANP-3639, Case 131 has three distinct heatups that correspond to the three major phases of an LBLOCA – blowdown, refill and reflood – and their associated phenomena and heat balances. The third cladding temperature increase during reflood is the largest for this case and results in the PCT of 1752°F at 260.6 seconds.

The SIT inventory is exhausted after 54 seconds. Thereafter, the only inventory addition to the RCS to counter the break flow is from pumped ECCS injection. Steam binding in the RCS loops and upper plenum retards the flow into the core and consequently, slows the quench front advancement. The slow quench front advancement coupled with the top skewed axial shape, burnup, and peaking factor, leads to the PCT occurring during the reflood phase of the event.

In order to better illustrate the discussion on the interconnectedness of the parameters and the system design, another case with a similar PCT has been selected for comparison. [

] Figure 2-1 presents a comparison of PCT independent of

elevation between Case 74 (early blowdown phase) and Case 131 (reflood phase). A comparison of the total reactor vessel mass between Cases 74 and 131 is shown in Figure 2-2. Figure 2-3 shows the comparison of total SIT flow. Figure 2-4 shows the comparison of hot assembly collapsed liquid level. Figure 2-1 clearly shows [

]

In conclusion, while Case 131 appears to be an outlier from the PCT vs. PCT time scatter plot, the inspection of the transient details shows that that the PCT prediction is an expected and reasonable response for this plant with the given case-specific event and plant characteristics.

Figure 2-1: PCT Trace Comparison (Case 74 vs. Case 131)

Figure 2-2: Total Reactor Vessel Mass (Case 74 vs. Case 131)

Figure 2-3: Total SIT Flow (Case 74 vs. Case 131)

Figure 2-4: Hot Assembly Collapsed Liquid Level (Case 74 vs. Case 131)

Response (a, Part 2):

The relationship between PCT and maximum local oxidation (MLO) is not direct. MLO is strongly influenced by both the magnitude of the cladding temperature and by the time at these temperatures. [ ] Metal water reaction is exponentially dependent on temperature, so for a given time at temperature, a higher cladding temperature would cause more oxidation. That is, the longer the cladding remains at elevated temperature, the greater the oxidation will be.

This relationship becomes apparent when looking into more detail at the PCT and the time of PCT. Figure 3-4 of ANP-3639 shows an MLO of 7.68% with a corresponding PCT of 1752°F for Case 131. In the same temperature range, Figure 3-4 shows an [ ] The difference between these cases is that the PCT time for Case 131 is 260.6 seconds, while [ ] Even though both PCTs are high, the earlier PCT time results in a lower MLO value because cladding temperature is not at elevated temperatures as long.

An alternate way to illustrate this point is by examining Figure 2-5, which shows MLO as a function of time of PCT. Case 131 has an MLO of 7.68% and a PCT time of 260.6 seconds. In the same time range, [ ] The difference between these cases is that the PCT for Case 131 is 1752°F while [ ] Even though both PCT times are later in the event, the lower PCT results in significantly lower MLO.

In conclusion, it is the combination of two conditions, high cladding temperatures and long time-at-temperature, which explain the high MLO cases shown in Figure 3-4 of ANP-3639P.

Figure 2-5: Maximum Local Oxidation versus Time of PCT

Response (b):

The MLO for each case shown in Table A-1 of ANP-3639P is presented in Table 2-1.

Table 2-1: Maximum Local Oxidation

[

]

3.0 RAI 20

Request:

Please clarify and justify the modeling of non-Framatome fuel in the large-break LOCA analysis in ANP-3639P.

In particular, ANP-3639P states that

> *In addition to the Framatome HTP™ fuel, the hydraulic characteristics of other fuel types that could be present in the core were considered.* [
>
>]

However, Limitation 4.3 associated with EMF-2103P-A states that the methodology is applicable to fuel with M5® cladding, and that application of the evaluation model to fuel with other types of cladding has not been reviewed. Furthermore, it is not clear how the fuel pellet and cladding thermal-mechanical behavior is modeled for non-Framatome fuel (1) as a function of burnup to support initialization of the LOCA calculation and (2) during the LOCA transient calculation.

Clarification and justification regarding the modeling of non-Framatome fuel in the large-break LOCA analysis is necessary to ensure that figures of merit are calculated in a representative or conservative manner in order to satisfy the acceptance criteria in 10 CFR 50.46(b) during operating cycles with multiple fuel types present in the reactor core.

Response:

With respect to the calculation of the fuel rod performance, the Palo Verde RLBLOCA analysis only models Framatome fuel with $M5_{Framatome}$ cladding. [

] using input and models for Framatome fuel. Similarly, the embedded fuel rod code models and cladding models in the S-RELAP5 calculation in ANP-3639P are specific to Framatome fuel properties and design. [

] The Palo Verde analysis as described in ANP-3639P, and quoted in the RAI, follows the recommended approach described in Appendix A of EMF-2103P-A, Rev. 3 (Reference 2, page-A-66).

Since only the fuel rod performance for Framatome fuel is calculated, the reported Peak Cladding Temperature (PCT) and MLO are only for Framatome fuel and are applicable and/or conservative for both a transition core and full Framatome fuel core scenarios. The reported Core Wide Oxidation (CWO), used to demonstrate compliance with the maximum hydrogen generation criterion, is applicable to both core scenarios as well because the transient oxidation is calculated universally with the Cathcart-Pawel (CP) correlation and differences between fuel rod performance, on a core-wide basis, are insignificant.

4.0 CONDITION REPORTS ON ANALYTICAL MODEL

There are two CRs, 2019-840 and 2019-1130, associated with the RLBLOCA calculations which were discovered and written after the Palo Verde RLBLOCA calculations in ANP-3639P (Reference 1) were performed. The CRs were evaluated against the Palo Verde RLBLOCA analysis and confirmed that the figure of merits reported in Reference 1 remain applicable and valid as the licensing bases for the Palo Verde RLBLOCA analysis. The two CRs are listed and described below.

[ ]

5.0 REFERENCES

1. ANP-3639P, Revision 1, *Palo Verde Units 1, 2 and 3 Realistic Large Break LOCA Summary Report*, May 2018.

2. EMF-2103(P)(A) Revision 3, *Realistic Large Break LOCA Methodology for Pressurized Water Reactors,* Framatome, June 2016.

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

ENCLOSURE
ATTACHMENT 9

Framatome ANP-3640Q1NP, Revision 0
Palo Verde Units 1, 2 and 3
Small Break LOCA Summary Report
NRC RAI Responses

[NON-PROPRIETARY VERSION]

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

ANP-3640Q1NP
Revision 0

Licensing Report

September 2019

Nature of Changes

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

Contents

Page

1.0 SUMMARY 1-1
2.0 RAI-13 2-1
3.0 RAI-15 3-1
4.0 RAI-18 4-1
5.0 RAI-19 5-1
6.0 FRAMATOME SUPPORTING MATERIAL 6-1
    6.1 RAI-14 6-1
    6.2 RAI-16 6-2
7.0 REFERENCES 7-1

List of Tables

Table 2-1: Summary of Palo Verde RCP Trip Delay Cold Leg SBLOCA Results......... 2-2

Table 2-2: Summary of Palo Verde RCP Trip Delay Hot Leg SBLOCA Results 2-2

Table 2-3: Delayed RCP Trip Analysis – Limiting Cases Sequence of Events, 5 Minute Delay 2-3

Table 6-1: Break Spectrum PCT and Event Timing 6-4

List of Figures

Figure 4-1: Steam Generators and Turbine Control Valve Mass Flow Rates 4-3
Figure 4-2: Steam Generators and Secondary Total Mass 4-4
Figure 4-3: Core Exit and Upper Plenum Void Fractions.. 4-6
Figure 6-1: [ ]....................................... 6-5
Figure 6-2: [ ]....................................... 6-6
Figure 6-3: [ ]....................................... 6-7
Figure 6-4: [ ]....................................... 6-8
Figure 6-5: [ ]....................................... 6-9
Figure 6-6: [ ]..................................... 6-10
Figure 6-7: [ ]..................................... 6-11
Figure 6-8: [ ]..................................... 6-12
Figure 6-9: [ ]..................................... 6-13
Figure 6-10: [ ]................................... 6-14
Figure 6-11: [ ]................................... 6-15
Figure 6-12: [ ].................................. 6-16
Figure 6-13: [ ]................................... 6-17
Figure 6-14: [ ]................................... 6-18
Figure 6-15: [ ]................................... 6-19
Figure 6-16: [ ]................................... 6-20
Figure 6-17: [ ]................................... 6-21
Figure 6-18: [ ]................................... 6-22
Figure 6-19: [ ]................................... 6-23
Figure 6-20: [ ]................................... 6-24
Figure 6-21: [ ]................................... 6-25
Figure 6-22: [ ]................................... 6-26
Figure 6-23: [ ]................................... 6-27
Figure 6-24: [ ]................................... 6-28
Figure 6-25: [ ]................................... 6-29

Nomenclature

Acronym	Definition
AFW	Auxiliary Feedwater
APS	Arizona Public Service
CWO	Core-Wide Oxidation
EOC	End-of-Cycle
HPSI	High Pressure Safety Injection
LAR	License Amendment Request
LPSI	Low Pressure Safety Injection
LOCA	Loss of Coolant Accident
LSC	Loop Seal Clearing
MLO	Maximum Local Oxidation
NPSH	Negative Pump Suction Head
NRC	Nuclear Regulatory Commission
PCT	Peak Cladding Temperature
PZR	Pressurizer
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
SBLOCA	Small Break Loss of Coolant Accident
SE	Safety Evaluation
SG	Steam Generator
SIAS	Safety Injection Actuation Signal
TCD	Thermal Conductivity Degradation

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1.0 SUMMARY

This report contains the responses to Request for Additional Information (RAI) numbers 13, 15, 18, and 19 as asked by the NRC for the Palo Verde Small Break LOCA Analyses. Information to support Arizona Public Service (APS) responses to RAIs 14 and 16 are also provided.

2.0 RAI-13

Request:

Section 4.3 of ANP-3640P, "Palo Verde Units 1, 2, and 3 Small Break LOCA [SBLOCA] Summary Report," describes delayed reactor coolant pump trip sensitivity studies. To ensure that these sensitivity studies are sufficient to provide confidence that the most severe postulated conditions consistent with 10 CFR 50.46 have been calculated, please provide the following information:

(a) *The results of the break sizes that were analyzed for the hot leg and cold leg sensitivity studies in tables similar to those provided in Table 4-1, "Summary of SBLOCA Break Spectrum Transient Results," of ANP-3640P.*

(b) *The results for the limiting break sizes for the cold leg and hot leg cases in a table similar to those provided in Table 4-2, "Sequence of Events for Break Spectrum (seconds)," of ANP-3640P.*

(c) *Discuss the modeling used for loop seal biasing in the hot leg and cold leg sensitivity studies and discuss if it was necessary to* [

] *for the studies.*

Response (a):

The results of the break sizes that were analyzed for hot and cold leg sensitivity studies are shown in Table 2-1 and Table 2-2.

Table 2-1: Summary of Palo Verde RCP Trip Delay Cold Leg SBLOCA Results

[ ]

Table 2-2: Summary of Palo Verde RCP Trip Delay Hot Leg SBLOCA Results

[ ]

[ ]

Response (b):

The sequence of events for the limiting break sizes for the cold and hot leg cases are presented in Table 2-3.

Table 2-3: Delayed RCP Trip Analysis – Limiting Cases Sequence of Events, 5 Minute Delay

Event	6 inch Cold Leg Break Time (sec)	6 inch Hot Leg Break Time (sec)
Break Opening	0	0
Low PZR Pressure Trip	20	31
SIAS Issued	20	31
Reactor Scram and Turbine Trip	22	32
Pressurizer pressure reached NPSH setpoint (1471.0 psia)	29	40
HPSI Flow: Loop 1A/1B/2A/2B (Broken)	50/50/50/50	61/61/61/61
Break Uncovery	250	284
Loop Seal Clearing: Loop 2B, Broken	266	285
Loop Seal Clearing: Loop 2A	265	285
Loop Seal Clearing: Loop 1A	265	296
Loop Seal Clearing: Loop 1B	265	296
Core Uncovery	278	302
RCP Trip	329	340
SIT Flow: Loop 1A/1B/2A/2B (Broken)	443/443/443/443	451/451/451/451
PCT Time	453	456
Approximate Core Quench	550	500
LPSI Flow: Loop 2A/2B (Broken)	-	1863
First Switch to HEM Break Model	-	-
AFW SG-1	-	-
AFW SG-2	-	-

Response (c):

3.0 RAI-15

Request:

[ ]

Response:

For the SBLOCA analysis, RODEX2-2A is used to determine the initial core and hot pin stored energy. [

]

The SBLOCA break spectrum includes break sizes up to 10% of the cold leg area. [

]

4.0 RAI-18

Request:

Several figures included in ANP-3640P that plot key parameters predicted for the limiting SBLOCA event display behavior that may appear non-physical. To assure that the predicted figures of merit for this event are calculated correctly such that reasonable assurance exists that the acceptance criteria in 10 CFR 50.46(b) have been satisfied, please provide justification that the following calculated behavior is reasonable:

(a) *Prior to 200 seconds into the event, the steam generator total mass in Figure 4-14, "Steam Generator Total Mass – 9.10 Inch Break," of ANP-3640P begins to vary as a function of time, despite Figures 4-11 (main feedwater flow), 4-12 (steam generator safety valve mass flow) and 4-13 (auxiliary feedwater flow) showing no appreciable changes in the mass flow into or out of the steam generators.*

(b) *After about 200 seconds into the event, the hot assembly mixture level in Figure 4-22, "Hot Assembly Mixture Level – 9.10 inch Break," of ANP-3640P appears to take several different, approximately constant, values across several different periods, most prominently between approximately 355 and 620 seconds; whereas it is not obvious why the hot assembly mixture level should remain approximately constant during this period at a number of distinct values, all of which are below the top of active fuel.*

Response (a):

The mass plots shown in Figure 4-14 of ANP-3640P are the total mass for each Steam Generator (SG) vessel (ANP-3640P, Figure 3-3: [ ]) on the secondary side and do not reflect any mass from the remaining secondary system components (i.e., steam lines and steam line components). Small changes in SG mass are expected from steam redistribution across the secondary path as a result of SGs response to key events during the transient. The overall significance of the mass

variations over time is relatively small, but appears to be more significant due to the scaling in the y-axis (4%). Variations on key points along the SG mass plots prior to 200 seconds in Figure 4-14 are explained below with respect to key transient events.

The first SG mass change, shown as a step-like increase in the early part of the transient (14 seconds), is due to a mass imbalance between feedwater and steam flow to the turbine. At the beginning of the transient the system is operating normally with the MFW injecting, and MSIVs and Turbine Control Valve (TCV) fully open. Therefore, the mass flowing in and out of the SGs is balanced. When the RCS reaches the low pressurizer pressure setpoint at 14 seconds, the reactor trip occurs and the MFW and TCV are signaled for closure. The TCV closes almost immediately after the trip while the MFW gradually ramps down as a result of valve stroke for the next 10 seconds (Figure 4-1). The difference in closure timing between the valves increases the mass into the SGs, and consequently the total mass of the secondary, until the MFW valves fully close (Figure 4-2). Once the SGs are isolated from the MFW and the turbine, any changes in the mass of the SGs would occur via auxiliary feedwater (AFW) injection or the MSIVs. However, the safety valves and AFW system are not activated through the transient (ANP-3640P, Figures 4-12 and 4-13) and thus, the total secondary mass is preserved. The small variations in the secondary mass after about 30 seconds are less than 0.1% and within expected S-RELAP5 mass calculation error.

Prior to 200 seconds, there is a second mass change in the SGs which coincides with the start of the SIT injection (169 seconds) in the RCS. At this point the MSIVs are still open, allowing steam flow between the two SGs to maintain equal pressures. Consequently, the steam mass will redistribute in the secondary path in response to changes in the primary such as ECCS cooling. This explains the mass diversion between the two SGs observed bit prior to 200 seconds in Figure 4-14 of ANP-3640P.

As demonstrated in the discussions above, the total secondary mass is preserved through the transient and the SGs mass variations observed in the plots of ANP-3640P are the result of expected steam generator response. Therefore, the predicted behavior in the secondary is physical and reasonable.

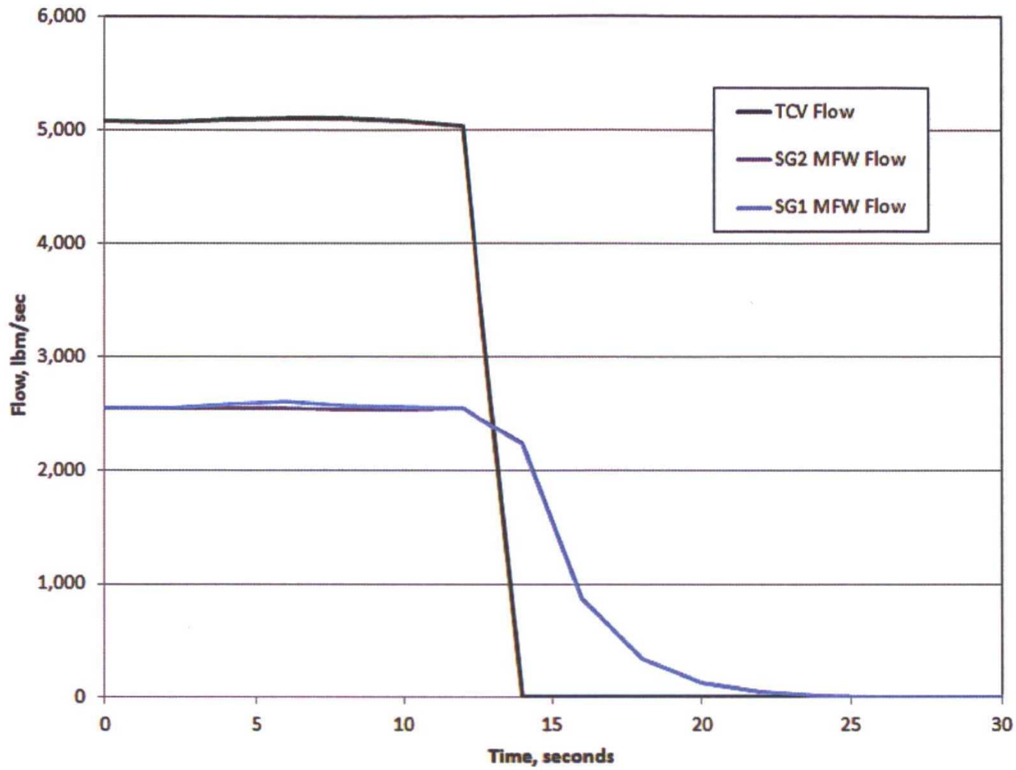

Figure 4-1: Steam Generators and Turbine Control Valve Mass Flow Rates

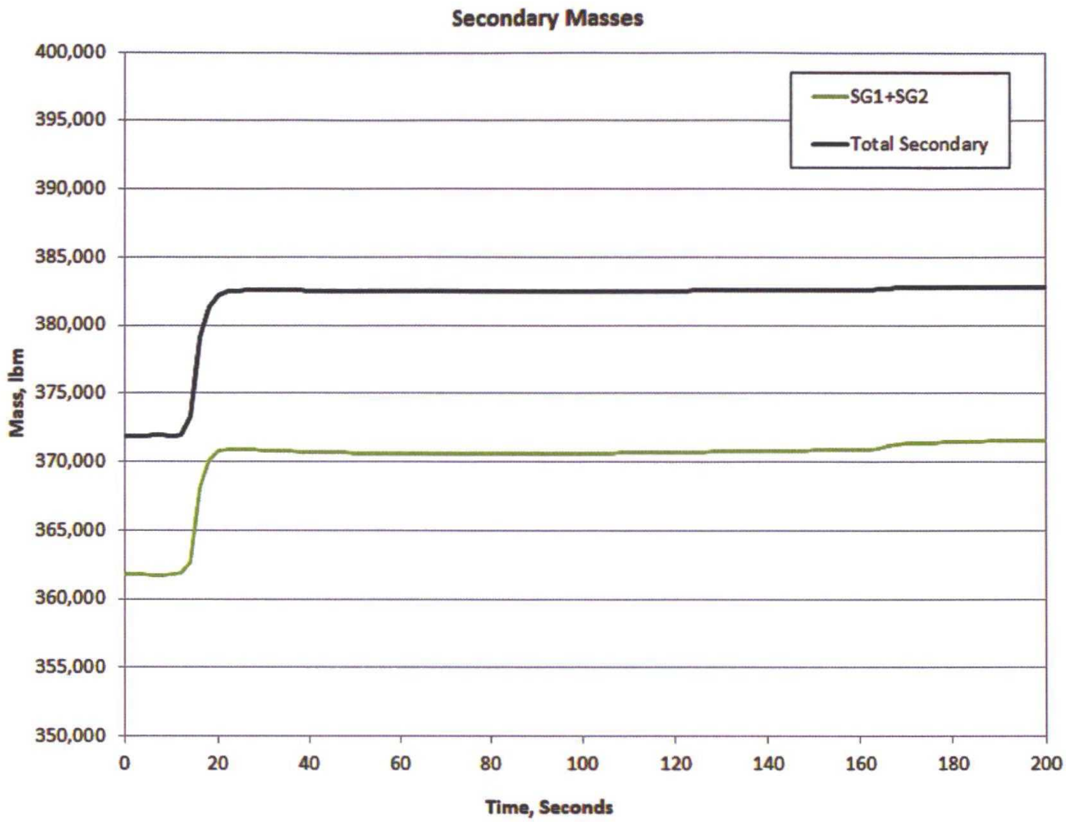

Figure 4-2: Steam Generators and Secondary Total Mass

Response (b):

Figure 4-22 of ANP-3640P shows a plot of the "Hot Assembly Mixture Level." [

] the plot can be misleading, particularly after the core begins to refill and PCT has occurred (approximately 200 seconds for this break size, Figure 4-23 of ANP-3640P).

[ ]

During an SBLOCA, with the exception of the accumulator injection, the core is essentially a boiling pot with the void fraction profile increasing with elevation above the bottom of the core. Consequently, there is no sharp delineation between the liquid region and steam region in the core. [ ] Instead, a plot of void fraction in the core exit and upper plenum provides a much clearer demonstration of the reactor vessel mixture level. Figure 4-3 demonstrates that the mixture level is above the top of the active fuel and that the mixture level is stable following the redistribution of inventory in the reactor vessel after the rapid accumulator surge.

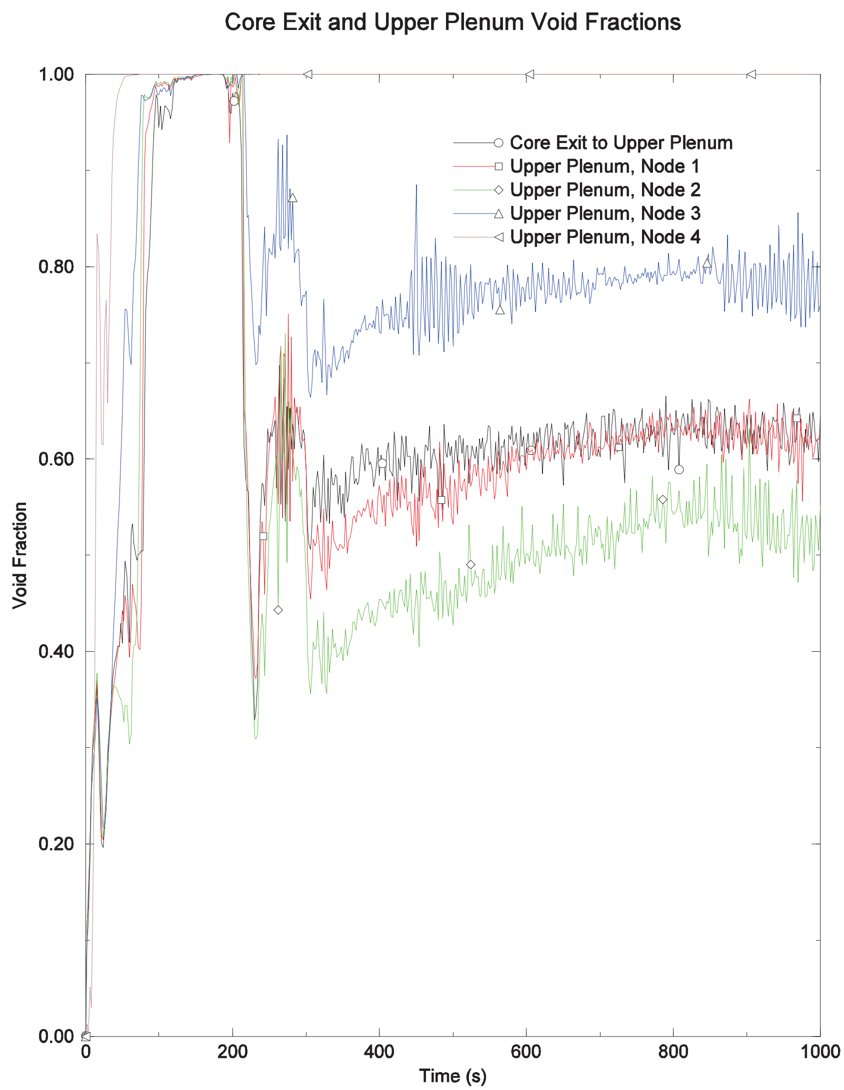

Figure 4-3: Core Exit and Upper Plenum Void Fractions

5.0 RAI-19

Request:

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

] *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:*

[

]

Response:

[

]

[

]

6.0 FRAMATOME SUPPORTING MATERIAL

The following subsections document supporting information for APS to complete the RAI responses.

6.1 *RAI-14*

Request:

To assure the conservatism of the SBLOCA analysis used to demonstrate compliance with the limits of 10 CFR 50.46(b), please provide justification that the 5-minute reactor coolant pump trip delay time assumed in ANP-3640P considers the limiting condition with respect to reactor coolant pump operation for the full range of postulated small breaks on both the cold and hot legs. Please address in particular the range of larger breaks in the SBLOCA spectrum. For such breaks, a 5-minute delay time is essentially equivalent to running the reactor coolant pumps throughout the event, which has long been known to result in reduced PCTs. In responding, please identify the most likely range of times over which plant operators will manually trip the reactor coolant pumps and explain why there is confidence that a reactor coolant pump trip in this time range would be bounded by the existing analyses.

Alternatively, please perform additional sensitivity studies that consider reduced reactor coolant pump trip time delays for break sizes 5 inches and larger to ensure satisfaction of the requirement in 10 CFR 50.46(a)(1)(i) that there is assurance that the most severe postulated LOCAs are calculated. Please further provide a basis for considering any revised sensitivity studies as covering the potential range of times when operators would be expected to complete actions to manually trip the reactor coolant pumps.

Response (supporting material):

An additional sensitivity study with reduced delayed RCP trip time of 30 seconds after RCP NPSH conditions are reached was performed with the same approach as the 5-minute RCP trip delay study. Both, hot leg and cold leg breaks were evaluated.

[

]

The evaluation concluded that the 30 second post-NPSH delay in RCP trip meets the 10 CFR 50.46(b)(1-4) criteria.

The results for the 5-minute RCP trip delay study in ANP-3640P are presented in Table 2-1 and Table 2-2.

6.2 *RAI-16*

Request:

The NRC staff's safety evaluation for EMF-2328(P)(A), Revision 0, Supplement 1, states that [

]

ANP-3640P does not discuss whether switchover to the containment recirculation sump could occur prior to core quench for some SBLOCA events, or whether this behavior was explicitly modeled. Therefore, to ensure a conservative calculation of the figures of merit required to satisfy the acceptance criteria in 10 CFR 50.46(b),

(a) Please identify whether the switchover to sump recirculation could occur prior to core quench for the spectrum of break sizes considered in ANP-3640P. If switchover prior to core quench is not possible, please provide justification.

(b) *If switchover to sump recirculation could occur prior to core quench for breaks in the size range considered in ANP-3640P, then please either (1) demonstrate that the reported figures of merit remain adequately conservative or (2)* [

]

Response (supporting material):

The following information is provided such that a SBLOCA switchover time evaluation can be performed. The intent is that this information in conjunction with pertinent plant information is used to determine the timing that switchover to the containment sump occurs relative to the temperature excursion (quench time) following PCT.

The SBLOCA analysis considered a cold leg break spectrum ranging from a break diameter of 1.0 inch up to 9.49 inches. Table 6-1 summarizes the PCT times and quench times for all of the breaks in the SBLOCA break spectrum analysis. The time of quench is found by observing plots of the temperature excursions for each case in the spectrum and determining when the temperature returns to saturation.

The plots of integrated ECCS mass flow (sum of HPSI and LPSI flows) is provided in Figure 6-1 to Figure 6-25 [ ]

Table 6-1: Break Spectrum PCT and Event Timing

[ ]

Figure 6-1: [ ]

[ ]

Figure 6-2: [ ]

[ ]

Figure 6-3: [ ]

Figure 6-4: [ ]

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Figure 6-5: [ ]

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Figure 6-6: [ ]

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Figure 6-21: [ ]

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Figure 6-22: [ ]

Figure 6-23: [ ]

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Figure 6-24: [ ]

[ ]

Figure 6-25: [ ]

7.0 REFERENCES

1. ANP-3640P, Revision 0, *Palo Verde Units 1, 2 and 3 Small Break LOCA Summary Report*, March 2018.
2. EMF-2328(P)(A), Revision 0, *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*, March 2001.
3. EMF-2328(P)(A), Revision 0, Supplement 1(P)(A), Revision 0, *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*, December 2016.
4. Safety Evaluation, U.S. EPR DC PSER, P2 Group II, (including 15.6.5.1 thru 3 but not including 15.6.5.4), Docket Number 05200020, (NRC Adams Accession Number ML14358A163).
5. EMF-2103(P)(A) Revision 3, *Realistic Large Break LOCA Methodology for Pressurized Water Reactors*, June 2016.

ENCLOSURE
ATTACHMENT 10

Framatome ANP-3785NP, Revision 0
Rod Ejection Accident (AREA) Analysis for Palo Verde

[NON-PROPRIETARY VERSION]

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Rod Ejection Accident (AREA) Analysis for Palo Verde

ANP-3785NP
Revision 0

May 2019

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Nature of Changes

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

Contents

Page

1.0 INTRODUCTION ... 1-1
2.0 OVERVIEW OF CYCLE INPUTS ... 2-1
3.0 REA LIMITS GENERATED BY GALILEO ... 3-1
4.0 FUEL INTEGRITY SUMMARIES ... 4-1
4.1 Core with CE-Standard and Framatome Fuel, Partial-Strength Rods Withdrawn ... 4-3
4.1.1 Minimum Margin to Fuel Melt Limits ... 4-3
4.1.2 Minimum Margin MDNBR SAFDL or Maximum DNBR Failures ... 4-4
4.1.3 Assessment of Limits for a Prompt Critical Excursion ... 4-5
4.1.4 Conservatism of Biasing Method ... 4-6
4.2 Core with CE-Standard and Framatome Fuel, Partial-Strength Rods Inserted ... 4-25
4.2.1 Minimum Margin to Fuel Melt Limits ... 4-25
4.2.2 Minimum Margin MDNBR SAFDL ... 4-26
4.2.3 Assessment of Limits for a Prompt Critical Excursion ... 4-26
4.2.4 Conservatism of Biasing Method ... 4-27
4.3 Core with All Framatome Fuel, Partial-Strength Rods Withdrawn ... 4-46
4.3.1 Minimum Margin to Fuel Melt Limits ... 4-46
4.3.2 Minimum Margin MDNBR SAFDL or Maximum DNBR Failures ... 4-47
4.3.3 Assessment of Limits for a Prompt Critical Excursion ... 4-48
4.3.4 Conservatism of Biasing Method ... 4-49
4.4 Core with All Framatome Fuel, Partial-Strength Rods Inserted ... 4-69
4.4.1 Minimum Margin to Fuel Melt Limits ... 4-69
4.4.2 Minimum Margin MDNBR SAFDL ... 4-70
4.4.3 Assessment of Limits for a Prompt Critical Excursion ... 4-70
4.4.4 Conservatism of Biasing Method ... 4-71
5.0 SUMMARY OF AREA RESULTS ... 5-1
6.0 CYCLE TO CYCLE VERIFICATION ... 6-1
7.0 REFERENCES ... 7-1

List of Tables

Table 2-1 Core Biasing Parameters and Values ... 2-3
Table 4-1 General Timing of Event ... 4-2
Table 4-2 Limiting Results Summary for [ ] CE-Standard/Framatome Core, Partial Strength Rods Withdrawn ... 4-7
Table 4-3 Limiting Results Summary for [ ] CE-Standard/Framatome Core, Partial Strength Rods Withdrawn ... 4-8
Table 4-4 Limiting Results Summary for [ ] CE-Standard/Framatome Core, Partial Strength Rods Withdrawn ... 4-9
Table 4-5 Limiting Results Summary for [ ] CE-Standard/Framatome Core, Partial Strength Rods Withdrawn ... 4-10
Table 4-6 Limiting Results Summary for [ ] CE-Standard/Framatome Core, Partial Strength Rods Withdrawn ... 4-11
Table 4-7 Results Summary for BOC HZP Forced Prompt Critical Event, CE-Standard/Framatome Core, Partial Strength Rods Withdrawn ... 4-12
Table 4-8 Measure of Conservatism for Limiting Result Cases, CE-Standard/Framatome Core, Partial Strength Rods Withdrawn ... 4-13
Table 4-9 Limiting Results Summary for [ ] CE-Standard/Framatome Core, Partial Strength Rods Inserted ... 4-28
Table 4-10 Limiting Results Summary for [ ] CE-Standard/Framatome Core, Partial Strength Rods Inserted ... 4-29
Table 4-11 Limiting Results Summary for [ ] CE-Standard/Framatome Core, Partial Strength Rods Inserted ... 4-30
Table 4-12 Limiting Results Summary for [ ] CE-Standard/Framatome Core, Partial Strength Rods Inserted ... 4-31
Table 4-13 Limiting Results Summary for [ ] CE-Standard/Framatome Core, Partial Strength Rods Inserted ... 4-32
Table 4-14 Results Summary for BOC HZP Forced Prompt Critical Event, CE-Standard/Framatome Core, Partial Strength Rods Inserted ... 4-33
Table 4-15 Measure of Conservatism for Limiting Result Cases, CE-Standard/Framatome Core, Partial Strength Rods Inserted ... 4-34
Table 4-16 Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Withdrawn ... 4-50
Table 4-17 Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Withdrawn ... 4-51

Table 4-18 Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Withdrawn 4-52
Table 4-19 Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Withdrawn 4-53
Table 4-20 Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Withdrawn 4-54
Table 4-21 Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Withdrawn 4-55
Table 4-22 Results Summary for BOC HZP Forced Prompt Critical Event, Framatome Core, Partial Strength Rods Withdrawn 4-56
Table 4-23 Measure of Conservatism for Limiting Result Cases, Framatome Core, Partial Strength Rods Withdrawn 4-57
Table 4-24 Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Inserted 4-72
Table 4-25 Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Inserted 4-73
Table 4-26 Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Inserted 4-74
Table 4-27 Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Inserted 4-75
Table 4-28 Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Inserted 4-76
Table 4-29 Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Inserted 4-77
Table 4-30 Results Summary for BOC HZP Forced Prompt Critical Event, Framatome Core, Partial Strength Rods Inserted 4-78
Table 4-31 Measure of Conservatism for Limiting Result Cases, Framatome Core, Partial Strength Rods Inserted 4-79
Table 6-1 Key Parameter Values for Mixed Core, Part-Strength Rods Withdrawn 6-3
Table 6-2 Key Parameter Values for Mixed Core, Part-Strength Rods Inserted 6-4
Table 6-3 Key Parameter Values for All Framatome Core, Part-Strength Rods Withdrawn 6-5
Table 6-4 Key Parameter Values for All Framatome Core, Part-Strength Rods Inserted 6-6

List of Figures

Figure 2-1 Depressurization Curve 2-4

Figure 3-1 Enthalpy Rise Limits for Zr4 Fuel Based on Excess Hydrogen 3-2

Figure 3-2 Enthalpy Rise Limits for M5 Fuel Based on Excess Hydrogen 3-3

Figure 4-1 Transient F_Q, $F_{\Delta H}$, and Core Power for Max Fuel Temperature Condition, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn) 4-14

Figure 4-2 Transient Fuel, Fuel Rim, and Clad Temperature for Max Fuel Temperature Condition, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn) 4-15

Figure 4-3 Maximum Fuel Temperature by Fuel Type – Margin to Limits for Max Fuel Temperature Condition, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn) 4-16

Figure 4-4 Transient F_Q, $F_{\Delta H}$, and Core Power for MDNBR Condition, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn) 4-17

Figure 4-5 Transient MDNBR for MDNBR Condition, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn) 4-18

Figure 4-6 SAFDL to MDNBR Ratio by Fuel Type as a Function of Burnup for MDNBR Condition, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn) 4-19

Figure 4-7 Transient F_Q, $F_{\Delta H}$, and Core Power for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn) 4-20

Figure 4-8 Transient Maximum Enthalpy for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn) 4-21

Figure 4-9 Maximum Enthalpy Rise and Limits by Clad Type for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn) 4-22

Figure 4-10 Transient Fuel, Fuel Rim, and Clad Temperature for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn) 4-23

Figure 4-11 Maximum Fuel Rim Temperature by Fuel Type for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn) 4-24

Figure 4-12 Transient F_Q, $F_{\Delta H}$, and Core Power for Max Fuel Temperature Conditions, CE-Standard and Framatome Core (Partial Strength Rods Inserted) 4-35

Figure 4-13 Transient Fuel, Fuel Rim, and Clad Temperature for Max Fuel Temperature Conditions, CE-Standard and Framatome Core (Partial Strength Rods Inserted) 4-36

Figure 4-14 Maximum Fuel Temperature by Fuel Type – Margin to Limits for Max Fuel Temperature Condition, CE-Standard and Framatome Core (Partial Strength Rods Inserted)........ 4-37

Figure 4-15 Transient F_Q, $F_{\Delta H}$, and Core Power for MDNBR Condition, CE-Standard and Framatome Core (Partial Strength Rods Inserted)... 4-38

Figure 4-16 Transient MDNBR for MDNBR Condition, CE-Standard and Framatome Core (Partial Strength Rods Inserted) 4-39

Figure 4-17 SAFDL to MDNBR Ratio by Fuel Type as a Function of Burnup for MDNBR Condition, CE-Standard and Framatome Core (Partial Strength Rods Inserted)........ 4-40

Figure 4-18 Transient F_Q, $F_{\Delta H}$, and Core Power for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Inserted) 4-41

Figure 4-19 Transient Maximum Enthalpy for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Inserted)........ 4-42

Figure 4-20 Maximum Enthalpy Rise and Limits by Clad Type for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Inserted) 4-43

Figure 4-21 Transient Fuel, Fuel Rim, and Clad Temperature for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Inserted) 4-44

Figure 4-22 Maximum Fuel Rim Temperature by Fuel Type for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Inserted) 4-45

Figure 4-23 Transient F_Q, $F_{\Delta H}$, and Core Power for Max Fuel Temperature Condition, Framatome Core (Partial Strength Rods Withdrawn) 4-58

Figure 4-24 Transient Fuel, Fuel Rim, and Clad Temperature for Max Fuel Temperature Condition, Framatome Core (Partial Strength Rods Withdrawn)........ 4-59

Figure 4-25 Maximum Fuel Temperature by Fuel Type – Margin to Limits for Max Fuel Temperature Condition, Framatome Core (Partial Strength Rods Withdrawn)........ 4-60

Figure 4-26 Transient F_Q, $F_{\Delta H}$, and Core Power for MDNBR Condition, Framatome Core (Partial Strength Rods Withdrawn)............................ 4-61

Figure 4-27 Transient MDNBR for MDNBR Condition, Framatome Core (Partial Strength Rods Withdrawn)... 4-62

Figure 4-28 SAFDL to MDNBR Ratio by Fuel Type as a Function of Burnup for MDNBR Condition, Framatome Core (Partial Strength Rods Withdrawn)... 4-63

Figure 4-29 Transient F_Q, $F_{\Delta H}$, and Core Power for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Withdrawn) 4-64

Figure 4-30 Transient Maximum Enthalpy for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Withdrawn) 4-65

Figure 4-31 Maximum Enthalpy Rise and Limits by Clad Type for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Withdrawn) 4-66

Figure 4-32 Transient Fuel, Fuel Rim, and Clad Temperature for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Withdrawn) 4-67

Figure 4-33 Maximum Fuel Rim Temperature by Fuel Type for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Withdrawn) 4-68

Figure 4-34 Transient F_Q, $F_{\Delta H}$, and Core Power for Max Fuel Temperature Condition, Framatome Core (Partial Strength Rods Inserted) 4-80

Figure 4-35 Transient Fuel, Fuel Rim, and Clad Temperature for Max Fuel Temperature Condition, Framatome Core (Partial Strength Rods Inserted)... 4-81

Figure 4-36 Maximum Fuel Temperature by Fuel Type – Margin to Limits for Max Fuel Temperature Condition, Framatome Core (Partial Strength Rods Inserted)... 4-82

Figure 4-37 Transient F_Q, $F_{\Delta H}$, and Core Power for MDNBR Condition, Framatome Core (Partial Strength Rods Inserted) 4-83

Figure 4-38 Transient MDNBR for MDNBR Condition, Framatome Core (Partial Strength Rods Inserted).. 4-84

Figure 4-39 SAFDL to MDNBR Ratio by Fuel Type as a Function of Burnup for MDNBR Condition, Framatome Core (Partial Strength Rods Inserted)... 4-85

Figure 4-40 Transient F_Q, $F_{\Delta H}$, and Core Power for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Inserted) 4-86

Figure 4-41 Transient Maximum Enthalpy for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Inserted) 4-87

Figure 4-42 Maximum Enthalpy Rise and Limits by Clad Type for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Inserted) 4-88

Figure 4-43 Transient Fuel, Fuel Rim, and Clad Temperature for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Inserted) 4-89

Figure 4-44 Maximum Fuel Rim Temperature by Fuel Type for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Inserted) 4-90

Nomenclature

Acronym	Definition
AREA	ARCADIA Rod Ejection Accident
AO	Axial Offset
BOC	Beginning of Cycle
CE	Combustion Engineering
CEA	Control Element Assembly
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DTC	Doppler Temperature Coefficient
EFPD	Effective Full Power Days
EOC	End of Cycle
HFP	Hot Full Power
HZP	Hot Zero Power
LAR	License Amendment Request
LOCA	Loss of Coolant Accident
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MTC	Moderator Temperature Coefficient
NRC	Nuclear Regulatory Commission
PCMI	Pellet Cladding Mechanical Interaction
PDIL	Power Dependent Insertion Limit
PVE	Palo Verde Nuclear Generating Station
RAI	Request for Additional Information
REA	Rod Ejection Accident
RTP	Rated Thermal Power
SAFDL	Specified Acceptable Fuel Design Limit
T_{inlet}	Inlet temperature
TIL	Time In Life
VOPT	Variable Over Power Trip
VQP	Vendor Qualification Program
wt%	Weight Percent
Zr4	Zircaloy 4 Alloy

1.0 INTRODUCTION

This document provides results of the Rod Ejection Accident Analysis with ARCADIA (AREA) for Palo Verde Unit 2 (PVE). The results of this analysis are applicable to Unit 1 and to Unit 3 provided the conditions of the cycle are bounded by the analysis conditions.

This report is prepared in response to a potential RAI for the PVE LAR, submitted in August 2018 (Reference 1).

The analysis follows the approved methodology described in Reference 2 with the draft criteria defined in Reference 3. Application of the criteria within the AREA methodology is provided below:

- **[**

  ]

- The enthalpy rise limits are based on excess hydrogen as defined in Reference 3.
- The enthalpy limit used for high temperature cladding failure threshold in Reference 3 is a function of internal pin pressure with a maximum of 170 cal/g for internal pressures less than system pressure and a minimum limit of 100 cal/g for internal pressures higher than system pressure. The cladding failure limit is conservatively set to 100 cal/g for all internal pressures, regardless of burnup.
- Reference 3 has the following restrictions for coolability:
  - Peak radial average fuel enthalpy must remain below 230 cal/g.

- A limited amount of fuel melting is acceptable provided it is restricted to the fuel centerline region and is less than 10 percent of pellet volume. The peak fuel temperature in the outer 90 percent of the pellet's volume must remain below incipient fuel melting conditions.

Both these restrictions are easily met since there are no prompt critical ejected rod worths and no fuel melting. Thus, no additional discussion for coolability is provided in this report.

Clarifications to the methodology that are different than implied by Reference 2 are as follows:

1. GALILEO (Reference 4) is used as the fuel performance code. GALILEO is currently under review with the NRC and has not yet been approved. Reference 4 is a revised version of Reference 5, which was used in Reference 2. The benchmark analysis in Section 5.2.2 of Reference 2 was repeated with the Reference 4 GALILEO version and the results were comparable, with improvements seen in some of the results.
2. The enthalpy rise limits are based upon prompt critical testing. [

   ]
3. System pressure calculations are not performed as part of the current analysis.
4. CE standard fuel is conservatively modeled, as verified by the customer.

The current licensing bases for a rod ejection analysis for PVE include the following requirements:

- Fuel centerline temperature will be less than fuel melt temperature (e.g., no fuel melt, Reference 6, Section 15.4.8.6)
- An assumed failure rate of 19% is used in the radiological consequence evaluation for a CEA ejection event (Reference 6, Section 15.4.8.5 and Table 15.4.8-6).

The AREA analysis is based on the Vendor Qualification Program (VQP) transition cycles and includes analyses of a mixed core with CE Standard fuel as well as a full core of Framatome Fuel. Since Palo Verde has partial strength control rods that are positioned independently of the control banks, the analysis considers configurations with the partial strength rods withdrawn from the core and with the partial strength rods inserted into the core.

2.0 OVERVIEW OF CYCLE INPUTS

Two cycles of the VQP transition cycles are analyzed. The first, Cycle N, contains both CE standard (Zirlo cladding) and Framatome fuel (M5 cladding). The third VQP transition cycle, Cycle N+2, was also analyzed. This cycle contained all Framatome fuel with M5 cladding.

For each cycle, [ ] power levels are examined: [

]

For Cycle N, [ ] cycle burnups are selected. [

]

For Cycle N+2, [ ] cycle burnups are selected. [

]

PVE has a VOPT trip setpoint which is credited in the AREA analysis. The trips used in the simulation for the above power levels are, [

]

The required biases applied in the AREA analysis are stated in Table 2-1 with the depressurization curve supporting the MDNBR analysis as shown in Figure 2-1. Cycle flexibility multipliers were also included to extend the applicability of the cycle analyzed to future cycles. These include multipliers on rod worth, MTC, DTC, β_{eff}, and peaking factors.

Table 2-1
Core Biasing Parameters and Values

Figure 2-1
Depressurization Curve

3.0 REA LIMITS GENERATED BY GALILEO

The Cycle N core contains both M5 and Zirlo clad. The PCMI limit for excess hydrogen is calculated using GALILEO. [

] Zirc4 is used to model Zirlo in the GALILEO calculations. The hydrogen uptake model in GALILEO calculates the total hydrogen content in the clad at a given time in life. The enthalpy deposition limits from Reference 3 are formulated in terms of excess hydrogen, which is the amount of hydrogen above the solubility limit for the clad material. Reference 3 provides an acceptable correlation for solubility of hydrogen in zirconium alloys as a function of the clad temperature. [

] The enthalpy rise limits, based upon the excess hydrogen criteria from Reference 3, are shown in Figure 3-1 and Figure 3-2 for Zr4 and M5, respectively.

Figure 3-1
Enthalpy Rise Limits for Zr4 Fuel Based on Excess Hydrogen

Figure 3-2
Enthalpy Rise Limits for M5 Fuel Based on Excess Hydrogen

[ ]

4.0 FUEL INTEGRITY SUMMARIES

[

] The general timing of the event is provided in Table 4-1.

The most limiting results [ ] are provided [ ] for both the CE-Standard/Framatome core and the all Framatome core. The margins reported are based on the calculated value minus the limit so that a negative number is favorable. Minimum margin is defined as the least negative value. A positive value indicates the maximum amount over the limit. Additional detail is provided for the cases with the least margin to the limit for fuel melt and MDNBR. [

]

Table 4-1
General Timing of Event

Event	Timing
Time to eject rod	0.05 second * fraction of insertion
Trip signal reached	No trips occurred
Time to peak core neutron power	Included with power plot
Time to max enthalpy rise	1 pulse width past the time of peak core neutron power for prompt critical conditions
Rods begin to drop	Total delay time (0.75 seconds)
Rods to full insertion	Total drop time (4.78 seconds)
Simulation ended for the event	[ ]

4.1 *Core with CE-Standard and Framatome Fuel, Partial-Strength Rods Withdrawn*

For each analyzed burnup, limiting results [ ] are displayed in Table 4-2 through Table 4-6 for the CE-Standard/Framatome fuel core with partial strength rod withdrawn. More detail is provided for the case with the least margin to the limit for fuel temperature. DNB failures are found for the [ ] power case at [ ] The maximum number of fuel failures over the cycle is [ ] pin failures (less than [ ] of total number of pins).

[ ] None of the [ ] cases resulted in a prompt critical excursion for this core. A forced prompt critical event was generated and is discussed in Section 4.1.3.

4.1.1 Minimum Margin to Fuel Melt Limits

The current PVE licensing basis is that no fuel melt occurs. The minimum margin to the limit for fuel melt is [ ] which occurs for the [ ] initial power REA transient at [ ] No fuel melt occurs which meets the fuel melt criteria. The results for core power, $F_{\Delta H}$, and F_Q are shown in Figure 4-1. The fuel, fuel rim, and clad temperatures with time are shown in Figure 4-2. [ ] the difference between its fuel temperature and melt limit is shown in the [ ] in Figure 4-3.

4.1.2 Minimum Margin MDNBR SAFDL or Maximum DNBR Failures

Exceeding the MDNBR SAFDL is a failure limit [ ] For the core containing both CE-Standard and Framatome fuel with partial strength rods withdrawn, DNB failures (SAFDL/MDNBR – 1 > 0) are seen for [ ] initial power at [ ] The maximum number of failures is [ ] and occurs at [ ] This corresponds to less than [ ] of the total pins in the core.

The results for core power, $F_{\Delta H}$, and F_Q for the case with the maximum DNB failures are shown in Figure 4-4. The MDNBR versus time is shown in Figure 4-5. The SAFDL is divided by the MDNBR for the [ ] The limit is reached when the ratio becomes 1.0. All fuel pins above 1.0 are assumed failed and those below 1.0 are not. SAFDL/MDNBR values [ ] are shown in the [ ] in Figure 4-6. Failures are seen only in the fresh fuel.

4.1.3 Assessment of Limits for a Prompt Critical Excursion

[

] None of the conditions analyzed [ ] with or without the Table 2-1 biases resulted in a prompt critical event. Therefore, a forced prompt critical excursion was generated at BOC HZP. The prompt critical excursion is forced by increasing the ejected rod worth and decreasing the β_{eff} values such that the ratio between biased rod worth and β_{eff} was ~1.3. The remaining biases for the resulting transient condition were consistent with the fuel thermal biasing provided in Table 2-1. Additional data for the forced prompt critical event is given in Table 4-7. No MDNBR data is shown in Table 4-7 since MDNBR is not defined at HZP.

The total enthalpy limit for all fuel types, regardless of internal pressure, is 100 cal/g. This limit represents the minimum value of the function given in Reference 3. For this case, the minimum margin to the enthalpy limit for high clad temperature failure criteria is [ ] The results for core power, $F_{\Delta H}$, and F_Q are shown in Figure 4-7. The maximum cal/g in the core with time is given in Figure 4-8. The total enthalpy limit is not challenged by this core design.

The minimum margin to the limit for enthalpy rise is [ ] No failures are seen for either Zr4 or M5 clad. The enthalpy rise is terminated one pulse width after the peak and is given in Figure 4-7. The Δcal/g results and limits for Zr4 and M5 clad types are shown in the [ ] in Figure 4-9 [

] As expected, the Zr4 clad has the least margin at high burnups but remains more than [ ] below the limit.

The minimum margin to the limit for fuel rim melt is [ ] No rim melt occurs, which meets the fuel rim melt criterion. The fuel, fuel rim, and clad temperatures with time are shown in Figure 4-10. The minimum difference [ ] between the fuel rim temperature and its limit is shown in the [ ] in Figure 4-11.

4.1.4 Conservatism of Biasing Method

Based on the results in Table 4-2 through Table 4-7, an assessment of the limiting case for enthalpy, enthalpy rise, fuel temperature, rim temperature, and MDNBR is presented and summarized in Table 4-8. For each of the applicable limiting criteria, the power level, cycle burnup, [ ] are provided. There is ample conservatism for each parameter given in this table.

Table 4-2
Limiting Results Summary for [ ] CE-Standard/Framatome Core, Partial Strength Rods Withdrawn

[ ]

Table 4-3
Limiting Results Summary for [ ]
CE-Standard/Framatome Core, Partial Strength Rods Withdrawn

[ ]

Table 4-4
Limiting Results Summary for [ ]
CE-Standard/Framatome Core, Partial Strength Rods Withdrawn

[ ]

Table 4-5
Limiting Results Summary for [ ]
CE-Standard/Framatome Core, Partial Strength Rods Withdrawn

[ ]

Table 4-6
Limiting Results Summary for [ ]
CE-Standard/Framatome Core, Partial Strength Rods Withdrawn

[ ]

Table 4-7
Results Summary for BOC HZP Forced Prompt Critical Event, CE-Standard/Framatome Core, Partial Strength Rods Withdrawn

[ ]

Table 4-8
Measure of Conservatism for Limiting Result Cases,
CE-Standard/Framatome Core, Partial Strength Rods Withdrawn

[ ]

Figure 4-1
Transient F_Q, $F_{\Delta H}$, and Core Power for Max Fuel Temperature Condition, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn)

Figure 4-2
Transient Fuel, Fuel Rim, and Clad Temperature for Max Fuel Temperature Condition, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn)

[

]

Figure 4-3
Maximum Fuel Temperature by Fuel Type – Margin to Limits for Max Fuel Temperature Condition, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn)

[

]

Figure 4-4
Transient F_Q, $F_{\Delta H}$, and Core Power for MDNBR Condition, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn)

[

]

Figure 4-5
Transient MDNBR for MDNBR Condition, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn)

[ ]

Figure 4-6
SAFDL to MDNBR Ratio by Fuel Type as a Function of Burnup for MDNBR Condition, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn)

[ ]

Figure 4-7
Transient F_Q, $F_{\Delta H}$, and Core Power for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn)

[ ]

Figure 4-8
Transient Maximum Enthalpy for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn)

Figure 4-9
Maximum Enthalpy Rise and Limits by Clad Type for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn)

[

]

Figure 4-10
Transient Fuel, Fuel Rim, and Clad Temperature for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn)

[

]

Figure 4-11
Maximum Fuel Rim Temperature by Fuel Type for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Withdrawn)

[ ]

4.2 *Core with CE-Standard and Framatome Fuel, Partial-Strength Rods Inserted*

For each analyzed burnup, limiting results [

] are displayed in Table 4-9 through Table 4-13 for the CE-Standard/Framatome fuel core with partial strength rods inserted. More detail is provided for the case with the least margin to the limit for fuel temperature and MDNBR.

[

] None of the [ ] cases resulted in a prompt critical excursion for this core. A forced prompt critical event was generated and is discussed in Section 4.2.3.

4.2.1 Minimum Margin to Fuel Melt Limits

The current PVE licensing basis is that no fuel melt occurs. The minimum margin to the limit for fuel melt is [ ] which occurs for the [ ] initial power REA transient at [ ] No fuel melt occurs which meets the fuel melt criteria. The results for core power, $F_{\Delta H}$, and F_Q are shown in Figure 4-12. The fuel, fuel rim, and clad temperatures with time are shown in Figure 4-13. [

] the difference between its fuel temperature and melt limit is shown in the [ ] in Figure 4-14.

4.2.2 Minimum Margin MDNBR SAFDL

Exceeding the MDNBR SAFDL is a failure limit [ ] For the core containing both CE-Standard and Framatome fuel with partial strength rods inserted, no DNB failures (SAFDL/MDNBR – 1 > 0) are seen. The minimum margin to the limit (SAFDL/MDNBR – 1) is [ ] which occurs for the [ ] initial power REA transient at [ ] The results for core power, $F_{\Delta H}$, and F_Q are shown in Figure 4-15. The MDNBR versus time is shown in Figure 4-16. The SAFDL is divided by the MDNBR for the [ ] The limit is reached when the ratio becomes 1.0. All fuel pins above 1.0 are assumed failed and those below 1.0 are not. SAFDL/MDNBR values for [ ] are shown in the [ ] in Figure 4-17.

4.2.3 Assessment of Limits for a Prompt Critical Excursion

[ ] None of the conditions analyzed (burnup, power, AO) with or without the Table 2-1 biases resulted in a prompt critical event. Therefore, a forced prompt critical excursion was generated at BOC HZP. The prompt critical excursion is forced by increasing the ejected rod worth and decreasing the β_{eff} values such that the ratio between biased rod worth and β_{eff} was ~1.3. The remaining biases for the resulting transient condition were consistent with the fuel thermal biasing provided in Table 2-1. Additional data for the forced prompt critical event is given in Table 4-14. No MDNBR data is shown in Table 4-14 since MDNBR is not defined at HZP.

The total enthalpy limit for all fuel types, regardless of internal pressure, is 100 cal/g. This limit represents the minimum value of the function given in Reference 3. For this case, the minimum margin to the enthalpy limit for high clad temperature failure criteria is [ ] The results for core power, $F_{\Delta H}$, and F_Q are shown in Figure 4-18. The maximum cal/g in the core with time is given in Figure 4-19. The total enthalpy limit is not challenged by this core design.

The minimum margin to the limit for enthalpy rise is [ ] No failures are seen for either Zr4 or M5 clad. The enthalpy rise is terminated one pulse width after the peak and is given in Figure 4-18. The Δcal/g results and limits for Zr4 and M5 clad types are shown in the [ ] in Figure 4-20 [ ] As expected, the Zr4 clad has the least margin at high burnups but remains more than [ ] below the limit.

The minimum margin to the limit for fuel rim melt is [ ] No rim melt occurs, which meets the fuel rim melt criterion. The fuel, fuel rim, and clad temperatures with time are shown in Figure 4-21. The minimum difference [ ] between the fuel rim temperature and its limit is shown in the [ ] in Figure 4-22.

4.2.4 Conservatism of Biasing Method

Based on the results in Table 4-9 through Table 4-14, an assessment of the limiting case for enthalpy, enthalpy rise, fuel temperature, rim temperature, and MDNBR is presented and summarized in Table 4-15. For each of the applicable limiting criteria, the power level, cycle burnup, [ ] are provided. There is ample conservatism for each parameter given in this table.

Table 4-9
Limiting Results Summary for [ ] CE-Standard/Framatome Core, Partial Strength Rods Inserted

[ ]

Table 4-10
Limiting Results Summary for [ ]
CE-Standard/Framatome Core, Partial Strength Rods Inserted

[ ]

Table 4-11
Limiting Results Summary for [ ]
CE-Standard/Framatome Core, Partial Strength Rods Inserted

[ ]

Table 4-12
Limiting Results Summary for [ ]
CE-Standard/Framatome Core, Partial Strength Rods Inserted

[ ]

Table 4-13
Limiting Results Summary for [ ]
CE-Standard/Framatome Core, Partial Strength Rods Inserted

[

]

Table 4-14
Results Summary for BOC HZP Forced Prompt Critical Event, CE-Standard/Framatome Core, Partial Strength Rods Inserted

Table 4-15
Measure of Conservatism for Limiting Result Cases,
CE-Standard/Framatome Core, Partial Strength Rods Inserted

[ ]

Figure 4-12
Transient F_Q, $F_{\Delta H}$, and Core Power for Max Fuel Temperature Conditions, CE-Standard and Framatome Core (Partial Strength Rods Inserted)

[ ]

Figure 4-13
Transient Fuel, Fuel Rim, and Clad Temperature for Max Fuel Temperature Conditions, CE-Standard and Framatome Core (Partial Strength Rods Inserted)

[

]

Figure 4-14
Maximum Fuel Temperature by Fuel Type – Margin to Limits for Max Fuel Temperature Condition, CE-Standard and Framatome Core (Partial Strength Rods Inserted)

[ ]

Figure 4-15
Transient F_Q, $F_{\Delta H}$, and Core Power for MDNBR Condition, CE-Standard and Framatome Core (Partial Strength Rods Inserted)

[ ]

Figure 4-16
Transient MDNBR for MDNBR Condition, CE-Standard and Framatome Core (Partial Strength Rods Inserted)

[

]

Figure 4-17
SAFDL to MDNBR Ratio by Fuel Type as a Function of Burnup for MDNBR Condition, CE-Standard and Framatome Core (Partial Strength Rods Inserted)

[

]

Figure 4-18
Transient F_Q, $F_{\Delta H}$, and Core Power for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Inserted)

[

]

Figure 4-19
Transient Maximum Enthalpy for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Inserted)

[

]

Figure 4-20
Maximum Enthalpy Rise and Limits by Clad Type for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Inserted)

[ ]

Figure 4-21
Transient Fuel, Fuel Rim, and Clad Temperature for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Inserted)

Figure 4-22
Maximum Fuel Rim Temperature by Fuel Type for HZP BOC Forced Prompt Critical Event, CE-Standard and Framatome Core (Partial Strength Rods Inserted)

[ ]

4.3 *Core with All Framatome Fuel, Partial-Strength Rods Withdrawn*

For each analyzed burnup, limiting results [ ] are displayed in Table 4-16 through Table 4-21 for the Framatome fuel core with partial strength rods withdrawn. More detail is provided for the case with the least margin to the limit for fuel temperature. DNB failures are found for the [ ] power case at [ ] The maximum number of fuel failures over the cycle is [ ] pin failures (less than [ ] of total number of pins).

[ ] None of the [ ] cases resulted in a prompt critical excursion for this core. A forced prompt critical event was generated and is discussed in Section 4.3.3.

4.3.1 Minimum Margin to Fuel Melt Limits

The current PVE licensing basis is that no fuel melt occurs. The minimum margin to the limit for fuel melt is [ ] which occurs for the [ ] initial power REA transient at [ ] No fuel melt occurs which meets the fuel melt criteria. The results for core power, $F_{\Delta H}$, and F_Q are shown in Figure 4-23. The fuel, fuel rim, and clad temperatures with time are shown in Figure 4-24. [ ] the difference between its fuel temperature and melt limit is shown in the [ ] in Figure 4-25.

4.3.2 Minimum Margin MDNBR SAFDL or Maximum DNBR Failures

Exceeding the MDNBR SAFDL is a failure limit [ ] For the core containing only Framatome fuel with partial strength rods withdrawn, DNB failures (SAFDL/MDNBR – 1 > 0) are seen for [ ] initial power at [ ] The maximum number of failures is [ ] pins and occurs at [ ] This corresponds to less than [ ] of the total pins in the core.

The results for core power, $F_{\Delta H}$, and F_Q for the case with the maximum DNB failures are shown in Figure 4-26. The MDNBR versus time is shown in Figure 4-27. The SAFDL is divided by the MDNBR for the [ ] The limit is reached when the ratio becomes 1.0. All fuel pins above 1.0 are assumed failed and those below 1.0 are not. SAFDL/MDNBR values [ ] are shown in the [ ] in Figure 4-28. Failures are seen only in the fresh fuel.

4.3.3 Assessment of Limits for a Prompt Critical Excursion

[

] None of the conditions analyzed (burnup, power, AO) with or without the Table 2-1 biases resulted in a prompt critical event. Therefore, a forced prompt critical excursion was generated at BOC HZP. The prompt critical excursion is forced by increasing the ejected rod worth and decreasing the β_{eff} values such that the ratio between biased rod worth and β_{eff} was ~1.3. The remaining biases for the resulting transient condition were consistent with the fuel thermal biasing provided in Table 2-1. Additional data for the forced prompt critical event is given in Table 4-22. No MDNBR data is shown in Table 4-22 since MDNBR is not defined at HZP.

The total enthalpy limit for all fuel types, regardless of internal pressure, is 100 cal/g. This limit represents the minimum value of the function given in Reference 3. For this case, the minimum margin to the enthalpy limit for high clad temperature failure criteria is [ ] The results for core power, $F_{\Delta H}$, and F_Q are shown in Figure 4-29. The maximum cal/g in the core with time is given in Figure 4-30. The total enthalpy limit is not challenged by this core design.

The minimum margin to the limit for enthalpy rise is [ ] No failures are seen. The enthalpy rise is terminated one pulse width after the peak and is given in Figure 4-29. The Δcal/g results and limits for M5 clad are shown in the [ ] in Figure 4-31 [ ]

The minimum margin to the limit for fuel rim melt is [ ] No rim melt occurs, which meets the fuel rim melt criterion. The fuel, fuel rim, and clad temperatures with time are shown in Figure 4-32. The minimum difference [ ] between the fuel rim temperature and its limit is shown in the [ ] in Figure 4-33.

4.3.4 Conservatism of Biasing Method

Based on the results in Table 4-16 through Table 4-22, an assessment of the limiting case for enthalpy, enthalpy rise, fuel temperature, rim temperature, and MDNBR is presented and summarized in Table 4-23. For each of the applicable limiting criteria, the power level, cycle burnup, [ ] are provided. There is ample conservatism for each parameter given in this table.

Table 4-16
Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Withdrawn

[ ]

Table 4-17
Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Withdrawn

[ ]

Table 4-18
Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Withdrawn

[

]

Table 4-19

Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Withdrawn

[ ]

Table 4-20
Limiting Results Summary for [ ] Framatome Core,
Partial Strength Rods Withdrawn

[ ]

Table 4-21
Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Withdrawn

[ ]

Table 4-22
Results Summary for BOC HZP Forced Prompt Critical Event, Framatome Core, Partial Strength Rods Withdrawn

[ ]

Table 4-23
Measure of Conservatism for Limiting Result Cases, Framatome Core, Partial Strength Rods Withdrawn

Figure 4-23
Transient F_Q, $F_{\Delta H}$, and Core Power for Max Fuel Temperature Condition, Framatome Core (Partial Strength Rods Withdrawn)

[ ]

Figure 4-24
Transient Fuel, Fuel Rim, and Clad Temperature for Max Fuel Temperature Condition, Framatome Core (Partial Strength Rods Withdrawn)

Figure 4-25
Maximum Fuel Temperature by Fuel Type – Margin to Limits for Max Fuel Temperature Condition, Framatome Core (Partial Strength Rods Withdrawn)

Figure 4-26
Transient F_Q, $F_{\Delta H}$, and Core Power for MDNBR Condition, Framatome Core (Partial Strength Rods Withdrawn)

[

]

Figure 4-27
Transient MDNBR for MDNBR Condition,
Framatome Core (Partial Strength Rods Withdrawn)

Figure 4-28
SAFDL to MDNBR Ratio by Fuel Type as a Function of Burnup for MDNBR Condition, Framatome Core (Partial Strength Rods Withdrawn)

[ ]

Figure 4-29
Transient F_Q, $F_{\Delta H}$, and Core Power for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Withdrawn)

[ ]

Figure 4-30
Transient Maximum Enthalpy for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Withdrawn)

Figure 4-31
Maximum Enthalpy Rise and Limits by Clad Type for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Withdrawn)

Figure 4-32
Transient Fuel, Fuel Rim, and Clad Temperature for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Withdrawn)

[

]

Figure 4-33
Maximum Fuel Rim Temperature by Fuel Type for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Withdrawn)

[

]

4.4 *Core with All Framatome Fuel, Partial-Strength Rods Inserted*

For each analyzed burnup, limiting results [ ] are displayed in Table 4-24 through Table 4-29 for the Framatome fuel core with partial strength rods inserted. More detail is provided for the case with the least margin to the limit for fuel temperature and MDNBR.

[ ] None of the [ ] cases resulted in a prompt critical excursion for this core. A forced prompt critical event was generated and is discussed in Section 4.4.3.

4.4.1 Minimum Margin to Fuel Melt Limits

The current PVE licensing basis is that no fuel melt occurs. The minimum margin to the limit for fuel melt is [ ] which occurs for the [ ] initial power REA transient at [ ] No fuel melt occurs which meets the fuel melt criteria. The results for core power, $F_{\Delta H}$, and F_Q are shown in Figure 4-34. The fuel, fuel rim, and clad temperatures with time are shown in Figure 4-35. [ ] the difference between its fuel temperature and melt limit is shown in the [ ] in Figure 4-36.

4.4.2 Minimum Margin MDNBR SAFDL

Exceeding the MDNBR SAFDL is a failure limit [ ] For the core containing only Framatome fuel with partial strength rods inserted, no DNB failures (SAFDL/MDNBR – 1 > 0) are seen. The minimum margin to the limit (SAFDL/MDNBR – 1) is [ ] which occurs for the [ ] initial power REA transient at [ ] The results for core power, $F_{\Delta H}$, and F_Q are shown in Figure 4-37. The MDNBR versus time is shown in Figure 4-38. The SAFDL is divided by the MDNBR for the [ ] The limit is reached when the ratio becomes 1.0. All fuel pins above 1.0 are assumed failed and those below 1.0 are not. SAFDL/MDNBR values [for every] are shown in the [ ] in Figure 4-39.

4.4.3 Assessment of Limits for a Prompt Critical Excursion

[ ] None of the conditions analyzed (burnup, power, AO) with or without the Table 2-1 biases resulted in a prompt critical event. Therefore, a forced prompt critical excursion was generated at BOC HZP. The prompt critical excursion is forced by increasing the ejected rod worth and decreasing the β_{eff} values such that the ratio between biased rod worth and β_{eff} was ~1.3. The remaining biases for the resulting transient condition were consistent with the fuel thermal biasing provided in Table 2-1. Additional data for the forced prompt critical event is given in Table 4-30. No MDNBR data is shown in Table 4-30 since MDNBR is not defined at HZP.

The total enthalpy limit for all fuel types, regardless of internal pressure, is 100 cal/g. This limit represents the minimum value of the function given in Reference 3. For this case, the minimum margin to the enthalpy limit for high clad temperature failure criteria is [ ] The results for core power, $F_{\Delta H}$, and F_Q are shown in Figure 4-40. The maximum cal/g in the core with time is given in Figure 4-41. The total enthalpy limit is not challenged by this core design.

The minimum margin to the limit for enthalpy rise is [ ] No failures are seen. The enthalpy rise is terminated one pulse width after the peak and is given in Figure 4-40. The Δcal/g results and limits for M5 clad are shown in the [ ] in Figure 4-42 [ ]

The minimum margin to the limit for fuel rim melt is [ ] No rim melt occurs, which meets the fuel rim melt criterion. The fuel, fuel rim, and clad temperatures with time are shown in Figure 4-43. The minimum difference [ ] between the fuel rim temperature and its limit is shown in the [ ] in Figure 4-44.

4.4.4 Conservatism of Biasing Method

Based on the results in Table 4-24 through Table 4-30, an assessment of the limiting case for enthalpy, enthalpy rise, fuel temperature, rim temperature, and MDNBR is presented and summarized in Table 4-31. For each of the applicable limiting criteria, the power level, cycle burnup, [ ] are provided. There is ample conservatism for each parameter given in this table.

Table 4-24
Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Inserted

[ ]

Table 4-25
Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Inserted

[ ]

Table 4-26
Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Inserted

[ ]

Table 4-27
Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Inserted

[ ]

Table 4-28
Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Inserted

[ ]

Table 4-29
Limiting Results Summary for [ ] Framatome Core, Partial Strength Rods Inserted

[ ]

Table 4-30
Results Summary for BOC HZP Forced Prompt Critical Event, Framatome Core, Partial Strength Rods Inserted

[ ]

Table 4-31
Measure of Conservatism for Limiting Result Cases, Framatome Core, Partial Strength Rods Inserted

[ ]

Figure 4-34
Transient F_Q, $F_{\Delta H}$, and Core Power for Max Fuel Temperature Condition, Framatome Core (Partial Strength Rods Inserted)

[ ]

Figure 4-35
Transient Fuel, Fuel Rim, and Clad Temperature for Max Fuel Temperature Condition, Framatome Core (Partial Strength Rods Inserted)

[ ]

Figure 4-36
Maximum Fuel Temperature by Fuel Type – Margin to Limits for Max Fuel Temperature Condition, Framatome Core (Partial Strength Rods Inserted)

[ ]

Figure 4-37
Transient F_Q, $F_{\Delta H}$, and Core Power for MDNBR Condition, Framatome Core (Partial Strength Rods Inserted)

[

]

Figure 4-38
Transient MDNBR for MDNBR Condition,
Framatome Core (Partial Strength Rods Inserted)

[ ]

Figure 4-39
SAFDL to MDNBR Ratio by Fuel Type as a Function of Burnup for MDNBR Condition, Framatome Core (Partial Strength Rods Inserted)

Figure 4-40
Transient F_Q, $F_{\Delta H}$, and Core Power for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Inserted)

[ ]

Figure 4-41
Transient Maximum Enthalpy for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Inserted)

[

]

Figure 4-42
Maximum Enthalpy Rise and Limits by Clad Type for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Inserted)

[ ]

Figure 4-43
Transient Fuel, Fuel Rim, and Clad Temperature for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Inserted)

[ ]

Figure 4-44
Maximum Fuel Rim Temperature by Fuel Type for HZP BOC Forced Prompt Critical Event, Framatome Core (Partial Strength Rods Inserted)

[ ]

5.0 SUMMARY OF AREA RESULTS

The AREA methodology in Reference 2 was applied to PVE for both a CE-Standard / Framatome fuel mixed core and a Framatome fuel only core. For each core, the insertion of partial strength rods was considered. In general, results from the cores in which the partial strength rods were withdrawn bound results from the cores in which the partial strength rods were inserted.

For both cores, the enthalpy, enthalpy rise, fuel melt, and fuel rim melt limits were not challenged. No failures were seen for these parameters.

For MDNBR, failures of less than [ ] of the total pins were seen in both cores at [ ] power during the second half of the cycle. Failures occurred only in the fresh fuel. No enhanced fission gas released is estimated since enthalpy rise at full power is negligible. The MDNBR failure rate seen in this analysis is significantly below the assumed failure rate of 19% assumed in the radiological consequence evaluation for a CEA ejection event (Reference 6, Section 15.4.8.5 and Table 15.4.8-6).

6.0 CYCLE TO CYCLE VERIFICATION

No limiting conditions are approached with high burnup fuel for the current AREA analysis. For this reason, checking the peak analysis static conditions as outlined in the AREA topical report (Reference 2, Section 8.2) is a valid method to perform cycle-to-cycle verification. Table 6-1 through Table 6-4 summarize the maximum values for the key parameters used in the current analyses. These tables form a basis to perform a cycle-specific verification to determine if the cycle key parameters remain within the range of the values used in the current analysis. The steady state key parameters are listed below:

- Ejected rod worth
- β_{eff}
- MTC
- DTC
- Initial F_Q (at power cases only)
- Initial $F_{\Delta H}$ (at power cases only)
- Static post ejection F_Q
- Static post ejection $F_{\Delta H}$

For a mixed core of CE standard and Framatome fuel, the limiting values for the key parameters are found in Table 6-1 with the partial strength rods withdrawn and in Table 6-2 with partial strength rods inserted.

For a fuel core of Framatome fuel, the limiting values for the key parameters are found in Table 6-3 with the partial strength rods withdrawn and in Table 6-4 with partial strength rods inserted.

If the key parameters are not exceeded in future cycles, then no additional analysis is required.

In the event that any of the key parameters are not bounded in future cycles, there are two approaches available besides redesigning the cycle to meet the conditions in the tables:

1. Complete reanalysis of the matrix of cases. This approach is selected when a new baseline matrix of cases is needed. This option is typically employed for major fuel design changes that are outside the scope of the original analysis.

2. Reanalysis of a portion of the matrix of cases for the condition where a specific parameter is found to be outside the initial application analysis range.
   [
   ] This option is typically employed for minor fuel design changes that are challenging isolated conditions of the original analysis.

Table 6-1
Key Parameter Values for Mixed Core, Part-Strength Rods Withdrawn

Table 6-2
Key Parameter Values for Mixed Core, Part-Strength Rods Inserted

Table 6-3
Key Parameter Values for All Framatome Core, Part-Strength Rods Withdrawn

Table 6-4
Key Parameter Values for All Framatome Core, Part-Strength Rods Inserted

7.0 REFERENCES

1. Palo Verde Nuclear Generating Station Units 1, 2, and 3, Docket Nos. STN 50-528, 59-529, and 50-530 License Amendment Request and Exemption Request to Support the Implementation of Framatome High Thermal Performance Fuel, ADAMS Accession No. ML18187A417.
2. ANP-10338P-A, Revision 0, "AREA™ – ARCADIA® Rod Ejection Accident," December 2017.
3. Draft Regulatory Guide DG-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents," US NRC, ML16124A200, November 2016.
4. ANP-10323P, Revision 1, "GALILEO Fuel Rod Thermal-Mechanical Methodology for Pressurized Water Reactors," June 2018.
5. ANP-10323P, Revision 0, "Fuel Rod Thermal-Mechanical Methodology for Boiling Water Reactors and Pressurized Water Reactors," July 2013.
6. Updated Final Safety Analysis Report, Revision 18C, Palo Verde Nuclear Generating Station Units 1, 2, and 3, July 2016.