10 CFR 50.90

10 CFR 50.12

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102-07727-MLL/SMM
July 6, 2018

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

Dear Sirs:

Subject: **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

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Arizona Public Service (APS) is submitting a request for an amendment of the Palo Verde Nuclear Generating Station (PVNGS) Operating License Nos. NPF-41, NPF-51, and NPF-74 by revising the Technical Specifications (TS) to support the implementation of Framatome (formerly AREVA, Inc.) Advanced Combustion Engineering 16x16 (CE-16) High Thermal Performance (HTP™) fuel design with M5® as a fuel rod cladding material and gadolinia as a burnable absorber. In addition to the license amendment, pursuant to 10 CFR 50.12, *Specific Exemptions*, APS is requesting an exemption from the requirements of 10 CFR 50.46, *Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors*, and 10 CFR 50, Appendix K, *ECCS Evaluation Models*, to allow the use of the Framatome M5® alloy as a fuel rod cladding material.

The enclosure to this letter provides a description and assessment of the proposed changes including a technical evaluation, a regulatory evaluation, a no significant hazards consideration, and an environmental evaluation. The enclosure also includes eleven attachments. Attachment 1 provides a new regulatory commitment (as defined by NEI 99-04, *Guidelines for Managing NRC Commitment Changes*, Revision 0) to be implemented. Attachment 2 provides marked-up existing TS pages. Attachment 3 provides revised (clean) TS pages. Attachment 4 provides marked-up TS Bases pages to show the conforming changes for information only. Attachment 5 provides an assessment of limitations and conditions contained in the safety evaluations for NRC-approved topical reports related to this license amendment request.

Attachment 6 contains two affidavits signed by Framatome that set forth the basis on which the proprietary information in Attachment 11 may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in 10 CFR 2.390(b)(4). Correspondence with respect to the proprietary aspects of Attachment 11 or the supporting Framatome affidavits should be addressed to Philip A. Opsal of Framatome.

Attachment 7 is an affidavit signed by APS that sets forth the basis on which the proprietary information in Attachment 10 may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in 10 CFR 2.390(b)(4). Correspondence with respect to the proprietary aspects of Attachment 10 or the supporting APS affidavit should be addressed to Bruce Rash of APS.

Attachment 8 provides a non-proprietary version of the technical analysis supporting this LAR. Attachment 10 provides a proprietary version of the technical analysis supporting this LAR, which contains information proprietary to APS.

Attachment 9 provides a non-proprietary version of the Framatome licensing summary reports for large and small break loss of coolant accidents (LOCAs) supporting this LAR. Attachment 11 provides proprietary versions of these reports supporting this LAR, which contains information proprietary to Framatome.

A pre-submittal meeting was held with the NRC on April 5, 2018 (Agency Document Access and Management System [ADAMS] Accession Number ML181028212) to discuss various aspects of this LAR.

In accordance with the PVNGS Quality Assurance Program Description, the Plant Review Board has reviewed and approved the proposed amendment. By copy of this letter, this license amendment request is being forwarded to the Arizona Department of Health Services Bureau of Radiation Control in accordance with 10 CFR 50.91(b)(1).

APS requests approval of the proposed license amendment within 12 months following acceptance of this LAR to support planned implementation during the Spring 2020 Unit 2 refueling outage. APS will implement the approved license amendment within 120 days.

Should you have any questions concerning the content of this letter, please contact Matthew S. Cox, Licensing Section Leader, at (623) 393-5753.

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

Executed on July 6, 2018
(Date)

Sincerely,

Lacal, Maria L(Z06149)

Digitally signed by Lacal, Maria L(Z06149)
DN: cn=Lacal, Maria L(Z06149)
Date: 2018.07.06 13:45:34 -07'00'

MLL/MSC/SMM/maw

Enclosure: Description and Assessment of Proposed License Amendment

cc:	K. M. Kennedy	NRC Region IV Regional Administrator
	M. D. Orenak	NRC NRR Project Manager for PVNGS
	M. M. O'Banion	NRC NRR Project Manager
	C. A. Peabody	NRC Senior Resident Inspector for PVNGS
	T. Morales	Arizona Department of Health Services – Bureau of Radiation Control (ADHS)

ENCLOSURE

Description and Assessment of Proposed License Amendment

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1. SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating Licenses NPF-41, NPF-51, and NPF-74, for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, respectively.

The license amendment request (LAR) will revise the Technical Specifications (TS) for PVNGS Units 1, 2, and 3 to support implementation of Framatome (formerly AREVA) Advanced Combustion Engineering 16x16 (CE-16) High Thermal Performance (HTP™) fuel design with M5® as a fuel rod cladding material and gadolinia as a burnable absorber. 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.

In addition to this license amendment request, PVNGS is requesting an exemption from certain requirements of 10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors*, and 10 CFR 50, Appendix K, *ECCS Evaluation Models*, to allow the use of the Framatome M5® alloy as a fuel rod cladding material.

This change is planned to be first implemented with the Spring 2020 Unit 2 refueling outage.

2. PROPOSED CHANGES TO PVNGS LICENSING BASIS

Implementation of Framatome fuel at PVNGS Units 1, 2 and 3 requires the following changes to the PVNGS licensing basis:

- TS 2.1.1.2 – Clarify that the existing TS is a Westinghouse-supplied fuel peak fuel centerline temperature safety limit and add a Framatome-supplied peak fuel centerline temperature safety limit.
- TS 4.2.1 – Simplify the list of fuel rod cladding materials to “zirconium-alloy clad,” add a paragraph break prior to the text addressing lead test assemblies and delete the last sentence which duplicates the requirements in 10 CFR 50.12 within the TS.
- TS 5.6.5 – Add to the listing of analytical methods used to determine the core operating limits as described in documents previously reviewed and approved by the NRC.
- A permanent exemption from certain requirements of 10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors*, and 10 CFR 50 Appendix K, *ECCS Evaluation Models*.

3. TECHNICAL SPECIFICATIONS CHANGES

The proposed implementation of Framatome fuel requires changes to TS 2.1.1.2, TS 4.2.1, and TS 5.6.5. The following subsections address these changes to the PVNGS licensing basis.

Mark-ups of the affected TS pages are provided in Attachment 2 to this enclosure. Clean TS pages are provided in Attachment 3 to this enclosure.

Mark-ups of the affected Technical Specification Bases pages are provided in Attachment 4 for information in support of the TS changes.

3.1. TS 2.1.1.2 – Peak Fuel Centerline Temperature Safety Limit

The restrictions of TS 2.1.1.2 prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. The current TS 2.1.1.2 text is consistent with the Standard Technical Specifications for Combustion Engineering Plants as specified in NUREG-1432 (References 3 and 4).

Centerline fuel melt analyses for Framatome supplied fuel will be performed with COPERNIC best-estimate predictions and nominal fuel rod design parameters consistent with the best estimate fuel temperature relationship, including uncertainty, defined by Equation 12-3 in approved COPERNIC Topical Report BAW-10231(P)(A) (Reference 10).

The proposed change will revise the PVNGS Units 1, 2, and 3 TS to clarify that the current TS 2.1.1.2 is applicable to Westinghouse supplied fuel. The proposed change will also revise the PVNGS Units 1, 2, and 3 TS to specify a peak fuel centerline temperature safety limit for Framatome supplied fuel that is consistent with Equation 12-3 in approved COPERNIC Topical Report BAW-10231(P)(A). Equation 12-3 is presented in units of degrees Centigrade; for consistency with the existing relationship for Westinghouse supplied fuel, this relationship is incorporated into the TS with units of degrees Fahrenheit.

3.2. TS 4.2.1 – Design Features (Fuel Assemblies)

Technical Specification 4.2.1, *Fuel Assemblies*, provides a list of approved fuel rod cladding materials, including Zircaloy, ZIRLO® and Optimized ZIRLO™. Optimized ZIRLO™ was previously added to this list following approval of a permanent exemption request from the requirements of 10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors* and 10 CFR 50 Appendix K, *ECCS Evaluation Models*.

The proposed change will revise the PVNGS Units 1, 2, and 3 TS to allow the use of M5® fuel rod cladding material. This change will be implemented by simplifying the TS 4.2.1 list of fuel rod cladding materials to the phrase “zirconium-alloy clad” rather than continuing to individually list each type of cladding. The phrase “zirconium-alloy” is a generic description that covers the fuel that is licensed at PVNGS.

The proposed change will also revise the PVNGS Units 1, 2, and 3 TS to add a paragraph break prior to the sentence beginning “A limited number of lead test assemblies….”. This paragraph break is appropriate because the use of lead test assemblies is not related to the TS 4.2.1 text characterizing the fuel assemblies.

The proposed change will also revise the PVNGS Units 1, 2, and 3 TS to delete the TS 4.2.1 sentence requiring an approved exemption of use of other cladding material. This sentence is unnecessary because it does not impose any requirement that is not already imposed by 10 CFR 50.46. In addition, the removal of this sentence is consistent with the NUREG-1432, *Standard Technical Specifications, Combustion Engineering Plants* (Reference 3).

3.3. TS 5.6.5.b – Core Operating Limits Report (COLR) Analytical Methods

Technical Specification 5.6.5.b, *Core Operating Limits Report (COLR)*, lists the documents which describe the COLR analytical methods used to determine the core operating limits presented in each PVNGS unit-specific COLR.

The proposed change will revise TS 5.6.5.b by updating the references to be consistent with the analytical methods that will be used to determine the core operating limits following Framatome fuel implementation. The proposed change will add the following NRC approved topical reports to the list of referenced analytical methods for consistency with the analytical methods that will be used to determine the core operating limits following Framatome fuel, critical heat flux (CHF) correlation, and gadolinia burnable absorber methodology implementation:

- EMF-2103P-A, "*Realistic Large Break LOCA Methodology for Pressurized Water Reactors*"
- EMF-2328(P)(A), *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*
- BAW-10231P-A, *COPERNIC Fuel Rod Design Computer Code*
- BAW-10241(P)(A), *BHTP DNB Correlation Applied with LYNXT*
- EPRI-NP-2511-CCM-A, *VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores*

Attachment 5 to this enclosure provides an assessment as to how the conditions and limitations contained in the NRC safety evaluations (SEs) for these topical reports are met.

Topical report CENPD-183-A, *Loss of Flow*, *C-E Methods for Loss of Flow Analysis*, is currently addressed in TS 5.6.5.b as Item 19. Section 6 of Attachment 5 to this enclosure addresses how topical report CENPD-183-A will be implemented with the introduction of Framatome HTP™ fuel.

Clean TS pages are provided in Attachment 3. PVNGS has adopted Technical Specification Task Force (TSTF) 363, *Revise Topical Report References in ITS 5.6.5, COLR*, (Reference 6) in Amendment No. 137 to the PVNGS Operating Licenses (Reference 18); therefore, the proposed change to TS 5.6.5.b, identifies the documents by number and title.

Each PVNGS unit-specific COLR specifies the complete identification (i.e., TS reference number, title, report number, revision, date, and any supplements) for each of the TS referenced topical reports used to prepare the COLR. The following table describes the complete identification for the additional proposed COLR reference additions to TS 5.6.5.b. Each PVNGS unit-specific COLR will be updated as part of the implementation of the approved license amendment.

Proposed TS Ref No.	Title	Report No.	Rev	Date	Suppl
27	*Realistic Large Break LOCA Methodology for Pressurized Water Reactors*	EMF-2103P-A	3	June 2016	N.A.
28	*PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*	EMF-2328(P)(A)	0	March 2001	N.A.
28	*PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*	EMF-2328(P)(A)	0	December 2016	1(P)(A)
29	*COPERNIC Fuel Rod Design Computer Code*	BAW-10231P-A	1	January 2004	N.A.
30	*BHTP DNB Correlation Applied with LYNXT*	BAW-10241(P)(A)	1	July 2005	N.A.
31	*VIPRE-01: A Thermal Hydraulic Analysis Code for Reactor Cores*	EPRI-NP-2511-CCM-A	Mod 02	Volume 1-4 (February 2017), Volume 5 (March 1988)	N.A.

4. REGULATORY ANALYSIS FOR TECHNICAL SPECIFICATIONS CHANGES

4.1. Applicable Regulatory Requirements

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

Title 10 of the Code of Federal Regulations (10 CFR) Paragraph 50.36(c)(2)(ii) requires that TS limiting conditions for operation be established for process variables, design features, or operating restrictions for which a value is assumed as an initial condition of a design basis accident or transient analysis in the licensee's safety analyses. To eliminate the need for an amendment to update the cycle-specific parameter limits for each fuel cycle while complying with 10 CFR 50.36(c)(2)(ii) requirements, the cycle-specific parameter limits are incorporated in the PVNGS unit-specific COLR.

The proposed change adds to TS 5.6.5.b several fuel vendor topical reports which have been reviewed and approved by the NRC for licensing application. The addition of these topical reports to the PVNGS Units 1, 2, and 3 TS is consistent with the current TS 5.6.5.b practice established in Amendment 137 to the PVNGS Operating Licenses (Reference 6, Section 2.5) to identify the documents by number and title. The proposed change to the PVNGS unit-specific COLR specifies the complete identification (i.e., technical specification reference number, title, report number, revision, date, and any supplements) for each of the TS referenced topical reports used to prepare the COLR.

4.2. Precedent

Precedent has been established for the implementation of Framatome fuel in Combustion Engineering pressurized water reactors (PWR). Most recently, St. Lucie Unit 2 was approved for the implementation of Framatome Combustion Engineering 16x16 HTP™ fuel design with M5® as a fuel rod cladding material as documented in ADAMS accession number ML16063A121 (Reference 18). St. Lucie Unit 2 was granted an exemption from the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50 to allow the use of M5® fuel rod cladding as documented in ADAMS Accession Number ML16015A286 (Reference 18).

4.3. Regulatory Discussion of Topical Reports

The proposed implementation of Framatome fuel requires changes to TS 2.1.1.2, TS 4.2.1 and TS 5.6.5.

The restrictions of TS 2.1.1.2 prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. The current TS 2.1.1.2 text is consistent with the Standard Technical Specifications for Combustion Engineering Plants as specified in NUREG-1432 (References 3 and 4). The proposed change will revise the PVNGS Units 1, 2, and 3 TS to clarify that the current TS 2.1.1.2 is applicable to Westinghouse supplied fuel. The proposed change will also revise the PVNGS Units 1, 2, and 3 TS to specify a peak fuel centerline temperature safety limit for Framatome supplied fuel that is consistent with the best estimate fuel temperature relationship, including uncertainty, defined by Equation 12-3 in approved COPERNIC Topical Report BAW-10231(P)(A) (Reference 10).

Technical Specification 4.2.1, *Fuel Assemblies*, provides a list of approved fuel rod cladding materials, including Zircaloy, ZIRLO® and Optimized ZIRLO™. Optimized ZIRLO™ was previously added to this list following approval of a permanent exemption request from the requirements of 10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors*, and 10 CFR 50 Appendix K, *ECCS Evaluation Models*. The proposed change will revise the PVNGS Units 1, 2, and 3 TS to allow the use of M5® fuel rod cladding material. This change will be implemented by simplifying the TS 4.2.1 list of fuel rod cladding materials to the phrase “zirconium-alloy clad.” The proposed change will also revise the PVNGS Units 1, 2, and 3 TS to add a paragraph break prior to the sentence beginning “A limited number of lead test assemblies….” This paragraph break is appropriate because the use of lead test assemblies is not related to the TS 4.2.1 text characterizing the fuel assemblies. The proposed change will also revise the PVNGS Units 1, 2, and 3 TS to delete the TS 4.2.1 sentence requiring an approved exemption of use of other cladding material. This sentence is unnecessary because it does not impose any requirement that is not already imposed by 10 CFR 50.46. In addition, the removal of this sentence is consistent with the NUREG-1432, *Standard Technical Specifications, Combustion Engineering Plants* (Reference 3).

TS 5.6.5.b lists the documents which describe the analytical methods used to determine the core operating limits. The proposed change will revise TS 5.6.5.b by updating the references to be consistent with the analytical methods that will be used to determine the core operating limits following Framatome fuel implementation. The proposed change will add the following topical reports to the list of referenced core operating analytical methods to be consistent with the analytical methods that will be used to determine the core operating limits following Framatome fuel, critical heat flux (CHF) correlation, and gadolinia burnable absorber methodology implementation.

- EMF-2103P-A, *Realistic Large Break LOCA Methodology for Pressurized Water Reactors*
- EMF-2328(P)(A), *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*
- BAW-10231P-A, *COPERNIC Fuel Rod Design Computer Code*
- BAW-10241(P)(A), *BHTP DNB Correlation Applied with LYNXT*
- EPRI-NP-2511-CCM-A, *VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores*

<u>EMF-2103P-A (Realistic Large Break LOCA Methodology)</u>

Topical Report EMF-2103P-A (Reference 7) describes the Framatome methodology developed for the realistic evaluation of a large break loss-of-coolant accident (LBLOCA) for PWRs with recirculation (U-tube) steam generators. Specifically, Westinghouse 3- and 4-loop designs; Combustion Engineering (CE) 2x4 designs; and AREVA 3- and 4-loop designs, all with fuel assembly lengths of 14 feet or less and emergency core cooling system (ECCS) injection to the cold legs, are covered.

APS has demonstrated that the conditions and limitations contained in the NRC SE for this topical report will be met, as described in Attachment 5.

<u>EMF-2328(P)(A) (Small Break LOCA Methodology)</u>

Topical Report EMF-2328(P)(A) (Reference 8) documents use of the S-RELAP5 thermal-hydraulic analysis computer code for analysis of the small break loss of coolant accident (SBLOCA) for Westinghouse and Combustion Engineering PWRs. The ANF-RELAP code was modified to bring it up to a standard that incorporates the thermal-hydraulic code RELAP5/MOD2, the fuel design code RODEX2, the ICECON containment model, and the hot rod model code TOODEE2, into a single system calculation. In so doing, the RELAP5/MOD2 code was modified to include selected models from the RELAP5/MOD3 code, improved neutronics, and models necessary to satisfy the requirements of 10 CFR Part 50, Appendix K.

It is noted that a collection of errata was released in January 2008 to incorporate corrections to the BETHSY testing facility assessment.

Topical Report EMF-2328(P)(A), Supplement 1(P)(A) (Reference 9) provides additional modeling information regarding the manner in which the SBLOCA evaluation model (EM) will treat the following eight areas:

- Spectrum of break sizes
- Core bypass flow paths in the reactor vessel
- Reactivity feedback

- Delayed reactor coolant pump (RCP) trip
- Maximum accumulator / safety injection tank (SIT) temperature
- Loop seal clearing
- Break in attached piping
- Core nodalization

Each issue is explained as to its treatment within the EM and the basis for that treatment, followed by a direct reference to any specific alteration of the treatment described in the main body of Revision 0 of the topical report. These changes are intended to improve the rigor and completeness of the original methodology, while also addressing and resolving several NRC staff issues raised regarding the Framatome small-break methodology over the preceding several years.

There are no conditions and limitations contained in the NRC SE for this topical report supplement.

<u>BAW-10231P-A (COPERNIC Code)</u>

Topical Report BAW-10231P-A (Reference 10) describes the COPERNIC (fuel rod design computer code) that performs the thermal/mechanical analyses necessary to accurately simulate the behavior of a fuel rod during its irradiation. The COPERNIC code is approved for UO_2 licensing applications with advanced cladding material, M5®, to a peak rod average burnup of 62 GWd/MTU.

APS has demonstrated that the conditions and limitations contained in the NRC SE for this topical report will be met, as described in Attachment 5.

<u>BAW-10241(P)(A) (BHTP DNB Correlation)</u>

Topical Report BAW-10241(P)(A) (Reference 11) describes the HTP™ departure from nucleate boiling (DNB) correlation and its proposed use with the LYNXT computer code. The LYNXT form of the HTP™ correlation is referred to as the BHTP CHF correlation. Topical Report BAW-10241(P)(A), Revision 1, addresses the extension of the range of applicability of the independent variables in the BHTP CHF correlation.

It is noted that errata 1P-001 was released in February 2016 to incorporate corrections to the bundle conditions (test conditions) listed for one of the tests identified in Table A.3 of the topical report.

APS has demonstrated that the conditions and limitations contained in the NRC SE for this topical report will be met, as described in Attachment 5.

<u>EPRI-NP-2511-CCM-A (VIPRE-01)</u>

Topical Report EPRI-NP-2511-CCM-A (References 13, 14 and 15), *VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores*, describes the Versatile Internals and Components Program for Reactors; EPRI (VIPRE) computer code to assist utilities in performing detailed thermal-hydraulic analyses of reactor cores. The mathematical modeling used in the code is discussed in Volume 1. Volume 2 is the user's manual and the programmer's manual are contained in Volume 3. Volume 4 documents the experimental data comparisons, sensitivity studies and plant behavior simulations. Input guidelines and capabilities and limitations of the code are presented in Volume 5.

Two versions of VIPRE-01 were submitted to the NRC for review. VIPRE-01 MOD01 was submitted in 1985 and approved in 1986. VIPRE-01 MOD01 is essentially restricted to PWR use only and is considered an outdated and obsolete version.

VIPRE-01 MOD02 was submitted to the NRC in 1990 to address issues raised from the review of MOD01 and to improve some of the limitations on boiling water reactor (BWR) usage. The VIPRE-01 MOD02 SE was issued in 1993. Since there were no substantive modeling changes which would impact PWR calculations, VIPRE-01 MOD02 was approved for PWR applications subject to the same limitations given in the VIPRE-01 MOD01 SE. The MOD02 version inherited the limitations from the MOD01 SE.

APS has demonstrated that the conditions and limitations contained in the NRC SE for these topical reports will be met, as described in Attachment 5.

Assumptions used for accident initiators and/or safety analysis acceptance criteria are not altered by the addition of these topical reports.

Use of the referenced methodologies will support implementation of Framatome fuel. The ability to use either Westinghouse or Framatome fuel will ensure the stability of the fuel supply.

4.4. No Significant Hazards Consideration Determination

Arizona Public Service (APS) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, *Issuance of amendment*, as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes establish a COPERNIC fuel rod design computer code peak fuel centerline temperature safety limit for Framatome HTP™ fuel, allows for the use of M5® fuel rod cladding material by simplifying the TS 4.2.1 list of fuel rod cladding materials to the phrase “zirconium-alloy clad,” and updates the TS 5.6.5.b list of documents describing the core operating limits report (COLR) analytical methods to implement Framatome fuel, BHTP critical heat flux (CHF) correlation, gadolinia burnable absorber, and VIPRE-01 (Versatile Internals and Component Program for Reactors) code methodology.

The requested Technical Specification (TS) changes do not involve any plant modifications that could affect system reliability, component performance, or the possibility of operator error. There is a new time requirement for an existing operator action, but it has been demonstrated to be able to be performed successfully well within the time requirement. The requested TS changes do not affect any postulated accident precursors, do not affect any accident mitigation systems, and do not introduce any new accident initiation methods. The response of the Framatome fuel to postulated accidents has been analyzed using the proposed safety limit, fuel design characteristics, and associated methodologies. These evaluation results show that the fuel response to postulated accidents is within applicable acceptance criteria.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes establish a COPERNIC peak fuel centerline temperature safety limit for Framatome HTP™ fuel, allows for the use of M5® fuel rod cladding material by simplifying the TS 4.2.1 list of fuel rod cladding materials to the phrase “zirconium-alloy clad”, and updates the TS 5.6.5.b list of documents describing the COLR analytical methods to implement Framatome fuel, BHTP CHF Correlation, gadolinia burnable absorber, and VIPRE-01 code methodology.

Physical changes associated with Framatome HTP™ fuel (e.g., M5® cladding, fuel assembly spacer grids, gadolinia as a burnable absorber, MONOBLOC™ construction, FUELGUARD™ lower tie plate) do not introduce any new accident initiators and do not adversely affect the performance of any structure, system, or component previously credited for accident mitigation. Use of Framatome fuel with M5® cladding in Palo Verde Nuclear Generating Station reactor cores is compatible with the plant design and does not introduce any new safety functions for plant structures, systems, or components. The fuel design performs within the fuel design limits.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident than any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed changes establish a COPERNIC peak fuel centerline temperature safety limit for Framatome HTP™ fuel, allows for the use of M5® fuel rod cladding material by simplifying the TS 4.2.1 list of fuel rod cladding materials to the phrase “zirconium-alloy clad,” and updates the TS 5.6.5.b list of documents describing the COLR analytical methods to implement Framatome fuel, BHTP CHF Correlation, gadolinia burnable absorber, and VIPRE-01 code methodology.

The existing TS safety limits for fuel supplied by Westinghouse are not being changed. The proposed COPERNIC peak fuel centerline temperature safety limit provides assurance that Framatome HTP™ fuel fission product barriers will perform within applicable acceptance criteria for normal operation, anticipated operational occurrences, and postulated accidents. The methodology implementing the BHTP CHF correlation for Framatome HTP™ fuel ensures that the applicable margin of safety is maintained (i.e., there is at least 95% probability at a 95% confidence level that the hot fuel rod in the core will not experience departure from nucleate boiling (DNB)).

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

<u>Conclusion</u>

APS concludes that operation of the facility in accordance with the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified. Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

5. ENVIRONMENTAL EVALUATION

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6. PERMANENT EXEMPTION – 10 CFR 50.46 AND 10 CFR PART 50 APPENDIX K FOR M5®

The proposed change will revise the PVNGS Units 1, 2, and 3 TS to allow the use of M5® fuel rod cladding material. Acceptable fuel rod cladding material is identified in PVNGS Units 1, 2, and 3 Technical Specification (TS) 4.2.1, Fuel Assemblies. The proposed change will add M5® fuel rod cladding material as an acceptable material by simplifying the TS 4.2.1 list of fuel rod cladding materials to the phrase “zirconium-alloy clad.” A permanent exemption from certain requirements of 10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors*, and 10 CFR 50, Appendix K, is required to support this change. By letter dated October 14, 2008, the NRC staff approved a temporary exemption from these requirements to support the PVNGS Units 1, 2 and 3 Framatome Lead Fuel Assembly (LFA) program (References 1 and 2). The requested permanent exemption will replace the approved temporary exemption.

Part 50.46(a)(l)(i) of Title 10 of the Code of Federal Regulations (10 CFR 50.46(a)(l)(i)) states in part:

> *"Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties*

> *sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated."*

10 CFR 50.46 continues with a delineation of specifications for peak cladding temperature, maximum hydrogen generation, coolable geometry, and long-term cooling. Since 10 CFR 50.46 specifically refers to fuel with Zircaloy or ZIRLO® cladding and doesn't list M5® cladding, the use of M5® cladding requires a permanent exemption from this section of the regulations.

10 CFR 50, Appendix K, paragraph I.A.5, states in part:

> *"The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation."*

The Baker-Just equation presumes the use of Zircaloy or ZIRLO® cladding. The routine use of M5® cladding requires a permanent exemption from this section of the regulations.

Topical Report EMF-2328(P)(A) (Reference 8) documents analysis of the SBLOCA for Westinghouse and Combustion Engineering PWRs. The SBLOCA topical report acknowledges that the Baker-Just metal-water reaction model is used for oxidation during the transient. Per Topical Report BAW-10240(P)-A (Reference 12), Baker-Just is approved for use with M5®.

Topical Report EMF-2103P-A (Reference 7) describes the Framatome methodology developed for the realistic evaluation of a LBLOCA for PWRs with recirculation (U-tube) steam generators. The realistic LBLOCA topical report acknowledges that energy released through the transient oxidation of cladding is calculated using the Cathcart-Pawel correlation for oxide layer growth, and that the Cathcart-Pawel equation is applicable to all zirconium based cladding alloys currently used, including M5® cladding.

Pursuant to 10 CFR 50.12, *Specific Exemptions*, APS is requesting a permanent exemption from the requirements of 10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors*, and 10 CFR 50, Appendix K, *ECCS Evaluation Models*, for PVNGS Units 1, 2 and 3.

The permanent exemption will allow the use of fuel assemblies manufactured by Framatome with M5® alloy clad fuel rods, consistent with NRC-approved APS and Framatome design and analysis methodologies.

The use of M5® alloy cladding has been approved by the NRC for several US PWRs. Most recently, St. Lucie Unit 1 received approval for the use of M5® in 2014 (Reference 16), and St. Lucie Unit 2 received approval for the use of M5® in 2016 (Reference 18).

<u>10 CFR 50.12, Specific Exemption</u>

The standards set forth in 10 CFR 50.12 provide that the Commission may grant exemptions from the requirements of the regulations for reasons consistent with the following:

- The exemption is authorized by law;
- The exemption will not present an undue risk to the public health and safety;
- The exemption is consistent with the common defense and security; and

- Special circumstances are present.

This exemption is authorized by law. This exemption results in changes to the operation of the plant by allowing the use of M5® as fuel rod cladding material in lieu of Zircaloy or ZIRLO®. As previously stated, 10 CFR 50.12 allows the NRC to grant exemptions from the requirements of 10 CFR Part 50. The NRC staff has previously determined that granting of this type of proposed exemption will not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. Therefore, as required by 10 CFR 50.12 (a)(1), this requested exemption is "authorized by law."

The exemption will not present an undue risk to public health and safety. The M5® zirconium-alloy cladding has very low corrosion and hydrogen pickup rates; providing substantial margin for end of life corrosion and hydrogen limits. This material has been used extensively both in Europe and the United States for fuel rod cladding. The material has been generically reviewed and accepted by the NRC for use on CE fuel designs as addressed in Topical Report BAW-10240(P)-A (Reference 12). Reloads with M5® cladding have been provided in the United States since 2000 and on CE-14 designs since 2006.

The exemption is consistent with the common defense and security. The ability to use either Westinghouse or Framatome fuel will ensure security (i.e., assurance) of the PVNGS fuel supply by providing for multiple fuel vendors with reliable fuel designs and geographically diverse manufacturing facilities. This change in fuel material used in the plant has no relation to plant security issues. Therefore, the common defense and security are not impacted by this exemption request.

Special circumstances are present. As set forth in 10 CFR 50.12(a)(2)(ii), which states that special circumstances are present whenever:

> *"Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule..."*

10 CFR 50.46 identifies acceptance criteria for ECCS system performance at nuclear power facilities. The effectiveness of the ECCS in PVNGS Units 1, 2 and 3 will not be affected by the use of M5® clad fuel assemblies. Due to the similarities in the material properties of the M5® alloy to Zircaloy or ZIRLO® as identified in Topical Report BAW-10240(P)-A (Reference 12) it can be concluded that the ECCS effectiveness would not be adversely affected.

The intent of paragraph I.A.5 of Appendix K to 10 CFR 50 is to apply an equation for rates of energy release, hydrogen generation, and cladding oxidation from metal-water reaction that conservatively bounds all post-LOCA scenarios. The approved Framatome methodology for evaluating SBLOCA events uses the Baker-Just equation, which is approved for use with M5®. The approved Framatome methodology for evaluating realistic LBLOCA events uses the Cathcart-Pawel equation which is applicable to all zirconium based cladding alloys currently used, including M5® cladding.

The regulations of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, make no provision for use of fuel rods clad in a material other than Zircaloy or ZIRLO®. Since the chemical composition of the M5® alloy differs from the specifications for Zircaloy or ZIRLO®, a plant-specific exemption is required to allow the use of the M5® alloy as a cladding material at PVNGS. The expected performance of M5® clad material meets the intent of the regulations, as discussed in the M5® Topical Report (Reference 12). Therefore, application of these regulations in this circumstance would not serve the underlying purpose of the rule and is not necessary to achieve the underlying purpose of the rule, so special circumstances exist.

7. REFERENCES

1. Letter from D. C. Mims (APS) to NRC of March 08, 2008, *Palo Verde Nuclear Generating Station (PVNGS) Unit 1; Docket No. STN 50-528; Request for Temporary Exemption from the Provisions of 10 CFR 50.46 and 10 CFR 50, Appendix K for Lead Fuel Assemblies*, (ADAMS Accession No. ML080790524).
2. Letter from J. R. Hall (NRC) to R. K. Edington (APS) of October 14, 2008, *Palo Verde Nuclear Generating Station, Unit 1 – Temporary Exemption from the Requirements of 10 CFR Part 50, Section 50.46 and Appendix K (TAC No. MD8330)*, (ADAMS Accession Nos. ML082730003 and ML082730006).
3. NUREG-1432, Revision 4.0, Volume 1, *Standard Technical Specifications - Combustion Engineering Plants: Specifications*, (ADAMS Accession No. ML12102A165).
4. NUREG-1432, Revision 4.0, Volume 2, *Standard Technical Specifications - Combustion Engineering Plants: Bases*, (ADAMS Accession No. ML12102A169).
5. SECY-16-0033, *Draft Final Rule – Performance-Based Emergency Core Cooilng System Requirements and Related Fuel Cladding Acceptance Criteria (RIN 3150-AH42)*, March 16, 2016 (ADAMS Accession No. ML15238A933).
6. Technical Specification Task Force (TSTF) Traveler Number 363, Revision 0, *Revise Topical Report References in ITS 5.6.5, COLR*, (ADAMS Accession No. ML040630088).
7. EMF-2103P-A, Revision 3, *Realistic Large Break LOCA Methodology for Pressurized Water Reactors*, June 2016.
8. EMF-2328(P)(A), Revision 0, *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*, March 2001.
9. EMF-2328(P)(A), Revision 0, Supplement 1(P)(A), *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*, December 2016.
10. BAW-10231P-A, Revision 1, *COPERNIC Fuel Rod Design Computer Code*, January 2004.
11. BAW-10241(P)(A), Revision 1 as amended by Errata 1P-001 (February 2016), *BHTP DNB Correlation Applied with LYNXT*, July 2005.
12. BAW-10240(P)-A, Revision 0, *Incorporation of M5® Properties in Framatome ANP Approved Methods*, May 2004.
13. EPRI NP-2511-CCM-A, Mod 02, Revision 3, *VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores*, Volume 1 through 4 (Revision 4, February 2001), and Volume 5 (March 1988), Electric Power Research Institute.
14. Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA Executive Committee), *Acceptance*

for Referencing of Licensing Topical Report, EPRI NP-2511-CCM, "VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores, *Volumes 1, 2, 3 and 4*", May 1, 1986.

15. Letter from A. C. Thadani (NRC) to Y. Y. Yung (VIPRE-01 Maintenance Group), *Acceptance for Referencing of the Modified licensing Topical Report, EPRI NP-2511-CCM, Revision 3, "VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores"*, (TAC No. M79498), October 30, 1993.

16. Letter, Perry H. Buckberg (NRC) to Mano Nazar (NEE), *St. Lucie Plant, Unit No. 2 – Issuance of Amendment Regarding Transitioning to Areva Fuel (CAC No. MF5495)*, April 19, 2016 (ADAMS Accession No. ML16063A121).

17. Letter, Lisa M. Regner (NRC) to Mano Nazar (NEE), *St. Lucie Plant, Unit 1 - Issuance of Amendment Regarding the Use of M5® Alloy Fuel Cladding in Core Reload Applications (TAC No. MF1817)*, March 31, 2014 (ADAMS Accession No. ML14064A129).

18. Letter, Perry H. Buckberg (NRC) to Mano Nazar (NEE), *St. Lucie Plant, Unit No. 2 – Exemption from the Requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50 to Allow the Use of M5® Fuel Rod Cladding (CAC No. MF5494)*, April 19, 2016 (ADAMS Accession No. ML16015A286).

19. Letters, L. Raynard Wharton (NRC) to Gregg R. Overbeck (APS), *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Issuance of Amendements Re: Various Administrative Controls (TAC Nos. MB1668, MB1669, and MB1670)*, October 15, 2001 (ADAMS Accession No. ML012880473) and *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Correction of an Administrative Error Re: Amendments for Various Administrative Controls (TAC Nos. MB1668, MB1669, and MB1670)*, October 22, 2001 (ADAMS Accession No. ML012950413).

ATTACHMENT 1

Regulatory Commitment

ATTACHMENT 1
Regulatory Commitment

The following table identifies a regulatory commitment 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 commits to incorporating into the Time Critical Action Program the requirement to stop all Reactor Coolant Pumps within 5 minutes following pressurizer pressure dropping below the RCP NPSH limits during a Small Break LOCA event. This new time requirement for LOCA mitigation operator action is discussed in section 7.2 of Attachment 10.	X		Upon implementation of Framatome HTP™ Fuel

ATTACHMENT 2

Technical Specifications Page Mark-ups

Affected Pages: 2.0-1, 4.0-1 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,

2.1.1.2.1 The peak fuel centerline temperature for Westinghouse supplied fuel 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 Framatome supplied fuel shall be maintained < 4901°F (decreasing by 13.7°F per 10,000 MWD/MTU for burnup).

2.1.2 Reactor Coolant System (RCS) Pressure SL

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

2.2 SL Violations

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

2.2.2 If SL 2.1.2 is violated:

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

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

4.0 DESIGN FEATURES

4.1 Site Location

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

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies. Each assembly shall consist of a matrix of zirconium-alloy clad ~~Zircaloy or ZIRLO or Optimized ZIRLO~~ fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases.

A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. ~~Other cladding material may be used with an approved exemption.~~

4.2.2 Control Element Assemblies

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

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

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

(continued)

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

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

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

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

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

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

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

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

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.

ATTACHMENT 3

Clean Technical Specifications Pages

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

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

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2.1.1.2.1 The peak fuel centerline temperature for Westinghouse supplied fuel 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 Framatome supplied fuel shall be maintained < 4901°F (decreasing by 13.7°F per 10,000 MWD/MTU for burnup).

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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.

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

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

4.0 DESIGN FEATURES

4.1 Site Location

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

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A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

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.

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(continued)

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

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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]

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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.

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5.6.8 Steam Generator Tube Inspection Report

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a. The scope of inspections performed on each SG.

b. Active degradation mechanisms found.

c. Nondestructive examination techniques utilized for each degradation mechanism.

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)

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

SL 2.1.1.1:

The CPC algorithm uses the CE-1 correlation and the DNBR-Low trip setpoint with an Allowable Value of 1.34. The DNBR limit used in the safety analyses is dependent on fuel type.

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 fuel types may be limiting, the most conservative DNBR analytical limit is used.

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.~~

SL 2.1.1.2:

below the limit

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.

For Westinghouse supplied fuel, the

~~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.

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-10231(P)(A), "Copernic Fuel Rod Design Computer Code," January 2004.

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)

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.

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 the event of a cold leg break, some or all of the inventory of the SIT attached to the faulted leg will spill to containment before it reaches the reactor core. ~~In cold leg breaks, the entire contents of one SIT are assumed to be lost through the break during the blowdown and reflood phases.~~

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump.

(continued)

ATTACHMENT 5

Assessment of Topical Report Limitations and Conditions

ATTACHMENT 5
Assessment of Topical Report Limitations and Conditions

Technical Specification (TS) 5.6.5, Section “b,” *Core Operating Limits Report (COLR)*, lists the documents which describe the Core Operating Limits Report analytical methods used to determine the core operating limits presented in each PVNGS unit-specific COLR.

The proposed changes to TS 5.6.5.b will add the following topical reports to the list of documents describing the analytical methods used to determine the core operating limits following implementation of Framatome HTP™ fuel:

- EMF-2103P-A, *Realistic Large Break LOCA Methodology for Pressurized Water Reactors*
- EMF-2328(P)(A), *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*
- BAW-10231P-A, *COPERNIC Fuel Rod Design Computer Code*
- BAW-10241(P)(A), *BHTP DNB Correlation Applied with LYNXT*
- EPRI-NP-2511-CCM-A, *VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores*

Sections 1 through 5 of this Attachment provide an assessment as to how the conditions and limitations contained in the NRC SEs for these topical reports are met.

Topical Report CENPD-183-A, *Loss of Flow, C-E Methods for Loss of Flow Analysis*, is currently addressed in TS 5.6.5.b as Item 19. Section 6 of this attachment addresses how topical report CENPD-183-A will be implemented with the introduction of Framatome HTP™ fuel.

1. EMF-2103P-A (Realistic Large Break LOCA)

Topical Report EMF-2103P-A, Revision 3, *Realistic Large Break LOCA Methodology for Pressurized Water Reactors*, (Reference 1) describes the Framatome methodology developed for the realistic evaluation of a LBLOCA for PWRs with recirculation (U-tube) steam generators. Specifically, Westinghouse 3- and 4-loop designs; Combustion Engineering (CE) 2x4 designs; and AREVA 3- and 4-loop designs, all with fuel assembly lengths of 14 feet or less and emergency core cooling system (ECCS) injection to the cold legs, are covered.

The conditions and limitations contained in the NRC SE for this topical report are met as discussed in Licensing Report ANP-3639, *Palo Verde Units 1, 2, and 3 Realistic Large Break LOCA Summary Report* (Reference 21, Section 2.7 and Table 3-3) provided in Attachment 9 (non-proprietary) and Attachment 11 (proprietary) to this enclosure.

2. EMF-2328(P)(A) (Small Break LOCA)

2.1. EMF-2328(P)(A), Revision 0

Topical Report EMF-2328(P)(A), Revision 0, *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*, (Reference 2) documents use of the S-RELAP5 thermal-hydraulic analysis computer code for analysis of the SBLOCA for Westinghouse and Combustion Engineering PWRs. The ANF-RELAP code was modified to bring it up to a standard that incorporates the thermal-hydraulic code RELAP5/MOD2, the fuel design code RODEX2, the ICECON containment model, and the hot rod model code TOODEE2, into a single system calculation. In so doing, the RELAP5/MOD2 code was modified to include selected models from the RELAP5/MOD3 code, improved neutronics, and models necessary to satisfy the requirements of 10 CFR Part 50, Appendix K.

It is noted that a collection of errata was released in January 2008 to incorporate corrections to the BETHSY testing facility assessment.

The conditions and limitations contained in the NRC SE for this topical report were negated with the approval of Supplement 1(P)(A). There are no other conditions and limitations contained in the NRC SE for this topical report as discussed in Licensing Report ANP-3640, *Palo Verde Units 1, 2, and 3 Small Break LOCA Summary Report* (Reference 22, Section 3.4) provided as Attachment 9 (non-proprietary) and Attachment 11 (proprietary) to this enclosure.

2.2. EMF-2328(P)(A), Revision 0, Supplement 1(P)(A)

Topical Report EMF-2328(P)(A), Revision 0, Supplement 1(P)(A), *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,* (Reference 3) provides additional modeling information regarding the manner in which the SBLOCA evaluation model (EM) will treat the following eight areas:

- Spectrum of break sizes
- Core bypass flow paths in the reactor vessel
- Reactivity feedback
- Delayed reactor coolant pump (RCP) trip
- Maximum accumulator / safety injection tank (SIT) temperature
- Loop seal clearing
- Break in attached piping
- Core nodalization

Each issue is explained as to its treatment within the EM and the basis for that treatment, followed by a direct reference to any specific alteration of the treatment described in the main body of the Revision 0 topical. These changes are intended to improve the rigor and completeness of the original methodology, while also addressing and resolving several NRC staff issues raised regarding the Framatome small-break methodology over the preceding several years.

There are no conditions and limitations contained in the NRC SE for this topical report as discussed in Licensing Report ANP-3640, *Palo Verde Units 1, 2, and 3 Small Break LOCA Summary Report* (Section 3.4) provided as Attachment 9 (non-proprietary) and Attachment 11 (proprietary) to this enclosure.

3. BAW-10231P-A (COPERNIC Code)

Topical Report BAW-10231P-A, Revision 1, *COPERNIC Fuel Rod Design Computer Code*, (Reference 6) describes the COPERNIC (Fuel Rod Design Computer Code) that performs the thermal/mechanical analyses necessary to accurately simulate the behavior of a fuel rod during its irradiation. The COPERNIC code is approved for UO_2 licensing applications with advanced cladding material, M5®, to a peak rod average burnup of 62 GWd/MTU.

The conditions and limitations contained in the NRC SE for this topical report are met as follows:

Condition and Limitation 1:

Rod Average Burnup

> The staff concludes that the COPERNIC code is acceptable for referencing in licensing applications up to a rod average burnup of 62 GWd/MTU, to the extent specified and under the limitations delineated in the associated topical report and NRC Safety Evaluations.
>
> Safety Evaluation for BAW-10231P-A
> Condition and Limitation 1

The April 18, 2002, Topical Report BAW-10231P-A SE addressed several major areas of the COPERNIC code, including maximum fuel pin centerline temperature, cladding corrosion and hydriding models, irradiation creep, high stress creep model, fuel rod internal pressure, and clad strain. These models were found to be acceptable.

The Framatome Advanced CE-16 HTP™ fuel assembly fuel rod performance was analyzed in accordance with the NRC-approved topical report BAW-10231P-A (Reference 6) utilizing the COPERNIC computer fuel performance code.

The Framatome Advanced CE-16 HTP™ fuel assembly design for PVNGS was analyzed in accordance with the NRC-approved generic mechanical design criteria in EMF-92-116(P)(A) (Reference 7) and in conjunction with NRC-approved topical report BAW-10240(P)(A) (Reference 8). Reference 8 incorporates the M5® cladding material properties that were previously approved by the NRC in BAW-10227(P)(A) (Reference 12) into the Framatome mechanical design methodology (Reference 7). All the mechanical design criteria evaluated by Framatome were shown to be met up to the licensed fuel rod burnup limits of 62 GWd/mtU for UO_2 rods and 55 GWd/mtU for Gd_2O_3 rods.

Condition and Limitation 2:

10 CFR 51.52 Compliance
Licensees that reference this topical report still need to meet 10 CFR 51.52, “Environmental Effects of Transportation of Fuel and Waste – Table S-4.
Safety Evaluation for BAW-10231P-A Condition and Limitation 2

APS will continue to meet the requirements of 10 CFR 51.52, Table S-4. The values in Table S-4 were reviewed generically for Uranium-235 fuel enriched up to 5 weight percent and irradiated up to 62,000 MWd/MTU in NUREG-1437 Volume 1 Addendum 1 (Reference 9, Section 4), and determined to be applicable for the original environmental review completed in NUREG-1437 (Reference 10). Further, as part of the PVNGS license renewal as documented in NUREG-1437 Supplement 43 (Reference 11, Section 6.0), the NRC reviewed APS environmental report APS 2008a and determined there are no impacts related to the uranium fuel cycle beyond those discussed in the generic environmental impact statement.

4. BAW-10241(P)(A) (BHTP DNB Correlation)

Topical Report BAW-10241(P)(A), *BHTP DNB Correlation Applied with LYNXT*, September 2004 (Reference 4) describes the HTP™ departure from nucleate boiling (DNB) correlation and its proposed use with the LYNXT computer code. The LYNXT form of the HTP™ correlation is referred to as the BHTP CHF correlation. Topical Report BAW-10241(P)(A), Revision 1, *BHTP DNB Correlation Applied with LYNXT*, July 2005 (Reference 5), addresses the extension of the range of applicability of the independent variables in the BHTP CHF correlation.

It is noted that errata 1P-001 was released in February 2016 to incorporate corrections to the bundle conditions (test conditions) listed for one of the tests identified in Table A.3 of the topical report.

The conditions and limitations contained in the NRC SE for this topical report are met as follows:

Condition and Limitation 1:

BHTP Variable Ranges

Based on the comparisons with the additional data, the quantitative statistical assurances continue to be met by the correlation in the regions of lower pressure, higher quality and lower mass velocity. Therefore, the independent variables of the BHTP correlation can be extended. The approved extended range is:

System Pressure, psia	1385 to 2425
Mass Velocity, Mlb/hr-ft^2	0.492 to 3.549
Thermodynamic Quality	≤ 0.512

*Safety Evaluation for BAW-10241(P)(A)
Condition and Limitation 1

The BHTP correlation verification for VIPRE-01 and VIPRE-W was based on the same CHF test points used in the testing to develop the HTP™ CHF correlation for use with the thermal-hydraulic subchannel code XCOBRA-IIIC. A VIPRE-01 and VIPRE-W based CHF Design Limit was determined in accordance with Standard Review Plan requirements and followed the method described in the BHTP topical report. The verification of the acceptability for using BHTP with VIPRE-01 and VIPRE-W is based on the adequacy of the correlation to represent the database.

4.1. Predicted to Measured CHF Performance

The test points were evaluated using VIPRE-01 to determine the predicted to measured (P/M) CHF performance of the present BHTP CHF correlation. The P/M values were examined for each independent variable and no biases were found to be introduced. The P/M values are shown with the 95/95 correlation limit lines in Figure 5-1 in Attachment 8 (non-proprietary) and Attachment 10 (proprietary) to this Enclosure.

A histogram of the P/M CHF values for the data points using VIPRE-01 is shown in Figure 5-2 in Attachment 8 (non-proprietary) and Attachment 10 (proprietary) to this enclosure. A normal distribution is provided for comparison.

4.2. Statistical Design Limit

The design limit for departure from nucleate boiling ratio (DNBRL) was calculated to protect 95 percent of the hot pins in the core with 95 percent confidence from departure of nucleate boiling. The DNBRL value for VIPRE-01 (and VIPRE-W) was determined using the same process in the BHTP topical report as addressed in Section 5.4.1 in Attachment 8 (non-proprietary) and Attachment 10 (proprietary) to this enclosure.

4.3. Ranges and Limitations

The BHTP CHF correlation was validated in VIPRE-01 based upon the same data points from Reference 4. The range of applicability was subsequently extended in Reference 5. The extension of the BHTP application ranges for system pressure, mass velocity, and thermodynamic quality using LYNXT is discussed in Reference 5. The use of VIPRE-01 or VIPRE-W local conditions does not alter the conclusions reached for supporting the range extensions; therefore, the BHTP extensions remain applicable when using VIPRE-01 or VIPRE-W. However, since the mass velocity and thermodynamic quality are code dependent, the following Table 4-1 values reflect the use of VIPRE-01 based on the original data points.

Table 4-1: BHTP CHF Correlation Range of Applicability

Independent Variable	Minimum Value	Maximum Value
Pressure, psia	1385	2425
Mass Velocity, 10^6 lbm/hr-ft^2	0.949	3.56
Thermodynamic Quality	No lower limit	0.357

Condition and Limitation 2:

BHTP Extended Variable Ranges

Actions for analyzing the operating conditions outside of the approved ranges of the maximum pressure (2425 psia) but less than 2600 psia are stated below:

- When pressures greater than the pressure limit of 2425 psia but less than 2600 psia are encountered, all of the local coolant conditions are calculated at the upper pressure limit of 2425 psia using the NRC-approved LYNXT thermal-hydraulic code and then used in the calculation of the BHTP CHF.
- Extrapolations below the minimum quality range are performed with no lower limit, consistent with EMF-92-153(P)(A) Revision1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel".

These methods were put forth in [EMF-92-153(P)(A) Revision 1]. Any other extrapolation requires a plant-specific review.

Safety Evaluation for BAW-10241(P)(A)
Condition and Limitation 2

The actions of Condition and Limitation 2 will be followed for pressure between 2425 psia and 2600 psia.

The extension of the BHTP application ranges for system pressure, mass velocity, and thermodynamic quality using LYNXT is discussed in Reference 5. The use of VIPRE-01 or VIPRE-W local conditions does not alter the conclusions reached supporting no lower quality limit; therefore, the BHTP lower quality limit of Table 4-1 remains applicable when using VIPRE-01 or VIPRE-W.

5. EPRI-NP-2511-CCM-A (VIPRE-01 Code)

Topical Report EPRI-NP-2511-CCM-A, *VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores*, (Reference 18) describes the Versatile Internals and Components Program for Reactors; EPRI (VIPRE) computer code to assist utilities in performing detailed thermal-hydraulic analyses of reactor cores. The VIPRE-01 code was developed by the Battelle Northwest National Laboratories under the sponsorship of Electric Power Research Institute (EPRI) (Reference 18).

The mathematical modeling used in the VIPRE-01 code is discussed in Volume 1. Volume 2 is the user's manual and the programmer's manual is contained in Volume 3. Volume 4 documents the experimental data comparisons, sensitivity studies and plant behavior simulations. Input guidelines and capabilities and limitations of the code are presented in Volume 5.

Two versions of VIPRE-01 were submitted to the NRC for review. VIPRE-01 MOD01 was submitted in 1985 and approved in 1986 (Reference 19). VIPRE-01 MOD01 is essentially restricted to PWR use only and is considered an outdated and obsolete version.

VIPRE-01 MOD02 was submitted to the NRC in 1990 to address issues raised from the review of MOD01 and to improve some of the limitations on BWR usage. The VIPRE-01 MOD02 SE was issued in 1993 (Reference 20). Since there were no substantive modeling changes which would impact PWR calculations, VIPRE-01 MOD02 was approved for PWR applications subject to the same limitations given in the VIPRE-01 MOD01 SE.

5.1. EPRI-NP-2511-CCM-A (VIPRE-01 MOD01 Conditions and Limitations)

The Conditions and Limitations contained in the NRC SE for VIPRE-01 MOD01 (Reference 19) are met as follows:

Condition and Limitation 1:

> **Application in Post-CHF Calculations**
>
> The application of VIPRE-01 MOD01 is limited to PWR licensing calculations with heat transfer regimes up to the CHF region. Any use of VIPRE-01 MOD01 in BWR calculations or PWRs in post-CHF calculations will require NRC review.
>
> Safety Evaluation for EPRI-NP-2511-CCM-A, MOD01
> Condition and Limitation 1

VIPRE-01 and VIPRE-W will limit the licensing calculations to heat transfer regimes up to the CHF region, with specific fuel design CHF correlation applied. No post-CHF calculations will be performed with VIPRE-01 or VIPRE-W as explained in Section 5.3.4 in Attachment 8 (non-proprietary) and Attachment 10 (proprietary) to this enclosure. No BWR calculations will be performed.

Condition and Limitation 2:

> **Application of Steady-State CHF Correlations**
>
> Use of a steady-state CHF correlation in VIPRE-01 MOD01 is acceptable for reactor transient analysis provided that the CHF correlation and its DNBR limit have been reviewed and approved by NRC and that the application is within the range of applicability of the correlation including fuel assembly geometry, spacer grid design, pressure, coolant mass velocity, quality, etc. Use of any CHF correlation that has not been approved will require the submittal of a separate topical report for staff review and approval. The use of a CHF correlation that has been previously approved for application in connection with another thermal hydraulic code other than VIPRE-01 will require an analysis showing that, given the correlation data base, VIPRE-01 gives the same or a conservative safety limit, or a new higher DNBR limit must be used, based on the analysis results.
>
> Safety Evaluation for EPRI-NP-2511-CCM-A, MOD01
> Condition and Limitation 2

This submittal is requesting use of the previously approved BHTP CHF Correlation in the VIPRE-01 and VIPRE-W codes for the DNB analysis of Framatome Advanced CE-16 HTP™ fuel above the first HMP grid. For Combustion Engineering 16x16 Standard fuel and Next

Generation Fuel designs, the Westinghouse CHF correlations of CE-1, ABB-NV, and WSSV are used as previously approved. Justification for the use of BHTP within VIPRE-01 and VIPRE-W is shown to give a conservative 95/95 analytical limit with the BHTP SAFDL for VIPRE-01 and VIPRE-W. See Section 5.4 in Attachment 8 (non-proprietary) and Attachment 10 (proprietary) to this enclosure for the analysis of the BHTP CHF correlation performance in the VIPRE-01 and VIPRE-W codes.

Condition and Limitation 3:

Code Limitations

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

Safety Evaluation for EPRI-NP-2511-CCM-A, MOD01
Condition and Limitation 3

No new CHF correlations are being used. This submittal is requesting use of the existing and NRC approved BHTP CHF Correlation in the VIPRE-01 and VIPRE-W codes for the DNB analysis of Framatome Advanced CE-16 HTP™ fuel and retaining the Westinghouse CE-1, ABB-NV, and WSSV correlations as previously approved. Section 5.4 in Attachment 8 (non-proprietary) and Attachment 10 (proprietary) to this enclosure shows the analysis of the BHTP CHF correlation performance in the VIPRE-01 and VIPRE-W codes and the modeling assumptions to be utilized.

Condition and Limitation 4:

Subcooled Boiling Model

For those boiling transients that use profile fit, subcooled boiling models based on steady-state data (such as Levy or EPRI models), the transient time-step control should be such that the Courant limit is not exceeded, i.e., the Courant number is less than 1

Safety Evaluation for EPRI-NP-2511-CCM-A, MOD01
Condition and Limitation 4

The Courant number (Nc) is defined as $N_c = u \times \Delta t / \Delta z$ where u is the axial mass flow velocity, Δt is the transient time step, and Δz is the axial node size. Use of VIPRE-01 in the transient analysis will assure the selection of the axial mass flow velocity, the transient time step, and the axial node size meet this limitation.

Condition and Limitation 5:

> **Quality Assurance Procedures**
>
> The VIPRE-01 user should abide by the quality assurance procedures described in Section 2-6 of (Reference 19)
>
> Safety Evaluation for EPRI-NP-2511-CCM-A, MOD01
> Condition and Limitation 5

The PVNGS quality assurance procedures for analysis and software change and control of usage will be applied. This quality assurance program exceeds the quality assurance procedures described in Section 2-6 of Reference 19.

5.2. EPRI-NP-2511-CCM-A (VIPRE-01 MOD02 Conditions and Limitations)

The conditions and limitations contained in the NRC SE for VIPRE-01 MOD02 (Reference 20) are met as follows:

Condition and Limitation 1:

> **BWR Licensing Applications**
>
> The use of [VIPRE-01] for BWR licensing applications is contingent upon full qualification of the (several models).
>
> Safety Evaluation for EPRI-NP-2511-CCM-A, MOD02
> Condition and Limitation 1

This condition and limitation is not applicable since PVNGS is a PWR.

Condition and Limitation 2:

> **GEXL Correlation**
>
> The GEXL correlation is the only correlation currently having NRC approval for use in CPR calculations of a core containing GE fuels. However, use of the GEXL correlation for other vendors' fuels or any other correlation requires a separate submittal for NRC review and approval.
>
> Safety Evaluation for EPRI-NP-2511-CCM-A, MOD02
> Condition and Limitation 2

This condition and limitation is not applicable since PVNGS does not use the GEXL correlation.

Condition and Limitation 3:

> **Limitations of the Code**
>
> Section 2.2 of Volume 5 (Reference 18) of the submittal identifies a spectrum of limitations of the code. Each user should ensure that the code is not being used in violations of these limitations.
>
> Safety Evaluation for EPRI-NP-2511-CCM-A, MOD02
> Condition and Limitation 3

PVNGS will comply with the VIPRE-01 limitations identified in Topical Report EPRI-NP-2511-CCM-A, Volume 5, Section 2.2. These limitations address the following:

Mathematical Formulation and Empirical Models: VIPRE-01 will not be applied to situations that entail conditions such as low-flow boil-off, annular flow, phase separation involving a sharp liquid/vapor interface, or countercurrent flow.

Thermally Expandable Flow Assumption: VIPRE-01 will not be used to simulate problems involving blowdown or other rapid pressure changes, or for low pressure analyses.

Subchannel Modeling Method: VIPRE-01 will only be used in geometries such as rod or tube bundles, or wherever the lateral flow resistance is large compared to axial flow resistance.

Correlation Databases: VIPRE-01 will not be used for reflood or hot wall rewet problems, or transients. VIPRE-01 will not be used with correlations outside their ranges of applicability.

Condition and Limitation 4:

> **Input Selections for Licensing Applications**
>
> By acceptance of this code version, we [NRC staff] do not necessarily endorse procedures and uses of this code described in Volume 5 as appropriate for licensing applications. As the code developer stated in Reference 5, the materials were provided by the code developers as their non-binding advice on efficient use of the code. Each user is advised to note that values of input recommended by the code developers are for best-estimate use only and do not necessarily incorporate the conservatism appropriate for licensing type analysis. Therefore, the user is expected to justify or qualify input selections for licensing applications.
>
> Safety Evaluation for EPRI-NP-2511-CCM-A
> Condition and Limitation 4

This condition and limitation is addressed in Section 5.2 in Attachment 8 (non-proprietary) and Attachment 10 (proprietary) to this enclosure.

6. CENPD-183-A (Loss of Flow Analysis)

Topical Report CENPD-183-A, *C-E Methods for Loss of Flow Analysis*, (Reference 13) is currently addressed in TS 5.6.5.b as Item 19. This will be implemented differently with the introduction of the Framatome Advanced CE-16 HTP™ fuel design.

Topical Report CENPD-183-A describes the assumptions, conservatisms and basic methods used for analyzing loss of reactor coolant forced flow events. The main body of the report describes a loss of flow analysis method for use with a computer code having transient core thermal hydraulic capabilities (referred to as the dynamic method). The appendix describes a similar loss of flow analysis method for use with a steady state core thermal hydraulic code (referred to as the static method).

The limitations and conditions imposed by the NRC on the loss of flow topical report are identical to the restrictions currently in effect at PVNGS. The conditions and limitations contained in the NRC SE for this topical report are met as follows:

Condition and Limitation 1:

Approved Codes

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

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

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

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

The non-LOCA related conditions imposed by the NRC in the CENPD-183-A SE have been complied with as part of the loss of flow analyses supporting implementation of the Framatome Advanced CE-16 HTP™ fuel design at PVNGS. The computer codes utilized in the Framatome Advanced CE-16 HTP™ fuel design loss of flow analyses differ from those cited in Condition 1. Since the time of submittal of CENPD-183-A (Reference 13), a revised set of computer codes have received NRC approval as alternates for those cited in Condition 1. Specifically, CENTS code (Reference 14) for both the flow coastdown curve and the system response replacing the COAST and CESEC codes, HERMITE code (Reference 15) for the transient's neutronics response, and VIPRE (Reference 16) or CETOP-D code (Reference 17) for the thermal-hydraulic (DNBR) response. VIPRE-01 (Reference 18) for the thermal-hydraulic (DNBR) response is being added by this License Amendment Request.

Condition and Limitation 2:

Assumptions

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

a. The assumptions referred to are:

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

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

The assumptions cited in Condition and Limitation 2 are consistent with those utilized in the PVNGS loss-of-flow analyses.

Condition and Limitation 3:

> **Fuel Damage Probability Distribution for COSMO/W-3**
>
> If COSMO/W-3 is used for DNBR calculations, the applicant is required to submit a fuel damage probability distribution for staff's approval.
>
> Safety Evaluation for CENPD-183-A
> Condition and Limitation 3

COSMO/W-3 is not used for DNBR calculations. The probability density function (pdf) correlation used for PVNGS has been generated with BHTP CHF data. The BHTP fuel failure data were generated inherently in the fuel failure calculation based on the NRC-approved CHF correlation statistics as described in Section 5.4 of Attachment 8 (non-proprietary) and Attachment 10 (proprietary) to this enclosure. The data interface for the rods-in-DNB calculation process no longer requires the DNB pdf data be provided in table form such as Table 2 of CENPD-183-A.

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17. CEN(S)-P, Revision 1-P, *CETOP-D Code Structure and Modeling Methods for San Onofre Nuclear Generating Station Units 2 and 3*, September 1981.
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20. Letter from A. C. Thadani (NRC) to Y. Y. Yung (VIPRE-01 Maintenance Group), *Acceptance for Referencing of the Modified licensing Topical Report, EPRI NP-2511-CCM, Revision 3, VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores, (TAC No. M79498)*, October 30, 1993.
21. Licensing Reports ANP-3639P Revision 0 and ANP-3639NP Revision 0, *Palo Verde Units 1, 2, and 3 Realistic Large Break LOCA Summary Report*, March 2018.
22. Licensing Reports ANP-3640P Revision 0 and ANP-3640NP Revision 0, *Palo Verde Units 1, 2, and 3 Small Break LOCA Summary Report*, March 2018.

ATTACHMENT 6

Affidavits from FRAMATOME
Submitted in Accordance with 10 CFR 2.390
to Consider Attachment 11 as a Proprietary Document

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Philip A. Opsal. I am Manager, Product Licensing, for Framatome Inc., (formally known as AREVA Inc.), and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the Framatome Inc. information contained in the following document: Framatome (formally AREVA Inc.) Document 103-3639P-001, Palo Verde Units 1, 2 and 3 Realistic Large Break LOCA Summary Report, ANP-3639P, Rev. 1, referred to herein as "Document." Information contained in this Document has been classified by Framatome Inc. 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 Inc. 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 Inc. to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome Inc.'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 Inc.

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

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

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(c), 6(d) and 6(e) above.

7. In accordance with Framatome Inc.'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 Inc. only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome Inc.'s 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 16th

day of May, 2018.

Ella Frances Carr-Payne
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/2021
Reg. # 309873

ELLA FRANCES CARR-PAYNE
NOTARY PUBLIC
REG# 309873
MY COMMISSION EXPIRES 08/31/2021
COMMONWEALTH OF VIRGINIA

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Philip A. Opsal. I am Manager, Product Licensing, for Framatome Inc., (formally known as AREVA Inc.), and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the Framatome Inc. information contained in the following document: Framatome (formally AREVA Inc.) Document 103-3640-000, Palo Verde Units 1,2 and 3 Small Break LOCA Summary Report, ANP-3640P, Rev. 0, referred to herein as "Document." Information contained in this Document has been classified by Framatome Inc. 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 Inc. 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 Inc. to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome Inc.'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 Inc.

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

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

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome Inc.'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 Inc. only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome Inc.'s 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 9th
day of March, 2018.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/18
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2018

ATTACHMENT 7

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

AFFIDAVIT

STATE OF ARIZONA)
) ss.
CITY OF PHOENIX)

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

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

3. I am familiar with APS information in the following document: Attachment 10 to the enclosure for APS Correspondence 102-07727, "License Amendment Request for Implementation of Framatome High Thermal Performance 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 natures 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, and Structural Integrity Associates, Inc. (SI) (the NDA is referred to as the "Westinghouse-AREVA-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.

9. 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.

10. 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 6
day of July, 2018.

____
NOTARY PUBLIC, STATE OF ARIZONA
MY COMMISSION EXPIRES:
Reg. #:

Lori A. Peters
Notary Public - Arizona
Maricopa County
Commission # 543079
Expires April 1, 2022

ATTACHMENT 8

Technical Analysis

[NON-PROPRIETARY VERSION]

ATTACHMENT 8

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1. INTRODUCTION AND SUMMARY

The Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 and 3 reactor cores currently consist of 241 fuel assemblies. Two types of Westinghouse supplied fuel rods are currently licensed for use at PVNGS. The first type is Combustion Engineering 16x16 Standard Fuel with ZIRLO® fuel rod cladding material (referred to as CE16STD fuel throughout this document). The second type is Combustion Engineering 16x16 Next Generation Fuel with Optimized ZIRLO™ fuel rod cladding material (referred to as CE16NGF throughout this document).

The current Westinghouse supplied PVNGS fuel assembly designs (CE16STD and CE16NGF), consist of 236 fuel rods, four outer guide tubes, and one center/instrument guide tube. The rods are arranged in a square 16 x 16 array.

The proposed change will support implementation of Framatome Advanced Combustion Engineering 16x16 High Thermal Performance (HTP™) fuel design with M5® as a fuel rod cladding material and gadolinia (Gd_2O_3) as a burnable absorber. The Framatome fuel design is referred to as CE16HTP throughout this document.

The proposed change also supports implementation of Framatome methodologies, parameters and correlations into the approved PVNGS Reload Analysis Methodology (Reference 1.1). This change is to be implemented commencing with the Spring 2020 Unit 2 refueling outage.

The Safety Evaluation approving the PVNGS Reload Analysis Methodology Report (Reference 1.1) addresses a change in fuel vendor. Per the Safety Evaluation, a change in fuel vendor requires *"an evaluation of any changes required to the physics and safety analysis methodology to accommodate that vendor's particular fuel designs. Changes of this type would undergo a thorough engineering evaluation, validation, and verification prior to use of the new fuel design."* This Technical Analysis documents the required engineering evaluation, validation, and verification of the Framatome fuel design and methods.

This Technical Analysis provides a detailed discussion of the design features of the CE16HTP fuel design and of the Framatome methodologies, parameters and correlations. Framatome will retain responsibility for the origination of:

- fuel mechanical design analysis as discussed in Section 2 of this Technical Analysis
- fuel rod behavior (performance) analysis as discussed in Section 4 of this Technical Analysis, and
- ECCS performance analysis as discussed in Section 7 of this Technical Analysis.

APS will retain responsibility for the application of the remainder of the reload methodology, including:

- physics (nuclear) design analysis as discussed in Section 3 of this Technical Analysis,
- core thermal hydraulic design analysis as discussed in Section 5 of this Technical Analysis,
- non-LOCA transient analysis as discussed in Section 6 of this Technical Analysis, and
- COLSS/CPCS setpoints analysis as discussed in Section 11 of this Technical Analysis.

The discussion presented in this Technical Analysis includes a demonstration of the evaluation methodologies performed with a representative core design. This core design was developed to provide key safety parameters to support a transition from Westinghouse fuel (CE16STD or CE16NGF) to the CE16HTP fuel prior to the development of cycle-specific designs. This provides assurance that the plant licensing bases are met for the operation of PVNGS with the CE16HTP fuel during transition and full core cycles.

1.1. Description of the Framatome Fuel Design

The following description of the Framatome CE16HTP fuel design is illustrative of the current fuel design. In the future, the fuel vendor may make design changes that would be incorporated in accordance with approved processes.

The CE16HTP fuel design for PVNGS is the same (with minor changes) as the lead fuel assemblies introduced at PVNGS Unit 1 in Cycle 15 in 2008 (References 1.2, 1.3 and 1.4).

The CE16HTP fuel design for PVNGS is very similar to the Framatome CE-16 HTP™ lead fuel assemblies that operated at San Onofre Unit 2 and shares the same design features currently in use in the Framatome CE-16 HTP™ fuel at St. Lucie Unit 2. Figure 1-1 shows a schematic of the PVNGS CE16HTP fuel assembly.

The Framatome CE16HTP fuel assembly is a 16x16 lattice with 4 large corner guide tubes and one large central instrument tube. The corner guide tube and center instrument tube each occupy 4 fuel rod lattice positions. The fuel rod array contains 236 rods. Some of the rods will contain fuel that has a burnable poison. The fuel rod uses M5® zirconium alloy cladding.

The assembly uses a cage structure (skeleton), with a lower HMP™ spacer grid depicted in Figure 1-2, 9 intermediate HTP™ spacer grids, and a top HTP™ spacer grid with a larger envelope than the intermediate HTP™ spacers. The lower tie plate (LTP) depicted in Figure 1-3 attaches to the cage structure by means of guide tube screws at the four corner guide tube locations. The upper tie plate (UTP) depicted in Figure 1-4 is installed by UTP corner locking nuts onto the guide tube locking sleeves.

Figure 1-1: Framatome Advanced CE-16 HTP™ Fuel Assembly for PVNGS

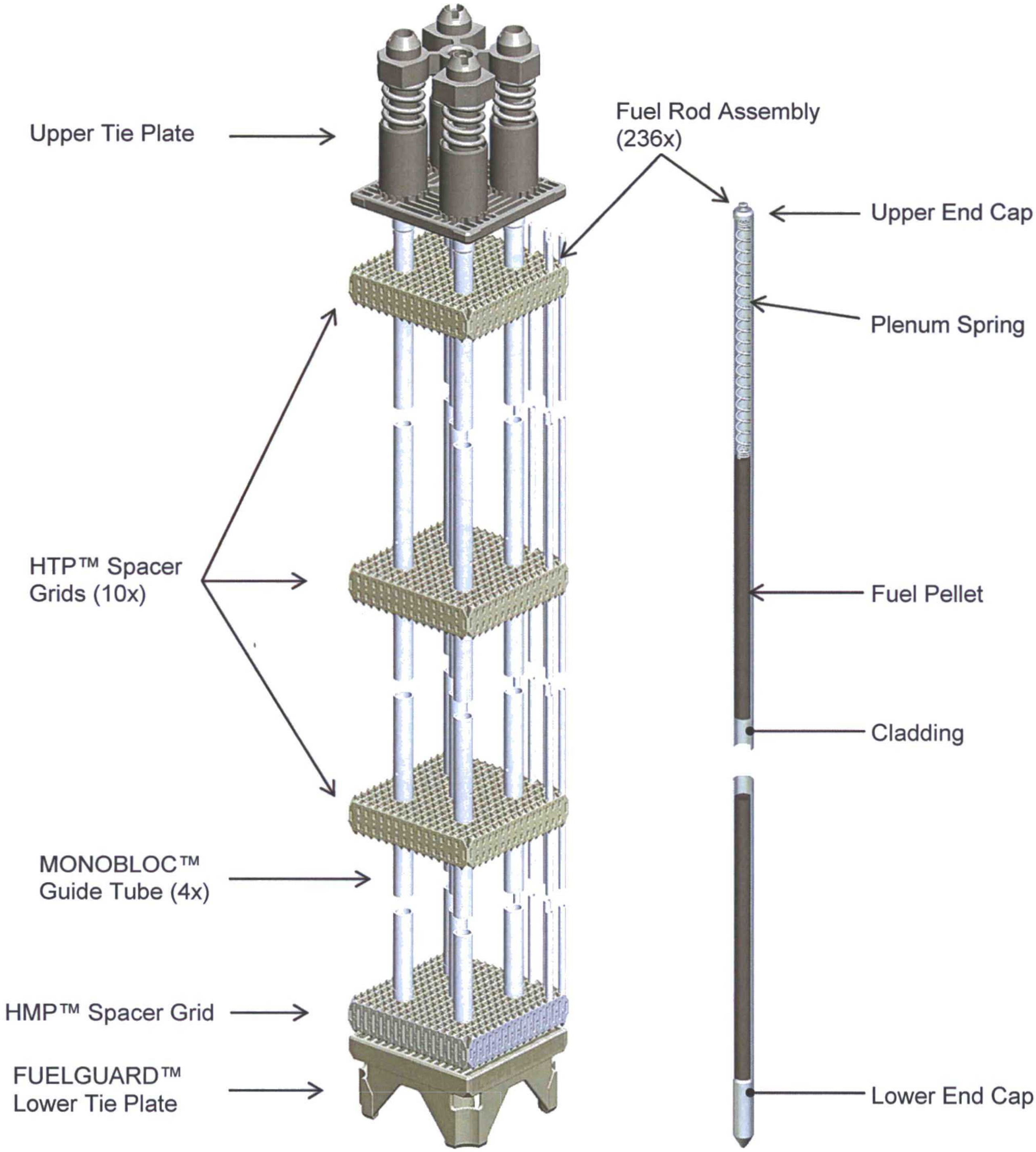


Figure 1-2: PVNGS HTP™ Spacer Grid

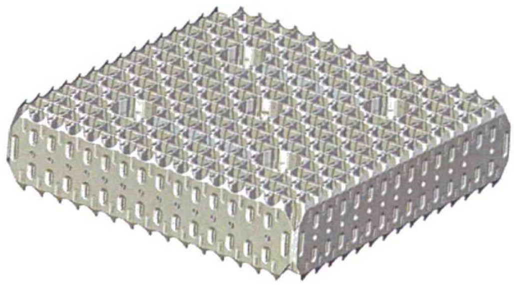


Figure 1-3: PVNGS FUELGUARD™ Lower Tie Plate

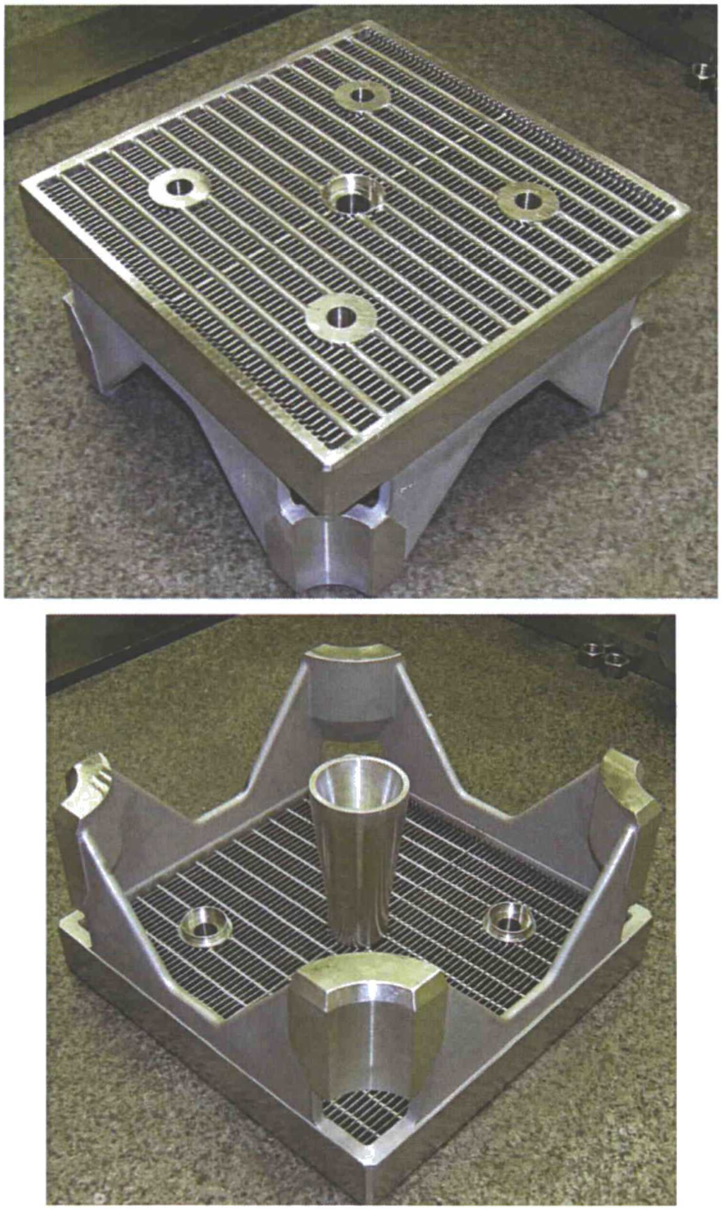
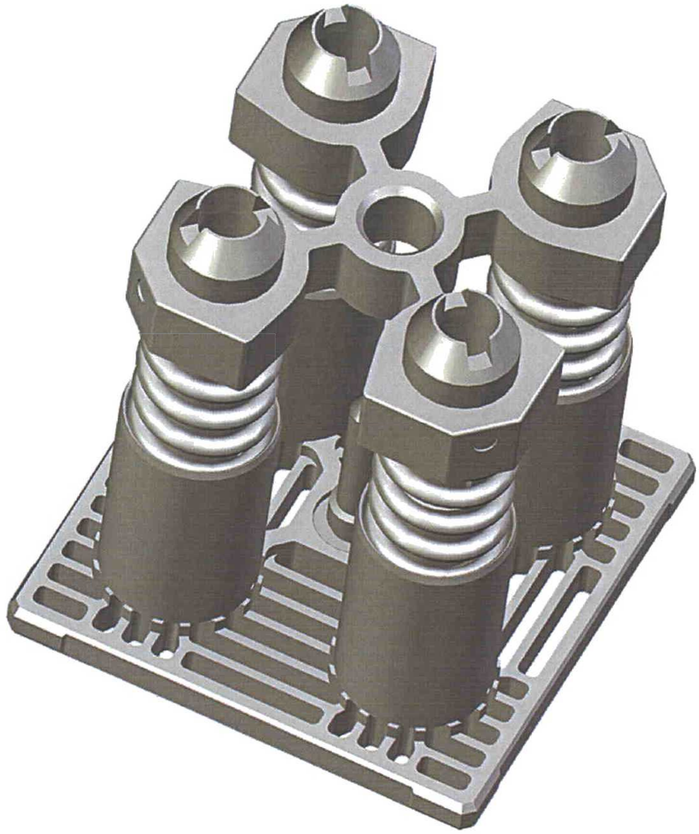


Figure 1-4: PVNGS Upper Tie Plate



1.2. Framatome HTP™ Lead Fuel Assemblies

In support of the proposed change, PVNGS undertook an evaluation of the HTP™ design by installing lead fuel assemblies (LFA). Eight HTP™ LFAs were introduced at PVNGS Unit 1 in Cycle 15 in 2008 (References 1.2, 1.3 and 1.4).

The CE16HTP fuel design for PVNGS is the same (with minor changes) as the lead fuel assemblies. The high level changes to the CE16HTP fuel design for PVNGS relative to the LFA design include the following:

- Incorporated the Framatome chamfered fuel pellet design which reduces pellet chipping during manufacturing activities, thereby improving fuel reliability. The chamfered fuel pellet configuration has been implemented on all US Framatome fuel designs.
- Small change in corner guide tube dashpot elevation to better match the co-resident fuel designs
- Reaction springs were modified to increase the installed spring deflections, producing higher fuel assembly hold-down force to provide positive lift-off margins at maximum reactor coolant pump flows

The eight LFAs completed their third cycle of irradiation in March 2013. Post Irradiation Examinations (PIE) were performed in January 2014 on these assemblies to confirm the in-core performance of the new fuel design and to provide the empirical basis for design licensing. Inspection parameters included a detailed four face visual to assess overall condition, fuel assembly growth, shoulder gap closure, spacer grid growth and oxide levels, fuel rod oxide and diameter, and grid-to-rod fretting (GTRF) wear depth.

The lead fuel assemblies operated three cycles with positive performance as demonstrated in the PIE Campaign. The inspections indicated excellent performance of the fuel design with no evidence of rod bow, a large shoulder gap present, no evidence of handling damage, and a very light layer of crud.

1.3. Overview of Changes to APS Reload Analysis Process and Methodology

Section 3.0 of the approved PVNGS Reload Analysis Methodology (Reference 1.1) provides a general overview of the NRC approved APS reload analysis process and methodology. In general, the requested licensing changes will not significantly change the flow paths or the relationships between APS and fuel vendor processes and procedures. The detailed analyses and their inputs will change in some cases, but the overall process remains the same, except as noted in the following paragraphs.

There will be small changes in the process, and in the fuel vendor interface responsibilities. The Figure 3.0-2 “Simplified Diagram of Reload Analysis Network” in Reference 1.1 has been updated to reflect this license amendment request as provided in Figure 1-5. Two areas of APS responsibility are directly affected by this potential change to Framatome fuel supply. The Core Thermal Hydraulic Analysis scope and the Fuel Rod Performance Analysis scope within the APS responsibilities will change. The indirect impacts on non-LOCA transient analyses performed by APS, including DNBR propagation and fuel failure determination via DNB statistical convolution, are discussed in Section 6.3 of this Attachment.

The current APS Core Thermal Hydraulic Analysis scope will be modified to the extent that [[[redacted]]] for compliance with Technical Specification limits on DNBR. Section 5 of this Attachment discusses the specifics of the new correlation implementation. There will be no change made to the existing DNBR technical specification limit.

The current APS Fuel Rod Performance Analysis scope will be modified [[[redacted]]] between APS and Framatome for Framatome supplied reloads. Currently, APS fully implements [[[redacted]]]. Section 4 of this Attachment discusses the implementation specifics.

Figure 1-5: Simplified Diagram of Reload Analysis Network – Framatome Fuel

[[

[redacted]

]]

1.4. Mixed Core Methodology

The APS reload methodology accounts for the specific fuel design in the areas of core physics, thermal hydraulics, transient analysis, fuel performance, and the COLSS and CPCS setpoint generation. Since each fuel vendor has proprietary design information for their fuel designs, neither vendor can fully analyze a mixed core containing fuel from different vendors. Because APS has access to the proprietary information from both fuel vendors, APS has the ability to explicitly model mixed cores.

The current APS reload methodology is applicable to mixed cores as the methodology:

- accounts for each assembly design in the physics analyses
- accounts for each assembly design in the thermal-hydraulic analyses
- accounts for each design in the fuel performance analyses
- accounts for each design in the transient analyses (for those analyses dependent on fuel assembly design)
- bases the COLSS and CPCS setpoints on the limiting fuel characteristics

Therefore, the existing APS analysis process is a mixed core analysis when multiple fuel designs are present in the core.

In addition to the APS analyses, in a mixed core the fuel vendors will continue to analyze their fuel in their areas of responsibility, utilizing their approved methodologies:

- Mechanical compatibility (See Section 2.2)
- Thermal-Hydraulic compatibility and stability (See Section 5.7)
- Mechanical design, coolability, and seismic performance (See Sections 2.2, 7.1, and 2.3.2)
- Fuel rod bow analysis (See Section 5.7.6)
- LOCA (See Section 7)

1.5. References

1.1. Letter from C. M. Trammell (NRC) to W. F. Conway (APS) of June 14, 1993, *Approval of Reload Analysis Methodology Report – Palo Verde Nuclear Generating Station (TAC Nos. M85153, M85154, and M85155)*.

1.2. BAW-10240(P)(A), Revision 0, *Incorporation of M5® Properties in Framatome ANP Approved Methods*, May 2004.

1.3. Letter from D. C. Mims (APS) to NRC of March 8, 2008, *Palo Verde Nuclear Generating Station (PVNGS) Unit 1; Docket No. STN 50-528; Request for Temporary Exemption from the Provisions of 10 CFR 50.46 and 10 CFR 50, Appendix K for Lead Fuel Assemblies*, (ADAMS Accession No. ML080790524).

1.4. Letter from J. R. Hall (NRC) to R. K. Edington (APS) of October 14, 2008, *Palo Verde*

Nuclear Generating Station, Unit 1 – Temporary Exemption from the Requirements of 10 CFR Part 50, Section 50.46 and Appendix K (TAC No. MD8330), (ADAMS Accession Nos. ML082730003 and ML082730006).

1.5. ANP-2725(P), Revision 1, *Palo Verde Lead Fuel Assemblies Fuel Design Criteria Review*, September 2008.

1.6. BAW-10241(P)(A), Revision 1 as amended by Errata 1P-001 (February 2016), *BHTP DNB Correlation Applied with LYNXT*, July 2005.

2. FUEL MECHANICAL DESIGN ANALYSIS

2.1. Fuel Mechanical Design Introduction

This section evaluates the mechanical design of the Framatome CE16HTP fuel design for the Palo Verde Nuclear Generating Station and its compatibility with the co-resident fuel. Framatome has performed mechanical compatibility evaluations to assure acceptable fit-up with PVNGS reactor core internals, fuel handling equipment, fuel storage racks, and two (2) co-resident fuel types. The first type is CE16STD with ZIRLO® fuel rod cladding material. The second type is CE16NGF with Optimized ZIRLO™ fuel rod cladding material. The Advanced CE-16 HTP™ Lead Fuel Assembly experience at PVNGS was also cited as a demonstration of the Framatome fuel compatibility. A summary of the mechanical compatibility evaluations performed by Framatome is provided in Section 2.2 of this Attachment.

The Framatome CE16HTP fuel assembly design for PVNGS was analyzed in accordance with the NRC-approved generic mechanical design criteria in EMF-92-116(P)(A) (Reference 2.1) in conjunction with NRC-approved topical report BAW-10240(P)(A) (Reference 2.2). Reference 2.2 incorporates the M5® cladding material properties that were previously approved by the NRC in BAW-10227(P)(A) (Reference 2.3) into the Framatome mechanical design methodology (Reference 2.1). All the mechanical design criteria evaluated by Framatome were shown to be met up to the licensed fuel rod burnup limits of 62 GWd/mtU for UO_2 rods and 55 GWd/mtU for Gd_2O_3 rods.

Section 2.2 provides a description of the mechanical compatibility assessments for Framatome fuel with respect to co-resident CE16STD and CE16NGF fuel.

Section 2.3 describes the Framatome mechanical evaluations performed to show acceptability with the NRC-approved generic design criteria.

Section 2.5 provides an overview of fuel operating experience (OE) gained by Framatome with the various CE-16 and CE-14 plants and with the various fuel assembly components.

2.2. Framatome Fuel Mechanical Compatibility

In support of the proposed change, PVNGS undertook an evaluation of the HTP™ design by installing lead fuel assemblies (LFA). Eight HTP™ LFAs were fabricated and introduced at PVNGS Unit 1 in Cycle 15 in 2008 (References 2.9, 2.10 and 2.11). Prior to insertion, the LFAs were shown to be compatible with the host reactor core internals, handling equipment, control element assemblies (CEAs), storage racks, and the co-resident fuel. The LFAs operated for three cycles and demonstrated excellent performance in every key core location to include instrumented locations, CEA locations, and baffle locations. The LFA operating experience has confirmed the results of the Framatome compatibility evaluations. At the time of the LFA program, the co-resident fuel was the CE16STD fuel and Framatome was part of AREVA.

To account for the implementation of CE16NGF co-resident fuel planned for future cycles at PVNGS, Framatome has re-performed the mechanical compatibility evaluations utilizing best practices from the St. Lucie Unit 2 CE-16 HTP™ fuel transition. The general methodology for these evaluations was to use the Framatome fuel assembly design details, available PVNGS plant drawings, co-resident fuel design drawings, and LFA operating experience to assess mechanical compatibility. In some cases, the Framatome fuel design is compared to the co-resident fuel design(s) to assess acceptable interfaces of a proven design in the PVNGS core.

The mechanical compatibility assessments demonstrate that the CE16HTP fuel design for PVNGS is mechanically compatible with the core interfaces, the control components, the plant handling equipment, storage racks, and the co-resident fuel designs (CE16STD and CE16NGF).

Table 2-1 shows a comparison of the major dimensions and features of the CE16HTP design, the CE16STD design, and the CE16NGF design. A summary of the mechanical compatibility evaluations is provided in the following sub-sections. Note that the CE16NGF design has incorporated intermediate flow mixing (IFM) grids and a reduced fuel rod pin diameter of 0.374 inch (versus 0.382 inch) as compared to the CE16STD and CE16HTP fuel designs. The Framatome CE16HTP fuel design does not incorporate IFM grids or reduced diameter fuel rod pins and is essentially the same as the Framatome LFA design.

Table 2-1: Comparison of Nominal Mechanical Design Features

Feature	CE16HTP Design	CE16STD Design	CE16NGF Design
Fuel Assembly Overall Length, inch	[[[redacted]]]	[[[redacted]]]	[[[redacted]]]
Bundle Pitch, inch	8.18	8.18	8.18
Number of Bundles in Core	241	241	241
Fuel Rod Overall Length, inch	161.9	161.9	162.6
Fuel Rod Pitch, inch	0.506	0.506	0.506
Number of Fuel Rods / Assembly	236	236	236
Number of Corner Guide Tubes / Assembly	4	4	4
Number of Center Guide Tubes / Assembly	1	1	1
Fuel Rod Cladding Material	M5®	ZIRLO®	Optimized ZIRLO™
Fuel Rod Cladding Outer Diameter (OD), inch	0.382	0.382	0.374
Fuel Rod Cladding Thickness, inch	0.025	0.025	0.0225
Fuel Pellet Diameter, inch	0.3255	0.3255	0.3225
Fuel Stack Height (BOL, cold), inch	150.00	150.00	150.00
Burnable Poison Material	Gadolinia (Gd_2O_3)	Erbia (Er_2O_3)	IFBA (ZrB_2) Coating
Corner Guide Tube Material	M5®	Stress Relief Annealed Zircaloy-4	Stress Relief Annealed ZIRLO™

Table 2-1: Comparison of Nominal Mechanical Design Features

Feature	CE16HTP Design	CE16STD Design	CE16NGF Design
Center Guide Tube Material	M5®	Stress Relief Annealed Zircaloy-4	Stress Relief Annealed ZIRLO™
Number of Structural Grids	11	11	11
Number of Intermediate Flow Mixing Grids	0	0	2
Bottom Grid	Alloy 718 (HMP™)	Inconel 625 (GUARDIAN™)	Inconel 625 (GUARDIAN™)
Upper Grids	M5® (HTP™)	Zircaloy-4 Wavy grid (Mid Grids) Inconel 625 (Top Grid)	Low-Tin ZIRLO™ (Mid Grids) Inconel 718 (Top Grid)

2.2.1. Fuel Assembly

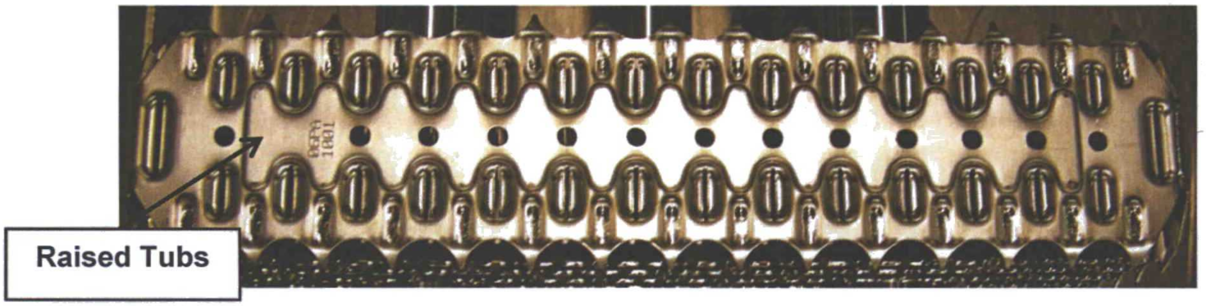
For the LFA design, the fuel assembly overall length was re-confirmed to be compatible with the dimensions of the core internals (spacing between the lower support structure and upper guide structure) at beginning of life (BOL) cold and hot conditions. Additionally, positive engagement of the upper tie plate corner locking nuts / reaction plate and the upper guide structure was demonstrated. An axial growth analysis confirmed adequate assembly to core internals and fuel rod / fuel assembly differential growth margins up to the licensed fuel rod and fuel assembly burnup limits.

The significant change relative to the Framatome LFA experience at PVNGS related to co-resident fuel mechanical compatibility is the introduction of the CE16NGF fuel design. The changes to the CE16NGF fuel affecting the mechanical compatibility evaluations include: slight change to spacer grid centerline elevations, addition of two (2) intermediate flow mixing (IFM) grids, and a reduced fuel rod diameter resulting in reduced fuel assembly weight. These differences have been accounted for in the fuel assembly compatibility evaluations. The results demonstrate acceptable structural spacer grid overlap through-out the fuel design life in a mixed core environment. The thermal-hydraulic impact of the CE16NGF IFMs on the Framatome fuel assembly has been addressed in a flow-induced vibration assessment and the crossflow velocities reported in Section 5.7.3 are within the HTP™ product experience base.

The array type, the number of fuel rods and guide tubes, and the fuel rod pitch dimensions of the Framatome LFA design are the same as the co-resident fuel designs.

The square and diagonal envelopes of the fuel assembly at the upper and lower tie plates and spacer grids were confirmed to be compatible with the core internals and co-resident fuel. The envelopes are unchanged from the Framatome LFA design. The Framatome top HTP™ spacer grid square envelope is slightly larger than the co-resident designs. This was done during the Framatome LFA program to ensure that during core on-load, the fuel assemblies could not lean to the extent that would cause interface issues with the upper guide structure and upper tie plate corner locking nut lead-in features. The Framatome top HTP™ spacer grid is the same as all other HTP™ spacer grids within the cage assembly, except that the side plates have raised tubs along the center line that project roughly [[[redacted]]] beyond the typical side plate outer surface. The remaining design features and manufacturing processes are the same as the other HTP™ spacer grids. Figure 2-1 shows the raised tubs on the top HTP™ spacer grid side plate.

Figure 2-1: Top HTP™ Spacer Grid Side Plate



These evaluations demonstrate that the Framatome fuel assembly design is compatible with the reactor components and co-resident fuel in the core.

2.2.2. Upper Tie Plate

The mechanical compatibility of the UTP is explicitly evaluated because it interfaces with the guide pipes in the upper guide structure (UGS), interfaces with all the fuel assembly grapples when moving the fuel assembly, and interfaces with the control element rods.

The UTP evaluations show that the UTP is mechanically compatible with the UGS, the fuel handling equipment, and control rods. A prototype upper tie plate was tested at the reactor site with plant equipment prior to the LFA delivery. The UTP is unchanged from the Framatome LFA program except for the reaction springs. The reaction springs were modified to increase the installed spring deflections to produce higher fuel assembly hold down force. This change is within the Framatome experience base in terms of spring deflection and load (based on the springs for the San Onofre LFAs). This change has been incorporated into all of the impacted analyses.

2.2.3. Lower Tie Plate

The mechanical compatibility of the LTP is important since it mates with the lower support structure features (alignment pins) and provides the lead-in and entry of the in-core detectors. The Framatome LTP envelope is slightly smaller [[[redacted]]] than that of the co-resident design which facilitates adjacent fuel assembly handling operations. The Framatome LTP is unchanged from the LFA program.

The PVNGS lower support structure is unique as compared to other CE14 and CE16 plants. The lower support structure for the fuel assemblies is a lattice of support beams located approximately two feet above the core internals. At the intersections of the support beams, an alignment pin is positioned for the fuel assembly interface. The alignment pins interface with the fuel assembly LTP at the corners where one pin is shared with four fuel assemblies (for inboard locations). The in-core instrumentation (ICI) is inserted through the center of the LTP, which has a conical shaped boss that facilitates the lead-in features and centering of the ICI. These critical interfaces have been successfully evaluated for proper engagement and lead-in capability. Additionally, the Framatome LTP footprint, interface with handling auxiliary equipment (e.g., lead-in shoes), and off-index assessments have been performed with acceptable results.

2.2.4. Guide Tubes

Besides being the structural components of the fuel assembly, the guide tubes interface with the control rods and in-core instrumentation. The mechanical compatibility of the guide tubes was divided into two parts since the corner guide tubes only interface with the control rods and the center guide tube / instrument tube only interfaces with the in-core instrumentation. The corner guide tubes are unchanged from the Framatome LFA program except for slightly lowering the dashpot elevation and allowing a more gradual inner diameter transition at the top transition. These changes were incorporated into the impacted evaluations. The instrument tube is dimensionally unchanged from the Framatome LFA program.

The Framatome corner guide tube design maintained the upper expanded region, as was done for the LFA design, which provides a larger annulus between the control rod outer diameter and the corner guide tube inner diameter in the parked elevation as compared to the NGF design. The NGF corner guide tube design has changed relative to the co-resident fuel (Westinghouse CE16STD) used to develop the Framatome LFA design. The Westinghouse CE16NGF design does not have an expanded inner diameter region at the top of the corner guide tube like its predecessor to accommodate optional inner wear sleeves.

The remaining inner diameters (central zone and dashpot region) and weep hole diameters are consistent with the co-resident fuel designs. The axial locations of the guide tube dashpot and weep holes are also similar to the co-resident designs. These critical dimensions assure that control element drop times and guide tube cooling are not significantly affected by the introduction of the Framatome fuel assemblies. The other difference is that the Framatome design uses MONOBLOC™ corner guide tubes which have a constant outer diameter as discussed in Section 1.1.

The center guide tube / instrument tube provides guidance for in-core instrumentation. [[[redacted]]]. The Framatome instrument tube is consistent with the co-resident design.

2.3. Mechanical Design Evaluations

The mechanical design evaluations are performed using NRC-approved design methods and evaluated to NRC-approved generic design criteria (Reference 2.1). These generic criteria are consistent with the specified acceptable fuel design limits (SAFDLs) identified in Chapter 4.2 of the Standard Review Plan (SRP) (Reference 2.7). The NRC-approved generic design criteria were used to assess the performance of the fuel assemblies under steady-state and faulted conditions were developed to satisfy certain objectives (Reference 2.1). The use of M5® cladding and guide tubes required that the Framatome design methods be modified to incorporate M5® properties and the generic design criteria be evaluated to assure continued applicability. This implementation was documented in Reference 2.2 and generically reviewed and accepted by the NRC. The fuel rod performance was analyzed in accordance with NRC-approved Framatome reload licensing criteria and methods (References 2.1 through 2.6). The COPERNIC and CROV computer codes (References 2.4 and 2.6, respectively) were used to perform the fuel rod thermal mechanical calculations.

The fuel mechanical analyses are broadly separated into fuel rod analyses and fuel assembly structural analyses. The input parameters used to perform the mechanical analyses included fuel design information derived from design documents, fuel assembly and component characteristics established by mechanical / hydraulic testing, plant parameters, fatigue duty cycles created using the fatigue transients provided in the PVNGS UFSAR, and fuel rod power histories generated for the representative cycles.

2.3.1. Fuel Rod Analyses

The fuel rod analyses include evaluations of the SAFDLs such as internal rod pressure, cladding creep collapse, Transient Cladding Strain (TCS), Centerline Fuel Melt (CFM), cladding fatigue, and cladding corrosion.

The fuel rod mechanical performance evaluations are dependent on the rod power. For the representative fuel cycles analyzed for this license amendment request, the power histories were created using expected typical cycle core designs projected to the design life of the fuel. For fuel reload applications, the approved methodology (Reference 2.4) allows using single rod power histories for the evaluation. The license amendment request representative fuel cycles are analyzed to demonstrate that the fuel design is acceptable and provides typical results showing SAFDL compliance. The specific reload results could be slightly different but will continue to show SAFDL compliance.

2.3.2. Fuel Assembly Structural Analyses

The fuel assembly structural analysis is separated into normal operating analysis, shipping and handling analysis, and faulted condition analysis. The normal operating analysis evaluates the fuel assembly stress state during start-up, steady state operation, shutdown, and Anticipated Operational Occurrences (AOOs) and compares it with the criteria established in EMF-92-116(P)(A) (Reference 2.1). The shipping and handling analysis evaluates the fuel assembly against handling limits established in EMF-92-116(P)(A) and shipping load limits established in the Framatome shipping container specifications. The Safety Evaluation Report (SER) for EMF-92-116(P)(A) approved the usage of this Topical Report for PWR licensing applications up to 62 GWd/mtU rod-average burnups with no restrictions or conditions.

The faulted condition analysis evaluates the structural response of the fuel assembly to externally applied forces such as earthquakes and postulated pipe breaks in the reactor coolant system. The faulted condition analysis was performed in accordance with the methodology and criteria established in ANP-10337P-A (Reference 2.8).

2.3.2.1. Additional Discussion of the Faulted Condition Analysis

Beyond the methods defined in ANP-10337P-A (Reference 2.8), additional adjustments were made to the predicted structural impact loads on the Advanced CE-16 HTP™ vertical model. For the vertical simulation, the Framatome fuel assembly model is excited by the vertical motions of the core plates. In the case of the vertical LOCA simulation, hydraulic forces acting directly on the fuel assembly are also considered. The Framatome fuel model from Reference 2.8 is conservative in that it does not explicitly represent the finite mass and stiffness of the lower core plate and other reactor internal connections. Consideration of these effects would result in lower impact loads. In contrast, the Framatome model considers infinitely rigid core plates with prescribed and unaltered motions, which effectively assigns infinite mass to the core plate as well. As a result, the Framatome model predicts vertical impact loads that greatly exceed the comparable loads on the co-resident fuel design that has been evaluated with a system model that includes a detailed representation of the reactor internals. The Framatome model predicts impact loads that are more than four (4) times the magnitude of the loads predicted with the system model.

Accounting for the finite mass of the reactor internals that participate in impacts with the fuel assembly, the predicted impact loads are reduced by at least a factor of two (2). Therefore, the vertical loads predicted for the Advanced CE-16 HTP™ with the Framatome simplified model are reduced by a factor of two (2). The adjusted loads remain conservatively higher than the loads predicted for co-resident fuel with similar mass and stiffness by more than a factor of two (2).

2.3.2.2. Fuel Assembly Damping Values

The damping ratios used for the horizontal core row model are defined in Sections 6.1.3.1 and 6.1.3.2 of ANP-10337P-A (Reference 2.14) for the BOL and EOL conditions, respectively. The values used for the viscous damping ratios for the vertical accident model are defined in Section 6.2.6 of ANP-10337P-A for OBE and SSE which are based on the damping recommendations provided in Revision 1 to Regulatory Guide (RG) 1.61 (Reference 2.8) for use in dynamic analysis of nuclear power plant structures.

The PVNGS plant design currently conforms to the guidelines in Revision 0 to RG 1.61. The use of the damping recommendations provided in Revision 1 to RG 1.61 will be identified in UFSAR Section 1.8 as a deviation from Revision 0 to RG 1.61.

2.3.2.3. Faulted Condition Topical Report Conditions and Limitations

The Conditions and Limitations contained in the NRC SE for topical report ANP-10337P-A are met as follows:

Condition and Limitation 1:

> **Dynamic Grid Crush Tests**
>
> Dynamic grid crush tests, must be conducted in accordance with Section 6.1.2.1 of ANP-10337P (as amended by RAI 16), and spacer grid behavior must satisfy the requirements in the TR, the key elements of which are:
>
> a. [[[redacted]]].
>
> b. [[[redacted]]].
>
> c. [[[redacted]]].
>
> Safety Evaluation for ANP-10337P-A
> Condition and Limitation 1

Framatome has conducted the necessary dynamic grid crush testing to demonstrate the behavior defined in items a, b, and c.

Condition and Limitation 2:

> **Allowable Grid Permanent Deformation**
>
> For fuel assembly designs where spacer grid applied loads are limited based on allowable grid permanent deformation (as opposed to buckling), the following limits from Table 4-1 of the TR apply:
>
> a. For all OBE analyses, allowable spacer grid deformation is limited to design tolerances and [[[redacted]]].
>
> b. For SSE, LOCA, and combined SSE+LOCA analyses, [[[redacted]]].
>
> Safety Evaluation for ANP-10337P-A
> Condition and Limitation 2

Framatome has defined allowable spacer grid deformation limits that are in accordance with items a and b.

Condition and Limitation 3:

Use of CASAC and ANSYS

The modification or use of the codes CASAC and ANSYS (or other similar industry standard codes) are subject to the following limitations:

a. CASAC computer code revisions, necessitated by errors discovered in the source code, needed to return the algorithms to those described in ANP-10337P (as updated by RAIs) are acceptable.

b. Changes to CASAC numerical methods to improve code convergence or speed of convergence, transfer of the code to a different computing platform to facilitate utilization, addition of features that support effective code input/output, and changes to details below the level described in ANP-10337P would not be considered to constitute a departure from a method of evaluation in the safety analysis. Such changes may be used in licensing calculations without NRC staff review and approval. However, all code changes must be documented in an auditable manner to meet the quality assurance requirements of 10 CFR Part 50, Appendix B.

c. ANSYS or other industry standard codes may be used if they are documented in an auditable manner to meet the quality assurance requirements of 10 CFR Part 50, Appendix B, including the appropriate verification and validation for the intended application of the code.

Safety Evaluation for ANP-10337P-A
Condition and Limitation 3

Framatome has applied code versions of CASAC and ANSYS consistent with those that were reviewed and approved in ANP-10337P.

Condition and Limitation 4:

> **Applicable to Current Fleet of PWRs**
>
> This methodology is limited to applications that are similar to the current operating fleet of PWR reactor and fuel designs. The core geometry should be comparable to the current fleet, in terms of dimensions, dimension tolerances, fuel assembly row lengths, and the gaps between fuel assemblies. Fuel designs should be comparable to the current fleet, in terms of materials, geometry, and dynamic behaviors.
>
> Safety Evaluation for ANP-10337P-A
> Condition and Limitation 4

The Palo Verde reactors are part of the "current fleet" of PWRs in place at the time of approval of ANP-10337P.

Condition and Limitation 5:

> **Generic Fixed Damping Values**
>
> ANP-10337P established generic fixed damping values intended to be used for all PWR designs. All applications of this methodology to new fuel assembly designs must consider the continued applicability of the fixed damping values of this methodology. If new materials, new geometry, or new design features of a new fuel assembly design may affect damping, additional testing and/or evaluation to determine appropriate damping values may be required.
>
> Safety Evaluation for ANP-10337P-A
> Condition and Limitation 5

This License Amendment Request addresses the application of an existing HTP™ fuel design to an existing reactor that is part of the "current fleet" of PWRs. Hence, the application of the generic damping values from ANP-10337P falls within the range of intended application.

Condition and Limitation 6:

> **Fuel Rod Performance Under Externally Applied Loads**
>
> The ANP-10337P methodology includes the generation of fuel rod loads, but does not provide a means to demonstrate compliance for fuel rod performance under externally applied loads (to applicable acceptance criteria). Applications of this methodology must provide an acceptable demonstration of fuel rod performance.
>
> Safety Evaluation for ANP-10337P-A
> Condition and Limitation 6

The performance of the CE16HTP fuel rods for Palo Verde is evaluated in the same manner as demonstrated in the sample problem for ANP-10337P.

Condition and Limitation 7:

> **Component Stresses**
>
> As indicated in ANP-10337P when orthogonal deflections from separate core locations are artificially superimposed to calculate component stresses, the component stresses must be compared against the design criteria associated with control rod positions.
>
> Safety Evaluation for ANP-10337P-A
> Condition and Limitation 7

The analysis performed under ANP-10337P appropriately considers the guide tube criteria for control rod positions.

Condition and Limitation 8:

> **Combination of Loads for Non-Grid Component Evaluation**
>
> In accordance with RG 1.92, the combination of loads for non-grid component evaluation should ideally be based on three orthogonal components (two horizontal and one vertical). [[[redacted]]].
>
> Safety Evaluation for ANP-10337P-A
> Condition and Limitation 8

The analysis performed under ANP-10337P is performed in accordance with RG 1.92 and combines load based on three-orthogonal components.

Condition and Limitation 9:

Grid Impact Model

[[[redacted]]].

Safety Evaluation for ANP-10337P-A
Condition and Limitation 9

The grid impact loads predicted by Framatome [[[redacted]].

2.3.3. Conformance with Mechanical Design Criteria

The generic design criteria (SAFDLs) for the fuel rod and fuel assembly are listed in Table 2-2 along with the corresponding section number from the criteria topical report (Reference 2.1) and the representative fuel cycle results. Table 2-3 and Table 2-4 present the Centerline Fuel Melt (CFM) and Transient Cladding Strain (TCS) linear heat generation rate (LHGR) limits, respectively, from the fuel rod analyses.

The fuel rod analysis criteria derived from References 2.4 through 2.6 are specifically noted as replacing the corresponding criteria from Reference 2.1 and some of the criteria are modified to incorporate M5® material properties for the cladding and structural material in accordance with the Reference 2.2 NRC-approved topical report. In some cases, the criteria specified in the tables are addressed in analyses other than the mechanical design evaluations. The Table 2-2 results represent the current representative fuel analyses of record; these results could vary slightly with future reloads.

The CE16HTP fuel design for PVNGS has been analyzed in accordance with NRC-approved mechanical design criteria using representative fuel cycle inputs. The analyses demonstrate that the fuel rod design criteria are satisfied for the representative fuel design under normal and faulted operating conditions to a UO_2 rod average burnup of 62 GWd/mtU and a Gd_2O_3 rod average burnup of 55 GWd/mtU in accordance with the SER for the NRC-approved COPERNIC topical report BAW-10231P-A (Reference 2.4).

Table 2-2: Fuel Design Evaluation Results

Criteria Section	Description	Criteria	Results
3.2	Fuel Rod Criteria		
3.2.1	Internal Hydriding	Hydrogen content in components controlled to a minimum level during manufacture to limit internal hydriding.	Controlled by manufacturing specifications and verified by Quality Control inspection.
3.2.2	Cladding Collapse	Creep collapse life must exceed maximum expected in-core life. (Reference 2.6)	Criteria met.
3.2.3	Overheating of Cladding	95% probability at 95% confidence that fuel rods do not experience DNB during steady state or AOOs.	The results for this analysis will be verified for each reload core design. See Section 5.
3.2.4	Overheating of Fuel Pellets	No fuel centerline melting during normal operation and AOOs. (Reference 2.4)	The results for this analysis will be verified for each reload core design. See Table 2-3 for centerline fuel melt linear heat rate (LHR) limits.
3.2.5	Stress and Strain Limits		
	Pellet / Cladding Interaction	M5® cladding uniform hoop strain < 1%. (Reference 2.4)	The results for this analysis will be verified for each reload core design. See Table 2-4 for transient cladding strain LHR limits and Section 4 for Fuel Rod Performance Analysis.
	Cladding Stress	ASME Section III, Division 1, Article III-2000, in combination with the specified 0.2% offset yield strength of the unirradiated cladding. M5® stress limit based on bi-axial burst strength of cladding (Reference 2.2) and buckling criteria at limiting overpressure at BOL (Reference 2.3).	Criteria met.
3.2.6	Cladding Rupture	The calculations with the deformation models must satisfy the event criteria in 10 CFR 50.46.	Clad rupture effects are incorporated in the LOCA licensing results. See Section 7.
3.2.7	Fuel Rod Mechanical Fracturing	ASME Section III, Division 1, Article III-2000, in combination with the specified 0.2% offset yield strength of the unirradiated cladding. M5® stress limit based on bi-axial burst strength of cladding.	Criteria met.

Table 2-2: Fuel Design Evaluation Results

Criteria Section	Description	Criteria	Results
3.2.8	Fuel Densification and Swelling	See Cladding Collapse, Overheating of Fuel Pellets, Pellet/Cladding Interaction, and Rod Internal Pressure.	Fuel densification and swelling models are included in NRC-approved fuel performance codes. See Cladding Collapse, Overheating of Fuel Pellets, Pellet/Cladding Interaction, and Rod Internal Pressure in this table. Criteria met.
3.3	Fuel System Criteria		
3.3.1	Stress, strain, and loading limits on assembly components. (See Item 3.3.9 in this table for handling and Item 3.4 in this table for accident conditions.)		
	Guide Tube	ASME Section III, Division I, Article III-2000 for Normal Operation.	Criteria met.
	Spacer Grid	Lateral load < load limit.	Criteria met.
	Upper and Lower Tie Plates	Limiting loads occur during handling (and shipping).	Criteria met.
	Accident Conditions	Maintain coolable geometry and ability to insert control rods. SRP 4.2 Appendix A and ASME Section III, Division 1, Article III-2000, Appendix F	See Section 2.3.2 for fuel assembly structural analysis.
3.3.2	Cladding Fatigue	Cumulative usage factor (CUF) [[[redacted]]] (References 2.2 and 2.3).	Criteria met.
3.3.3	Fretting wear	No fuel rod failures due to fretting wear.	Criteria met.
3.3.4	Oxidation, Hydriding, and Crud Buildup	Acceptable maximum oxide thickness. Best estimate oxide < 100 microns. Effects of oxidation and crud included in thermal and mechanical fuel rod analyses. Stress analysis to include metal loss due to oxidation (Reference 2.2).	Criteria met.
3.3.5	Rod Bow	Lateral displacement of the fuel rods shall not be of sufficient magnitude to impact thermal margins.	Section 5.7.6 demonstrates that no rod bow penalty is required.
3.3.6	Axial Irradiation Growth		

Table 2-2: Fuel Design Evaluation Results

Criteria Section	Description	Criteria	Results
	Fuel Rod	Clearance remains between fuel rod and UTP/LTP at EOL.	Criteria met.
	Fuel Assembly	The fuel assembly length shall not exceed the minimum space between upper and lower core plates in the cold condition at EOL.	Criteria met.
3.3.7	Rod Internal Pressure	Allowable internal pressure not to exceed system pressure [[[redacted]]]. When internal pressure exceeds system pressure, pellet-to-clad strain ratio ≤ 1 (References 2.4 and 2.5)	Criteria met.
3.3.8	Assembly Liftoff	No liftoff from core lower support.	Criterion is met for operation and 4th reactor coolant pump startup at 500 °F.
3.3.9	Fuel Assembly Handling	Assembly withstands 2½ times the weight as a static force.	Criteria met.
3.4	Fuel Coolability		
3.4.1	Cladding Embrittlement	Included in LOCA analysis.	Cladding embrittlement is satisfied by meeting the 10 CFR 50.46(b) acceptance criteria. See Section 7.
3.4.2	Violent Expulsion of Fuel	< 280 cal/gm energy deposition.	See Section 6.5.
3.4.3	Fuel Ballooning	Consider impact of flow blockage in LOCA analysis.	The impact of flow blockage effects via clad swell and rupture model is considered in the LOCA analysis and is reflected in the LOCA Summary Reports See Section 7).
4.1	Thermal and Hydraulic Criteria		
4.1.1	Hydraulic Compatibility	Hydraulic flow resistance similar to resident fuel assemblies.	Hydraulic compatibility acceptable. See Section 5.6.
4.1.2	Thermal Margin Performance	95/95 no DNB during steady state or AOO.	The results for this analysis will be verified for each reload core design. See Section 5.5.
4.1.3	Fuel Centerline Temperature	No fuel centerline melting.	The results for this analysis will be verified for each reload core design. See Section 4 and Item 3.2.4 in this table.

Table 2-2: Fuel Design Evaluation Results

Criteria Section	Description	Criteria	Results
4.1.4	Rod Bow	Protect thermal limits.	The results for this analysis will be verified for each reload core design. Framatome analysis demonstrates that no rod bow penalty is required for DNB analysis of CE16HTP fuel. See Section 5.7.6.
5.0	Neutronics Criteria		
5.1	Power Distribution	In accordance with Technical Specifications.	Criteria met. See Section 3.2
5.2	Kinetic Parameters		
	Doppler Reactivity Coefficient	Negative.	Criteria met.
	Power Coefficient	Negative relative to hot zero power (HZP).	The results for this analysis will be verified for each reload core design.
	Moderator Temperature Coefficient	In accordance with Technical Specification.	Criteria met. See Section 3.2.
5.3	Control Rod Reactivity	Technical Specification's margin maintained.	The results for this analysis will be verified for each reload core design.

Table 2-3: CFM Rod Local LHGR Limits

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[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
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Table 2-4: TCS Rod Local LHGR Limits

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[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
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2.4. End-of-Life Grid Crush Strength for CE16HTP Fuel

Appendix A to NUREG-0800 Standard Review Plan (SRP) Section 4.2, Fuel System Design, (Reference 2.12) provides NRC review guidance for the evaluation of fuel assembly structural response to externally applied forces. The review guidance contained in SRP Section 4.2 indicates that it is acceptable to assume that fuel spacer grid strength at the beginning-of-life is most limiting. However, NRC Information Notice (IN) 2012-09 (Reference 2.13) states that Operating Experience (OE) regarding the effects of in-reactor service on fuel assembly component response to externally applied forces (i.e., earthquakes and postulated pipe breaks in the reactor coolant system) challenge this existing NRC staff guidance. Specifically, OE shows that the crush strength of fuel assembly spacer grids may decrease during the life of a fuel assembly due to the effects of irradiation.

Framatome Topical Report ANP-10337P-A (Reference 2.8) presents a generic methodology to evaluate the structural response of PWR fuel assembly designs subjected to dynamic loads under seismic and loss-of-coolant accident (LOCA) events. The methodology is used to develop analytical models to describe the structural response of fuel assemblies in the horizontal and vertical directions. The revised methodology addresses issues raised in NRC Information Notice 2012-09. ANP-10337P proposes spacer grid impact force acceptance criteria based upon mechanical (grid crush) testing at beginning of life (BOL) and simulated end of life (EOL) conditions. Per the ANP-10337P-A Safety Evaluation, the Framatome generic methodology to evaluate the structural response of PWR fuel assembly designs subjected to dynamic loads under seismic and LOCA events is found to be acceptable subject to compliance with the ANP-10337P-A limitations and conditions. As addressed in Section 2.3.2.3, the conditions and limitations contained in the NRC SE for topical report ANP-10337P-A are met.

2.5. Operating Experience with HTP™ Fuel Assemblies

The Framatome fuel assembly for PVNGS Units 1, 2, and 3 is of a CE 16x16 lattice design. This lattice contains 236 fuel rods, four (4) corner guide tubes, and one (1) center guide tube / instrument tube. The corner and center guide tubes each occupy four (4) fuel rod positions. The fuel rods are positioned within the fuel assembly by eleven (11) spacer grids that are attached to the guide tubes. Figure 1-1 shows a schematic of the PVNGS CE16HTP fuel assembly.

The CE16HTP fuel design for the Palo Verde Nuclear Generating Station is the same (with minor changes) as the lead fuel assemblies that were introduced at PVNGS Unit 1 in Cycle 15 in 2008 (Reference 2.11). These lead fuel assemblies operated three cycles with positive performance as demonstrated in Post-Irradiation Examinations (PIE). Framatome has also supplied lead CE-16 HTP™ fuel assemblies at San Onofre Unit 2 that are very similar to the Palo Verde fuel design. The San Onofre Unit 2 fuel demonstrated positive performance after one cycle of operation (both in-board and core-periphery locations) before the plant was closed. Framatome has recently started supplying batch fuel of a similar CE-16 HTP™ design for St. Lucie Unit 2 starting with Cycle 23 (batch delivery in Spring of 2017). The St. Lucie Unit 2 CE-16 HTP™ fuel was designed to be more similar to the Framatome CE-14 HTP™ fuel (e.g., same structural material and active fuel height), but shares the same design features as the other Framatome CE-16 HTP™ fuel designs. The common design features for the Framatome CE-16 HTP™ fuel include: M5 clad fuel rods, MONOBLOC corner guide tubes, HTP™ / HMP™ spacer grids, a FUELGUARD™ lower tie plate (LTP), and a Framatome reconstitutable upper tie plate (UTP). The lead assembly programs and recent transition of the HTP™ fuel product for St. Lucie Unit 2 confirm the acceptable performance of the Framatome design.

Framatome provides the fuel for all of the CE-14 units in the United States (St. Lucie Unit 1, Millstone Unit 2, and Calvert Cliffs Units 1 and 2). The current Framatome designs for CE-14 fuel for these sister units use Zircaloy-4 HTP™ spacer grids at every elevation except the bottom grid. The bottom grid is an Alloy 718 HMP™ grid. The guide tubes are currently a Zircaloy-4 MONOBLOC design and the fuel rods use M5® cladding. The LTPs are the FUELGUARD™ design, and the UTPs are the Framatome reconstitutable design. The initial HTP™ / HMP™ / FUELGUARD™ transition began at St. Lucie Unit 1 in 2001 and that fuel design has operated for nine (9) cycles without failures. Fuel failures did occur at Millstone Unit 2, but this design did not have the lower Alloy-718 HMP™ grid. Since replacing the bottom grid at Millstone Unit 2 with an HMP™ grid, there have been no failures. Calvert Cliffs began their transition to the Framatome CE-14 HTP™ design in 2010. There were no Framatome fuel failures at the Calvert Cliffs units during the fuel transition.

2.6. References

2.1. EMF-92-116(P)(A), Revision 0, *Generic Mechanical Design Criteria for PWR Fuel Designs*, February 1999.

2.2. BAW-10240(P)-A, Revision 0, *Incorporation of M5® Properties in Framatome ANP Approved Methods*, May 2004.

2.3. BAW-10227(P)(A), Revision 1, *Evaluation of Advanced Cladding and Structural Material (M5®) in PWR Reactor Fuel*, June 2003.

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

2.5. BAW-10183(P)(A), Revision 0, *Fuel Rod Gas Pressure Criterion (FRGPC)*, July 1995.

2.6. BAW-10084(P)(A), Revision 3, *Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse*, July 1995.

2.7. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (SRP), *Section 4.2, Fuel System Design*, Revision 2, July 1981 (ADAMS Accession No. ML052340660).

2.8. ANP-10337P-A, Revision 0, *PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations*, April 2018.

2.9. Letter from D. C. Mims (APS) to NRC of March 08, 2008, *Palo Verde Nuclear Generating Station (PVNGS) Unit 1; Docket No. STN 50-528; Request for Temporary Exemption from the Provisions of 10 CFR 50.46 and 10 CFR 50, Appendix K for Lead Fuel Assemblies*, (ADAMS Accession No. ML080790524).

2.10. Letter from J. R. Hall (NRC) to R. K. Edington (APS) of October 14, 2008, *Palo Verde*

Nuclear Generating Station, Unit 1 – Temporary Exemption from the Requirements of 10 CFR Part 50, Section 50.46 and Appendix K (TAC No. MD8330), (ADAMS Accession Nos. ML082730003 and ML082730006).

2.11. ANP-2725(P), Revision 1, *Palo Verde Lead Fuel Assemblies Fuel Design Criteria Review*, September 2008.

2.12. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (SRP), Section 4.2, Revision 3, *Fuel System Design*, March 2007 (ADAMS Accession No. ML070740002).

2.13. Information Notice 2012-09, *Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength*, June 28, 2012 (ADAMS Accession No. ML113470490).

2.14. Regulatory Guide 1.61, Revision 1, *Damping Value for Seismic Design of Nuclear Power Plants*, March 2007 (ADAMS Accession No. ML070260029).

3. PHYSICS (NUCLEAR) DESIGN ANALYSIS

3.1. Physics (Nuclear) Design Introduction

The NRC-approved current licensing basis (CLB) for the reload core nuclear design is defined in PVNGS UFSAR Section 4.3. The purpose of the core analysis is to verify that the cycle-specific reload design and the key safety parameters are properly addressed for the reload design. The effects of introducing the CE16HTP fuel on the nuclear design bases and methodologies for PVNGS Units 1, 2, and 3 are evaluated in this section.

The Physics Design Analysis uses the CLB PVNGS reload methodology to establish an acceptable core design and to generate input to fuel performance, thermal-hydraulic, non-LOCA transient, and core protection setpoint reload analyses originated by APS. No physics methodology or code modifications are required to model Framatome fuel.

Section 3.2 describes the physics (nuclear) design analyses performed in support of this License Amendment Request.

Section 3.3 addresses the modeling of the center core assembly should it be manufactured by a vendor other than Framatome.

Section 3.4 addresses the gadolinia burnable absorber concentration limitation associated with Framatome fuel.

3.2. Description of Physics Design Analyses

Representative reactor core designs (representative fuel transition cycles) that meet APS fuel management guidelines were generated to validate that the analysis methodology tools, computer codes and procedures are adequate to perform mixed core and full core reloads for PVNGS with Framatome fuel. The cycles were designated as follows:

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

The evaluations and assessments of the PVNGS representative core designs containing Framatome fuel entailed the development of explicit neutronics models. The presence of the CE16HTP fuel in the PVNGS core was explicitly incorporated into these models, including the specific CE16HTP geometry and associated nuclear cross sections, and the use of gadolinia (Gd_2O_3) fuel burnable absorber (see Section 3.4). The differences between the various zirconium based cladding materials, M5®, ZIRLO®, and Optimized ZIRLO™, are neutronically insignificant. The calculations supporting the implementation of the Framatome CE16HTP fuel were performed using APS approved physics method codes (References 3.6 and 3.7). The features of the CE16HTP design are within the range of those methodologies. No methodology or code modifications are required to model M5® cladding or gadolinia burnable absorber.

The representative fuel transition cycles were not developed to be bounding of future cycle designs to be used at the plant but were developed to be representative of future cycle designs to demonstrate acceptable margins and to provide appropriately formatted physics data for input to representative fuel LOCA and Thermal-Mechanical evaluations originated by Framatome.

The representative fuel transition cycle loading patterns were developed based on design requirements (e.g., energy, peaking, and assembly placement) specified for PVNGS Unit 2. The loading patterns were depleted at a core power level of 3990 MWt. The first representative fuel transition cycle contains fresh CE16HTP fuel with once-burnt and twice-burnt CE16STD fuel. The second representative fuel transition cycle contains fresh and once-burnt CE16HTP fuel with twice-burnt CE16STD fuel. The third representative fuel transition cycle contains only CE16HTP fuel. These models show that sufficient margin exists between typical safety parameter values and the corresponding limits to allow flexibility in designing actual reload cores. Table 3-1 contains key core characteristics based on the representative fuel transition cycle designs. The fuel transition cycles are representative designs. The actual cycles may include combinations of three (3) types of fuel (CE16STD, CE16NGF, CE16HTP), depending on the fuel resident in the core at the time of the transition.

Validation of key safety parameters is performed by comparing calculated parameters for the cycle-specific reload to the values used in the safety analysis. These cycle-specific parameters are generated based on the current licensing basis methodology and APS code suite. If the key parameters are not within the reference safety analysis, then the transient will be re-analyzed or re-evaluated on a cycle-to-cycle basis using the stated methods.

As discussed in Section 4.3, the maximum CE16HTP fuel rod average burnup will be maintained less than the licensed burnup limits of 62 GWd/mtU for UO_2 rods and 55 GWd/mtU for gadolinia rods.

The standard methods of fresh fuel enrichment loading and integrated burnable poisons will be applied to control the power distribution and maintain compliance with the Technical Specifications and COLR. Changes in boron concentration and axial offset are typical of normal cycle-to-cycle variations in the core design.

The changes in fuel design and discharge burnup result in only a small impact on the results of the reload transition core analysis relative to the current design. The variations in these parameters are typical of the normal cycle-to-cycle variations that occur as fuel loading patterns are changed each cycle.

Changes to the core power distributions and peaking factors are the result of the normal cycle to cycle variations in core loading patterns. These will vary cycle-to-cycle based on actual energy requirements. The normal methods of feed enrichment variation and insertion of fresh burnable absorbers will be employed to control power distribution limits. Compliance with the Technical Specification power distribution limits will be assured using these methods.

Table 3-1: Representative Fuel Transition Cycle Core Characteristics

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3.3. Reactor Core Center Assembly

Typical PVNGS core reloads are 92 to 108 fresh feed assembly reloads. The PVNGS reactor core has an odd number of fuel assemblies (i.e., 241) and on occasion the center assembly is re-inserted from the spent fuel pool. After transition to CE16HTP fuel, the need to use an assembly from the pre-CE16HTP (i.e., CE16STD or CE16NGF) fuel design for the core center assembly may be necessary or desired. This center assembly is typically high burnup and not limiting with respect to power peaking and thermal performance. Use of this type of center assembly (pre-CE16HTP) will not be analyzed as a mixed core in the reload analysis process. For all non-physics analyses and for all administrative purposes, this type of core will be considered a full core of the new fuel type. However, the core physics analysis will specifically model this center assembly to ensure that the fuel pin burnup and fluence limitations are not exceeded and the peak integrated radial peaking factor for this assembly will be maintained at 0.95 or less of the core maximum integrated radial peaking factor at all times in core life to ensure that this assembly does not become limiting during cycle operation.

3.4. Gadolinia Burnable Absorber Limitation

Gadolinia has been approved by the NRC for use as a burnable absorber for Framatome fuel. Topical Report BAW-10231P-A (Reference 3.1), describes the Framatome analysis methods for gadolinia bearing fuel. The topical report describes the models, analytical methods, and procedures used by Framatome to evaluate the neutronics and performance of fuel containing up to 8 weight percent of gadolinia as a burnable absorber for United States applications.

The Framatome methods related to the modeling of fuel containing up to 8 weight percent of gadolinia as a burnable absorber are adopted by APS for use with the current licensing basis methodology using the CASMO/SIMULATE codes. To ensure meeting the Topical Report BAW-10231P-A gadolinia-related limitation, the PVNGS fuel management guidelines limit the burnable absorber use to no more than 8 weight percent gadolinia, and excludes gadolinia from the fuel rod top and bottom cutback regions. In addition, the PVNGS fuel management guidelines will specify a [[[redacted]]] in U-235 enrichment [[[redacted]]], for reload batch gadolinia bearing fuel rods.

3.5. Startup Test Activity Reduction (STAR) Program

WCAP-16011-NP-A (Reference 3.2) defines the generic Startup Test Activity Reduction (STAR) Program that allows a simplification in the startup testing program by eliminating for most cycles the Beginning of Cycle (BOC) Zero Power CEA Worth and Isothermal Temperature Coefficient (ITC) measurements. The use of WCAP-16011-NP-A for application of the STAR Program to PVNGS is justified in WCAP-17787-NP (Reference 3.3), as approved by PVNGS Operating License Amendment 195 (Reference 3.4). The STAR Program requires specified applicability requirements be met to utilize STAR for a given cycle. For the initial reload batch of CE16HTP fuel, the STAR Program applicability requirements will not be satisfied due to the change in burnable absorber (i.e., gadolinia). Therefore, for the initial reload batch the Startup Test Program will include the Beginning of Cycle (BOC) Zero Power CEA Worth and ITC measurements. The application of the STAR Program to future cores requires that the STAR Program applicability requirements as established in WCAP-16011-NP-A, and modified by WCAP-17787-NP as a result of the PVNGS unique design features, be met.

PVNGS Operating License Amendment 195 (Reference 3.4) approved modifications to the Moderator Temperature Coefficient (MTC) TS Surveillance Requirements 3.1.4.1 and 3.1.4.2 associated with implementation of WCAP-16011-NP-A. MTC SR 3.1.4.2 was changed to allow the option to eliminate the MTC measurement at two-thirds of expected core burnup, if the result of the 40 EFPD MTC measurement is within a tolerance limit of the design value. This result at 40 EFPD is a reflection of the fidelity (e.g., accuracy) of the cycle-specific core physics model. If the accuracy of the model is impacted by the introduction of CE16HTP fuel such that the 40 EFPD MTC cannot be predicted within the design tolerance limit, then the MTC measurement at two-thirds of expected core burnup will be performed in accordance with Technical Specification SR 3.1.4.2.

3.6. References

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

3.2. WCAP-16011-NP-A, Revision 0, *Startup Test Activity Reduction Program*, February 2005.

3.3. WCAP-17787-NP, Revision 0, *Palo Verde Nuclear Generating Station STAR Program Implementation Report*, August 2013.

3.4. License Amendment 195 to PVNGS Units 1, 2 and 3 Operating Licenses to change Moderator Temperature Coefficient Surveillance for Startup Test Activity Reduction Program, March 30, 2015, ADAMS Accession No. ML15070A124.

3.5. CE NPSD-911-A and Amendment 1-A, Analysis of Moderator Temperature Coefficients in Support of Change in the Technical Specifications End-of-Cycle Negative MTC Limits, September 15, 2000.

3.6. Letter USNRC to Arizona Public Service Company, *Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 – Issuance of Amendments on CASMO-4 / SIMULATE-3 (TAC Nos. MA9279, MA9280, and MA9281)*, March 20, 2001.

3.7. CENPD-266-P-A, *The ROCS and DIT Computer Codes for Nuclear Design*, April 1983.

4. FUEL ROD BEHAVIOR (PERFORMANCE) ANALYSIS

The primary objective of fuel rod behavior (performance) analysis is to evaluate the steady state fuel thermal and mechanical behavior of individual nuclear fuel rods as a function of time or burnup. Generally, this requires generation of representative values for fuel rod temperatures, rod internal gas pressure, and fuel rod deformation.

The approved PVNGS Reload Analysis Methodology for fuel performance analysis of Westinghouse supplied CE16STD and CE16NGF fuels uses the FATES3B fuel performance code. CE16STD and CE16NGF fuel assembly fuel rod performance will continue to be analyzed per existing approved methods with the FATES3B code.

The fuel rod performance of the CE16HTP fuel with M5® cladding is analyzed in accordance with the NRC-approved topical report BAW-10231P-A (Reference 4.1) utilizing the COPERNIC fuel performance code.

The COPERNIC fuel performance analysis performed, and data transmitted, depends on the specific application. The safety and LOCA analyses typically require initial fuel rod conditions, minimum rod internal pressure, core minimum and maximum gap conductance, rod minimum gap conductance, the minimum power to fuel centerline melt, the axial densification factor, and the engineering factor on linear heat rate (LHR). Additionally, the COPERNIC code is used to verify peaking factors, other limits to preclude fuel damage, and Technical Specifications on permissible LHRs.

The safety and LOCA analyses input data are in various forms, including values calculated directly by the COPERNIC code, and values calculated external to the COPERNIC code. [[[redacted]]].

4.1. Fuel Thermal Conductivity Degradation

Irradiation damage and the progressive buildup of fission products in fuel pellets result in reduced thermal conductivity of the pellets. NRC Information Notice (IN) 2009-23 and its Supplement 1 (References 4.3 and 4.4) acknowledge that thermal performance codes approved by the NRC before year 1999 did not include this reduction in thermal conductivity with increasing irradiation because earlier test data were inconclusive as to the significance of the effect. As such, per IN 2009-23, pre-1999 methods misrepresent fuel thermal conductivity, and safety analyses performed for reactors using methods that do not model fuel thermal conductivity degradation (TCD) as a function of burnup may be less conservative than previously understood.

The COPERNIC computer code is a more recent fuel performance code initially approved by the NRC in year 2002. The COPERNIC code addresses the effects of TCD as a function of burn-up as described in the Topical Report BAW-10231P-A Section 4.3 fuel thermal conductivity degradation model.

Per the Topical Report BAW-10231P-A Safety Evaluation, the staff performed comparisons of the COPERNIC code TCD model with other TCD models (Lucuta, et al., and NFI of Japan) as executed with the FRAPCON-3 code. The comparisons considered additional Framatome Cogema Fuels (FCF) and Halden data.

These comparisons led the staff to conclude that "based on an acceptable uncertainly level and good agreement of the temperature predictions between the COPERNIC and NRC audit codes, the staff considers that the thermal conductivity model is acceptable in the COPERNIC code." Therefore, no penalty is required to address the effects of TCD in CE16HTP fuel analysis using the COPERNIC code.

4.2. Fuel-to-Clad Gap Coefficient of Conductance

To ensure the fuel-to-clad gap coefficient of conductance (herein referred to as Hgap) that is currently modeled in the UFSAR Chapter 15 Safety Analysis is appropriate with CE16HTP fuel, the COPERNIC fuel performance code generated Hgap curves. [[[redacted]]].

To determine if the overall PVNGS UFSAR Chapter 15 methods and models are still acceptable for CE16HTP fuel, the Hot Zero Power (HZP) Control Element Assembly Withdrawal (CEAW) transient was chosen as the demonstration event. The CEAW transient is one of the few events that the maximum linear heat generation rate (LHGR) exceeds 21 kw/ft for short time intervals. Exceeding 21 kw/ft for short time intervals is acceptable as the analysis verifies that the total integrated energy deposited in the fuel does not result in peak centerline fuel temperature that exceed Technical Specification Safety Limit 2.1.1.2.

A separate COPERNIC analysis was performed to model and provide the [[[redacted]]]. The COPERNIC analysis used the time-dependent parameters generated by the NSSS transient code (e.g., calculated neutron power, fuel thermal response, surface heat transport, fluid conditions) to determine the peak fuel centerline temperature for the transient. The COPERNIC peak fuel centerline temperature was calculated to be significantly lower [[[redacted]]] than the value presented in the UFSAR (2600°F) using current conservative methods and models.

The current licensing basis utilized in UFSAR Chapter 15 remains applicable for use with the CE16HTP fuel design, when the COPERNIC generated Hgap is modeled in conjunction with the conservative methods and models in the current licensing basis.

4.3. Maximum CE16HTP Fuel Burnup Limits

The NRC Safety Evaluation for the Topical Report BAW-10240(P)-A (Reference 4.5) concluded that the Framatome ANP PWR design methods were acceptable with M5® material properties for rod average burnups of 62 MWd/kgU. This approval explicitly included application to Westinghouse and Combustion Engineering designed PWRs.

The CE16HTP fuel assembly mechanical design criteria were evaluated and shown to be met up to the licensed fuel rod average burnup limits of 62 GWd/mtU for UO_2 rods and 55 GWd/mtU for gadolinia rods.

4.4. References

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

4.2. Letter from C. M. Trammell (NRC) to W. F. Conway (APS) of June 14, 1993, *Approval of Reload Analysis Methodology Report – Palo Verde Nuclear Generating Station (TAC Nos. M85153, M85154, and M85155)*.

4.3. Information Notice 2009-23, *Nuclear Fuel Thermal Conductivity Degradation*, October 8, 2009 (ADAMS Accession No. ML091550527).

4.4. Information Notice 2009-23, Supplement 1, *Nuclear Fuel Thermal Conductivity Degradation*, October 26, 2012 (ADAMS Accession No. ML121730336).

4.5. BAW-10240(P)-A, Revision 0, *Incorporation of M5® Properties in Framatome ANP Approved Methods*, May 2004.

5. CORE THERMAL HYDRAULIC DESIGN ANALYSIS

5.1. Introduction

This section describes the core thermal-hydraulic (T-H) analysis methodology and analyses performed to support the qualification of Framatome CE16HTP fuel for PVNGS. The methodology and analyses support full cores of CE16STD, CE16NGF, or CE16HTP fuel. The methodology and analyses also support mixed cores.

The current T-H design basis for PVNGS includes the prevention of departure from nucleate boiling (DNB) on the limiting fuel rod with a 95 percent probability at a 95 percent confidence level (95/95) during normal operations and Anticipated Operational Occurrences (AOOs).

The thermal-hydraulic design methods remain the same as recently approved in License Amendment 205 (Reference 5.7). These methods include the following:

- Use of NRC-approved Westinghouse version of VIPRE-01 subchannel analysis code (referred to as VIPRE-W) for use in Departure from Nucleate Boiling Ratio (DNBR) calculations of CE fuel designs [[[redacted]]].
- Use of NRC-approved ABB-NV, WSSV, WSSV-T, and WLOP CHF correlations for use in DNBR calculations.
- Use of NRC-approved Statistical Combination of Uncertainties (SCU) methodology in Reference 5.16, with no change, to the SCU method described in Reference 5.14 and supplemented by Reference 5.15 to [[[redacted]]] to calculate an overall uncertainty factor.

To support use of CE16HTP fuel at PVNGS, the following methodology changes are proposed:

- Addition of EPRI-NP-2511-CCM-A (VIPRE-01 code) for use in DNBR calculations (Reference 5.10)
- Addition of BAW-10241(P)(A) (BHTP CHF correlation) for use in DNBR calculations (Reference 5.3)

5.2. Subchannel Analysis Codes

The current licensing basis includes multiple T-H codes (e.g., TORC, CETOP-D, HRISE, and VIPRE-W) for both CE16STD and CE16NGF fuel. APS requests approval to also allow the use of the VIPRE-01 code for Core Thermal-Hydraulic analysis.

The analysis of Framatome CE16HTP fuel at APS requires use of the BHTP CHF correlation with the VIPRE-01 and VIPRE-W codes. Sections 5.4.1 and 5.4.2 of this document provide the code verification and validation for the use of the BHTP CHF correlation with the VIPRE-01 and VIPRE-W codes, respectively.

5.3. CHF Correlations

PVNGS is currently licensed to implement a variety of CHF correlations. The analysis of Westinghouse CE16STD fuel is performed using the CE-1 or ABB-NV CHF correlation. The analysis of Westinghouse CE16NGF fuel is primarily performed using the ABB-NV and WSSV CHF correlations for non-mixing and mixing vane grids, respectively. In addition, the analysis of both CE16STD and CE16NGF fuel utilizes either the WLOP or Macbeth CHF correlation for modeling low pressure and low flow conditions when the primary CHF correlation is not applicable because the pressure or flow is outside the primary CHF correlation range of applicability.

5.3.1. CHF Correlations for Use Below First HTP™ Grid

The Framatome CE16HTP fuel design for PVNGS features the FUELGUARD Lower End Fitting with the Inconel High Mechanical Performance (HMP™) structural non-mixing vane grid below the first (i.e., lowermost) HTP™ mixing vane grid. Framatome recommends that locations downstream of the HMP™ grid and below the first HTP™ grid utilize one of their non-mixing vane grid CHF correlations, either XNB or BWU-N, and that locations above the first HTP™ grid utilize the BHTP CHF correlation.

An evaluation was performed to determine the need to model a CHF correlation for the bottom grid span region (i.e., between the bottom of the fuel assembly and the first HTP™ grid). The evaluation considered the Technical Specification on axial shapes and the safety analyses analytical space. The analytical space covered the entire core (axially and radially) and the entire operating space (temperature, pressure, and flow). The evaluation determined that there is no need to model a CHF correlation for the bottom grid span region, since a limiting CHF or minimum DNBR cannot be achieved in this region. Even though minimum DNBR cannot occur in the bottom grid span region, the VIPRE-01 and VIPRE-W code modeling will model an arbitrary CHF correlation in this region to facilitate proper code execution.

Therefore, use of a specific CHF correlation for the bottom grid span (i.e., between the start of fuel and the first HTP™ grid) is unnecessary. Even though minimum DNBR cannot occur in this region, the VIPRE-01 and VIPRE-W code modeling will utilize the BHTP CHF correlation in that region to facilitate proper code execution.

5.3.2. BHTP CHF Correlation

For Framatome CE16HTP fuel, DNBR margins in the core are predicted with the BHTP critical heat flux correlation. The BHTP CHF correlation was developed for a variety of fuel designs using the HTP™ spacer grid, based on CHF data obtained from 5x5 rod bundle testing in the Heat Transfer Research Facility of Columbia University (Reference 5.3).

The parameter ranges over which the BHTP CHF correlation is valid are addressed in Table 5-1. The BHTP CHF correlation was validated in VIPRE-01 based upon the same [[[redacted]]] data points from Topical Report BAW-10241(P)(A) Revision 0 (Reference 5.26). The range of applicability was subsequently extended in Topical Report BAW-10241(P)(A) Revision 1 (Reference 5.3). The extension of the BHTP application ranges for system pressure, mass velocity, and thermodynamic quality using LYNXT is discussed in Reference 5.3. The use of VIPRE-01 or VIPRE-W local conditions does not alter the conclusions reached for supporting the range extensions; therefore, the BHTP extensions remain applicable when using VIPRE-01 or VIPRE-W. However, since the mass velocity and thermodynamic quality are code dependent, the values in Table 5-1 reflect the use of VIPRE-01 based on the original [[[redacted]]] data points.

Table 5-1: BHTP CHF Correlation Parameter Ranges (1)

Parameter	Range
Pressure (psia)	1385 to 2425 (2)
Local Mass Velocity (Mlbm/hr-ft^2)	0.949 to 3.56
Local (Thermodynamic) Quality	≤ 0.357
Heated Length (feet)	9.8 to 14.0
Axial Spacer Span (inches)	10.5 to 26.2
Hydraulic Diameter (inches)	0.4571 to 0.5334

(1) [[[redacted]]].

(2) Refer to Condition and Limitation 2 in the Safety Evaluation for BAW-10241(P)(A), Revision 1 (Reference 5.3) for application from 1385 to 2600 psia.

Use of the BHTP CHF correlation with the VIPRE-01 and VIPRE-W codes is discussed in Sections 5.4.1 and 5.4.2, respectively.

5.3.3. Low Pressure CHF Correlation

5.3.3.1. Macbeth CHF Correlation

Departure from Nucleate Boiling evaluations of Westinghouse-supplied fuel for low pressure events such as the post-trip steam line break event are currently approved for modeling CE16STD and CE16NGF fuel using the Macbeth CHF correlation (References 5.6 and 5.7) or the WLOP CHF correlation (Reference 5.8) 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.

The Macbeth CHF correlation is approved for use with a DNBR limit of 1.30 for the rod cluster correlation. The Macbeth CHF correlation calculates the CHF as a function of mass flux, inlet subcooling, system pressure, heated diameter, and channel length. Specifically, the Macbeth CHF correlation parameter ranges for use as approved by the NRC are as follows:

Table 5-2: Macbeth CHF Correlation Parameter Ranges

Parameter	Range
Mass Velocity (Mlbm/hr-ft^2)	0.09 to 4.1
Inlet Subcooling (Btu/lbm)	- 150 to 380
Pressure (psia)	500 to 1000
Hydraulic Diameter (inches)	0.113 to 0.902

As such, the Macbeth CHF correlation can be used to model fuel that meets the Table 5-2 requirements. Framatome-supplied CE16HTP fuel meets the preceding parameter range limitations.

5.3.3.2. WLOP CHF Correlation

The WLOP CHF correlation with its 95/95 DNBR safety limit of 1.18 is applicable as an alternative to the Macbeth CHF correlation with its conservative 95/95 DNBR safety limit of 1.30. The WLOP CHF correlation's range of applicability provides more flexibility to DNBR evaluation at the hypothetical hot zero power steam line break (HZPSLB) conditions.

The WLOP CHF correlation parameter ranges approved by the NRC are given in WLOP Topical Report WCAP-14565-P-A Addendum 2-P-A (Reference 5.8), and the heated hydraulic diameter clarification given in LTR-NRC-07-49P (Reference 5.9, response to RAI 2), as follows:

Table 5-3: WLOP CHF Correlation Parameter Ranges

Parameter	Range
Pressure (psia)	185 to 1800
Local Coolant Quality	< 0.75
Local Mass Velocity (Mlbm/hr-ft^2)	0.23 to 3.07
Heated Hydraulic Diameter (inches)	0.679 to 1.000
Heated Length, HL (inches)	48* to 168
Inlet Subcooling (Btu/lbm)	150 to 380
Grid Spacing Term	27 to 115
* Set as minimum HL value, applied at all elevations below 48 inches	

As such, the WLOP CHF correlation can be used to model fuel that meets the Table 5-3 requirements. Framatome-supplied CE16HTP fuel meets the preceding WLOP parameter range limitations.

5.3.4. Post-DNB CHF Correlation

A post-DNB CHF correlation is not required because the VIPRE-W and VIPRE-01 T-H codes will not model the time-dependent physical changes that may occur within fuel rods at elevated temperatures in the post-DNB region. Note that the minimum DNBR value during the transient, which may be below the DNBR SAFDL, is used to determine the fuel failure, if any, that occurs in the various UFSAR Chapter 15 events (e.g., see Section 6.6 for the AOO from SAFDL event).

The post-DNB CHF correlation would only be modeled if DNB propagation were to occur. DNB propagation would not occur because either the overall time in DNB will be evaluated to ensure that it is less than 4.5 seconds before returning to a value greater than the DNBR limit, or the maximum strain will be analyzed to ensure that it is less than the strain limit that induces DNB propagation. Refer to Section 6.2 for additional discussion of DNB propagation.

5.4. Subchannel Analysis Codes Implementation of BHTP CHF Correlation

5.4.1. VIPRE-01 Code with BHTP CHF Correlation

The approved APS Reload Analysis Methodology uses multiple T-H codes (e.g., TORC, CETOP-D, HRISE, and VIPRE-W) for core thermal hydraulic analysis. APS is adding VIPRE-01 as an approved code for core thermal hydraulic analysis. For analysis of Framatome CE16HTP fuel, this requires use of the BHTP CHF correlation with the VIPRE-01 code.

The VIPRE-01 code has the capability of calling CHF correlations through the use of dynamically linked external files without requiring modification of the VIPRE-01 code. The APS formal software update process defined in Reference 5.18 was used to create the BHTP CHF correlation external file.

APS uses VIPRE-01 within the defined ranges of applicability of the BHTP CHF correlation as addressed in Section 5.3.2.

The BHTP CHF correlation determines water properties in a manner consistent with the current APS process. Specifically, the BHTP CHF correlation as implemented by Framatome in the LYNXT code utilizes water and steam properties based on ASME steam table evaluation functions (Reference 5.10 for VIPRE-01, Reference 5.3 for LYNXT). The BHTP CHF correlation has been generically approved by the NRC for application to Framatome fuel in Reference 5.3. The BHTP CHF correlation is not licensed for the bottom High Mechanical Performance (HMP™) grid in the Framatome CE16HTP fuel design. As discussed in Section 5.3.1, there is no need to model a CHF correlation for the bottom grid span region, since a limiting CHF or minimum DNBR cannot be achieved in this region. Even though minimum DNBR cannot occur in the bottom grid span region, the VIPRE-01 and VIPRE-W code modeling will model an arbitrary CHF correlation in this region to facilitate proper code execution.

The VIPRE-01 code with BHTP CHF correlation was tested against the same database used for LYNXT code with BHTP licensing in Reference 5.3 [[[redacted]]]. The VIPRE-01/BHTP results were compared to the measured test data from Reference 5.3.

The comparison shows that VIPRE-01/BHTP predicts CHF results that are in good agreement with the measured CHF test data, with no significant, unexpected, or unusual deviations. The comparison between VIPRE-01/BHTP predictions and measured test data is displayed in Figure 5-1.

Figure 5-1: Predicted VIPRE-01/BHTP DNB Heat Flux Data Compared to Measured DNB Heat Flux Data

[[

[redacted]

]]

As shown in Table 5-4, the resulting VIPRE-01/BHTP statistics (as further discussed in Section 5.4.1.2) are slightly better than those obtained with the NRC-approved LYNXT/BHTP thermal hydraulics model.

Table 5-4 – Overall BHTP CHF Benchmark Statistics Comparison

[[

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

]]

5.4.1.1. VIPRE-01 / BHTP Modeling Options

Table 5-5 presents the options employed in modeling VIPRE-01 with the BHTP CHF correlation.

Table 5-5 – VIPRE-01 / BHTP Modeling Options

Category	Modeling Option Selected
Water Properties	EPRI water properties
Subcooled Void Model	EPRI
Bulk Void Model	EPRI
Single Phase Forced Convection to Liquid	Dittus-Boelter
Subcooling and Saturated Boiling	Combination of Thom and Dittus-Boelter
Correlation for Boiling Curve Peak	EPRI
Transition Boiling	Tong-Young
Film Boiling	Bishop-Sandberg-Tong
CHF Correlations	BHTP (Framatome HTP™ fuel only)
Turbulent Mixing	VIPRE-01 turbulent mixing model with recommended parameters. Turbulent Momentum Factor = 0.8, abeta = 0.02.

5.4.1.2. Statistical Characterization of the BHTP DNB Correlation

The standard deviation of the data for each test falls between [[[redacted]]] across all test data. The average ratio of predicted to measured data ranges [[[redacted]]].

Figure 5-2 provides a frequency distribution of all VIPRE-01 generated BHTP P/M ratios with a superimposed normal distribution for comparison.

Figure 5-2: VIPRE-01/BHTP P/M Ratio Frequency Distribution

[[

[redacted]

]]

The BHTP DNB correlation safety limit of 1.123 is derived using the ratio of the predicted DNB heat flux to the measured DNB flux (P/M) ratio. The correlation safety 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 a non-parametric method (References 5.3 and 5.11). This method has been previously used to define a safety limit for the LYNXT code.

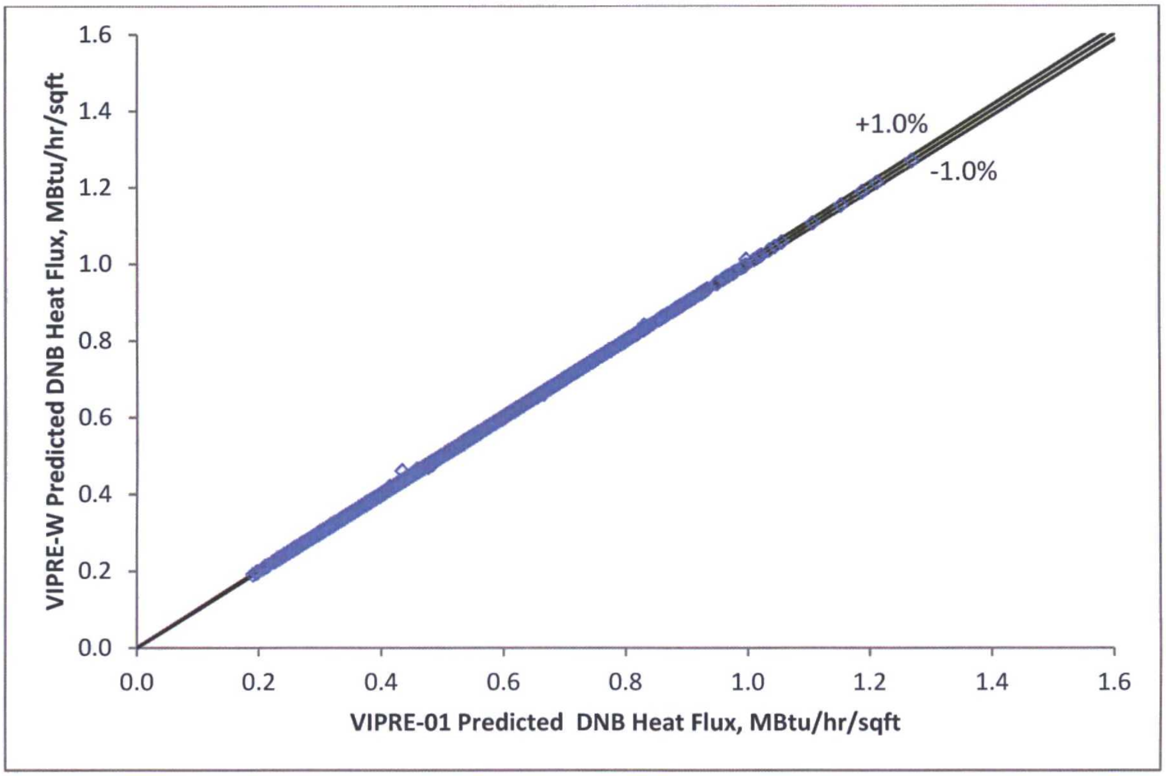
5.4.2. VIPRE-W Code with BHTP CHF Correlation

The VIPRE-W code has the capability of calling CHF correlations through the use of dynamically linked external files without requiring modification of the VIPRE-W code. The APS formal software update process defined in Reference 5.18 was used to create the BHTP CHF correlation external file.

The benchmarking of the VIPRE-W code with the BHTP CHF correlation was performed by calculating the CHF using the VIPRE-W code with the BHTP CHF correlation (VIPRE-W/BHTP) and demonstrating that it produces results consistent with the VIPRE-01 code with the BHTP CHF correlation (VIPRE-01/BHTP) [[[redacted]]].

The comparison shows that VIPRE-W/BHTP predicts CHF results that are in good agreement with the VIPRE-01/BHTP predicted CHF results, with no significant, unexpected, or unusual deviations. The comparison between VIPRE-W/BHTP predictions and VIPRE-01/BHTP predictions is displayed in Figure 5-3.

Figure 5-3: VIPRE-W vs VIPRE-01 Evaluation of all BHTP CHF Data



The overall average ratio of the predicted VIPRE-W/BHTP to the predicted VIPRE-01/BHTP is 1.0001 and overall standard deviation of 0.0019.

Given the small differences between VIPRE-W/BHTP and VIPRE-01/BHTP CHF results, VIPRE-W/BHTP is benchmarked and acceptable for use. Therefore, either code with the BHTP CHF correlation may be used to model Framatome CE16HTP fuel.

5.4.2.1. VIPRE-W / BHTP Modeling Options

Table 5-7 presents the options employed in modeling VIPRE-W with the BHTP CHF correlation. These options are consistent with those presented in Table 5-5 for modeling VIPRE-01 with the BHTP CHF correlation.

Table 5-7 – VIPRE-W / BHTP Modeling Options

Category	Modeling Option Selected
Water Properties	EPRI water properties
Subcooled Void Model	EPRI
Bulk Void Model	EPRI
Single Phase Forced Convection to Liquid	Dittus-Boelter
Subcooling and Saturated Boiling	Combination of Thom and Dittus-Boelter
Correlation for Boiling Curve Peak	EPRI
Transition Boiling	Tong-Young
Film Boiling	Bishop-Sandberg-Tong
CHF Correlations	BHTP (Framatome HTP™ fuel only)
Turbulent Mixing	VIPRE turbulent mixing model with recommended parameters. Turbulent Momentum Factor = 0.8, abeta = 0.02.

5.4.3. Process for Implementation of CHF Correlations with Other T-H Codes

This LAR addresses the proposed implementation of the BHTP CHF correlation with the VIPRE-01 and VIPRE-W thermal-hydraulic codes at Palo Verde. On a forward-going basis APS may also decide to implement any CHF correlation that the NRC has approved for use at Palo Verde, with any thermal-hydraulic code that the NRC has also approved for use at Palo Verde. Examples of this include the following:

- Implementing the ABB-NV or WSSV CHF correlations, which the NRC has previously approved for use with the VIPRE-W code, for use with the VIPRE-01 code;
- Implementing the CE-1 CHF correlation, which the NRC has previously approved for use with the VIPRE-W, CETOP-D, and TORC codes, for use with the VIPRE-01 code;
- Implementing the BHTP CHF correlation, which is anticipated to be approved for use with the VIPRE-01 and VIPRE-W code as described in this license amendment request, with the CETOP-D code and/or the TORC code.

The following explains the process that APS proposes to qualify combinations of codes and CHF correlations for licensing applications, including combinations that have not been previously reviewed by NRC. This explanation is offered in light of the NRC's Draft Regulatory Guide DG-1334, "Guidance for Implementation of 10 CFR 50.59, 'Changes, Tests, and Experiments,'" which was published in December 2016 (References 5.19 and 5.20) and which may eventually be approved and issued as Revision 1 to Regulatory Guide (RG) 1.187. APS is currently committed to Revision 0 of RG 1.187 (Reference 5.21) as described in Section 1.8 of the PVNGS UFSAR.

The proposed process is generally based on the June 1999 guidelines of NRC Generic Letter 83-11, Supplement 1, “Licensee Qualification for Performing Safety Analyses” (Reference 5.23), with clarifications made to address the subsequent issuance in November 2000 of RG 1.187. 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 three (3) months before the 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 offer to make available for NRC inspection, audit, or review 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 5.25). With respect to Software Quality Assurance (SQA), the Palo Verde QAPD invokes the design control and test control provisions of the American Society of Mechanical Engineers (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. Application procedures shall also include flexibility to allow comparison tests between different methodologies to show that conservative assessments can be made.
- 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 5.24), 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 difference computer codes. Such calculations shall verify and validate each new combination in accordance with the method used in either Section 5.4.1 (comparison to CHF test data) or 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.
- 10 CFR 50.59 Change Control – Because eligibility will apply only to computer codes that are already specified as COLR methods in Palo Verde Technical Specification 5.6.5, APS will treat the addition of a CHF correlation to an existing code as a change in an element of methodology, not a change in a method of evaluation. Thus APS will apply the "conservative or essentially the same" test of Section 4.3.8 of Nuclear Energy Institute (NEI) 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation" (Reference 5.22), to determine whether a new code and CHF correlation combination constitutes a departure from a method of evaluation described in the FSAR (as updated). For the purposes of making this determination, the "conservative or essentially the same" test may be successfully met on the basis of benchmark or comparison calculations that are performed as described above. APS believes the notification process described above will ensure the NRC is well-informed of any changes in thermal-hydraulic analysis practices at Palo Verde, before they are implemented in the plant. APS shall therefore interpret the notification provided to NRC via this process as fulfilling any code-related or CHF correlation-related Safety Evaluation constraint or limitation, that would otherwise require prior NRC review and approval of new code and correlation combinations.
- The PVNGS UFSAR will explicitly identify any correlation and code combinations approved under this process.

This proposed license amendment will authorize the use of Framatome fuel for PVNGS. The preceding items will ensure the NRC is well-informed of any future changes in thermal-hydraulic analysis practices at Palo Verde, before they are implemented in the plant.

5.5. T-H Analysis

The APS Core Thermal-Hydraulic analysis methodology remains unchanged except for the use of the BHTP CHF correlation with the VIPRE-W and VIPRE-01 codes. The impact of analyzing Framatome CE16HTP fuel in the APS Core T-H analysis process is discussed in this section.

The APS Core Thermal Hydraulic Analysis process [[[redacted]]]. This process is performed by benchmarking the CETOP-D model to the more detailed VIPRE model, the results of which are addressed in Section 5.5.2.

The fuel assembly hydraulics are examined when different fuel designs are present in the reactor core to assess the impact on the core inlet flow distribution (IFD). Section 5.5.1 provides detail on the calculation process that uses a detailed T-H code set up as a full core model with each assembly modeled to estimate the impact on the IFD of a mixed core.

[[[redacted]]].

5.5.1. Fuel Assembly Hydraulics (Inlet Flow Distribution)

The core inlet flow distribution (IFD) is an input to the core thermal-hydraulic analysis for TORC, VIPRE-01, and VIPRE-W. The initial reference IFD was developed by Combustion Engineering (CE) based on a small-scale model test. This IFD was developed with uniform core inlet pressure losses (i.e., all fuel assemblies had the same grid design). When different fuel designs are used, the varying fuel assembly inlet flow resistances could impact the IFD. Creating a new scale model to test is not a practical venture for every mixed core, so CE developed a calculation process using the detailed core TH code to estimate the impact on the IFD of a mixed core. This process is applicable to different fuel types, but also can be used to model specific design changes in an existing fuel design such as the implementation of the GUARDIAN™ grid to the CE16STD design.

A demonstration Inlet Flow Distribution calculation was performed. Since the actual mixed core for the first batch of Framatome fuel has not been determined, a sample mixed core with all three fuel types (i.e., CE16STD, CE16NGF, and CE16HTP) was developed. The APS T-H models utilize a full core design with each quarter-assembly modeled. The IFD calculation shows that the [[[redacted]]]. This inlet flow reduction (or increase) would be recovered within a few feet through crossflow that is driven by fluid pressure differences.

Downstream T-H analyses would utilize the mixed core IFD.

Mixed cores also result in modified crossflow between dissimilar fuel assemblies because of unequal pressure drop differences from dissimilar spacer grids. The detailed T-H model used by APS models every grid for three fuel types with their respective grid loss coefficients. This assures that crossflow between assemblies of different fuel types is calculated and applied in the analysis.

5.5.2. CETOP-D Benchmarking

The APS Core Thermal-Hydraulic Analysis methodology utilizes a detailed TORC code or VIPRE code quarter-core model where the fuel assembly of interest is modeled in detailed subchannels, and other assemblies in the core are each modeled as individual lumped one-quarter fuel assembly channels. Per Section 5.4.2, either the VIPRE-01 or VIPRE-W code with the BHTP CHF (VIPRE/BHTP) correlation may be used to model Framatome CE16HTP fuel.

The first part of this analysis screens for fuel assemblies that have sufficiently high power during the cycle to be potentially limiting with respect to DNBR. Several different assemblies can be found to be close to DNBR limiting during the fuel cycle, and these assemblies are all evaluated in CETOP-D benchmarking. [[redacted]] for use in the COLSS/CPCs Setpoints Analysis as well as in the Transient Analysis.

In a mixed core scenario, the VIPRE code uses the appropriate CHF correlations and grid loss coefficients for each fuel type that is potentially DNBR limiting. Because each fuel assembly is modeled with its design loss coefficients, grids, fuel rods, channel sizes, etc., the APS Core Thermal-Hydraulic methodology is inherently a mixed core analysis process, and no generic mixed core DNBR penalty is necessary.

A demonstration CETOP-D benchmarking calculation for Framatome fuel was performed. This analysis performed benchmarking for CE16HTP fuel with CETOP-D using the CE-1 CHF correlation and VIPRE using the BHTP CHF correlation. [[redacted]].

5.6. DNB Methodology / DNBR Limit for CE16HTP Fuel

The DNB analyses continue to be based on the Modified Statistical Combination of Uncertainties (MSCU) process as approved by the NRC in CEN-356(V)-P-A, Revision 01-P-A (Reference 5.12) and WCAP-16500-P-A Supplement 1, Revision 1 (Reference 5.13) for CE-NSSS's with digital setpoint systems. With the MSCU methodology, uncertainties are treated in two groups. One group combines system parameter uncertainties with CHF correlation uncertainty and subchannel code uncertainty statistically to generate a 95/95 DNBR safety limit and associated probability density function (pdf). The other group uses this pdf and statistically combines it with state parameter uncertainties and uncertainties related to Core Operating Limits Supervisory System (COLSS) and Core Protection Calculator (CPC) algorithm, simulator model, computer processing and startup measurements to determine the COLSS and CPC overall uncertainty factors. The section herein discusses the system parameter SCU treatment method of the first group.

The system parameters are characterized by the physical system through which the coolant passes. The parameter and code uncertainties included in the overall system parameter DNB uncertainty factor are:

- Core inlet flow distribution
- Engineering factor on enthalpy rise
- Systematic fuel rod pitch
- Systematic fuel rod outer diameter (OD)
- Engineering factor on heat flux
- Subchannel code uncertainty

The PVNGS UFSAR Section 4.4 references Enclosure 1-P to LD-82-054 (Reference 5.14) and Supplement 1-P to Enclosure 1-P to LD-82-054 (Reference 5.15) as the licensing documents for PVNGS-specific SCU system parameter uncertainty treatment method. The PVNGS specific system parameter SCU method involves [[[redacted]]], presented in Section 5.3 of CEN-139(A)-P (Reference 5.16) to combine inlet flow factor uncertainty to calculate an overall uncertainty factor. Other system parameter uncertainties are combined using a [[[redacted]]], as discussed in Reference 5.14. For the applications, the system parameter SCU methodology approved by the NRC (Reference 5.16) was used to [[[redacted]]]. The 95/95 probability/confidence DNBR safety limit was determined from the resultant pdf and was further adjusted to account for the rod bow penalty, if applicable, to arrive at the final DNBR safety limit.

With the introduction of the VIPRE-W code, the system parameter SCU process was improved by [[[redacted]]] with VIPRE-W runs, instead of through a [[[redacted]]]. The main purpose of the SCU [[[redacted]]] was to facilitate a large number of [[[redacted]]] without making excessive detailed subchannel code runs which was impractical with the available technology in the past. With the significant improvements in the computer technology in recent years, it is feasible to perform a large number of [[[redacted]]]. The improved SCU process allows running [[[redacted]]] for robust sampling and calculations with the VIPRE-W code [[[redacted]]] when linked with an uncertainty analysis code, instead of the uncertainty convolution through the [[[redacted]]]. The uncertainty analysis code utilizes [[[redacted]]] approved by the NRC in CEN-356(V)-P-A, Revision 01-P-A (Reference 5.12). Replacement of the DNBR [[[redacted]]] is consistent with the existing SCU methodology as approved by the NRC (Reference 5.16).

To perform [[[redacted]]] with the improved SCU process, data population based on the input of the system parameter uncertainties were generated by [[[redacted]]]. All of the data population for the DNBR distribution was generated at [[[redacted]]] of state parameter conditions. The improved SCU process still maintains the same level of conservatism pertaining to the [[[redacted]]] that is searched at the [[[redacted]]]. Consistent with the existing SCU process, the DNBR distribution was further adjusted to deterministically account for rod bow penalty, if needed, and to preserve artificial margin to bound future reload designs.

Since no changes have been introduced to the lower core support structure, flow skirt, and In-Core Instrument (ICI) arrangement as part of Framatome CE16HTP fuel implementation, the inlet flow factors and uncertainties remain unchanged as reported in Reference 5.15. The variations and tolerance deviations pertaining to CE16HTP [[[redacted]]] were evaluated to obtain new values for heat flux and enthalpy rise engineering factor appropriate for the CE16HTP design. Framatome does not report [[[redacted]]]. APS conservatively calculated these parameter uncertainties for inclusion in the SCU DNBR safety limit. The uncertainty in systematic rod pitch for the CE16HTP design was derived using the [[[redacted]]]. [[[redacted]]] on the standard deviation value of rod OD was used to obtain the systematic uncertainty on rod OD for the CE16HTP design. The parameter uncertainties used to develop SCU DNBR safety limit for CE16HTP design are presented in Table 5-8. Since the uncertainties are considered in determining the SCU DNBR safety limit, the DNBR calculations in the plant safety analyses are performed using the nominal values (without the uncertainties) of the system parameters.

The current PVNGS SCU DNBR safety limit and pdf based on CE-1 CHF correlation include allowance for NRC imposed HID-1 (STD) grid DNBR penalty and for rod bow DNBR penalty. The BHTP CHF correlation includes [[[redacted]]]. Therefore, no HID-1 (STD) DNBR penalty is required to apply on the CE16HTP fuel due to different grid spacing. For Framatome fuel, [[[redacted]]]. However, by the time the assembly reaches this burnup, the assembly will be operating at a much lower power than the limiting rod in the core. That difference in power will more than offset the rod bow penalty. Thus, [[[redacted]]] is applied.

The BHTP CHF correlation was applied to the CE16HTP design. Based on the applicable system parameter uncertainties, BHTP CHF correlation uncertainty, and the VIPRE-W based DNBR sensitivity, an SCU DNBR safety limit value of 1.27 was established for the DNBR analyses using VIPRE-W and the BHTP CHF correlation for the CE16HTP fuel. Based on the benchmarking provided in Section 5.4.2, this value is also applicable to VIPRE-01 and the BHTP correlation. The 95/95 CHF correlation DNBR safety limit for the BHTP CHF correlation was preserved in the statistical treatment per NRC IN-2014-1 (Reference 5.17).

Table 5-8 – Components Combined in the DNBR PDF for Palo Verde CE16HTP Fuel

[[

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted]]].

5.7. Thermal-Hydraulic Compatibility

The thermal hydraulic compatibility analyses consider the effect of the fuel transition on core pressure drop, total bypass flow, crossflow velocity, RCS flow rate, control element assembly drop time, fuel rod bow, and guide tube heating. The purpose of these analyses is to assess the impact of the fuel transition on the Framatome CE16HTP fuel assemblies and to assess the thermal hydraulic behavior of the mixed and equilibrium cores as compared to the current core.

5.7.1. Core Pressure Drop Analysis

The CE Standard Westinghouse fuel assemblies have a lower overall resistance to flow than the Framatome CE16HTP fuel design; therefore, as the core transitions from a full core of Westinghouse CE16STD fuel to a full core of Framatome CE16HTP fuel, the core pressure drop increases. However, the Westinghouse CE16NGF fuel assemblies have a higher overall resistance to flow than the Framatome CE16HTP fuel design; therefore, as the core transitions from a full core of Westinghouse CE16NGF fuel to a full core of Framatome CE16HTP fuel, the core pressure drop decreases. An analysis was performed to assess the change in core pressure drop associated with introduction of Framatome CE16HTP fuel.

The core pressure drop for a full core of Framatome CE16HTP fuel design is [[[redacted]]]. The total pressure drop associated with the full core of Framatome CE16HTP fuel design is [[[redacted]]] than the total pressure drop of the Westinghouse CE16STD core and [[[redacted]]] than the total pressure drop of the Westinghouse CE16NGF core. Therefore, the total pressure drop for the Framatome CE16HTP fuel design falls between the total pressure drop for the Westinghouse CE16STD and CE16NGF fuel designs.

5.7.2. Total Bypass Flow Analysis

The change in total bypass flow was examined to determine if the active heat transfer coolant flow will be adversely impacted by the introduction of Framatome CE16HTP fuel. The bypass flow includes the following flow paths: guide tubes, vessel upper head, inlet-to-exit nozzle, and core barrel/baffle. The change in total bypass flow was determined by examining the change due to non-guide tube paths and 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 core pressure drop for a full core of Framatome CE16HTP fuel is higher than the core pressure drop for a Westinghouse CE16STD core and lower than the core pressure drop for a Westinghouse CE16NGF core. As a result, the driving force for bypass flow increases and the total bypass flow increases transitioning from Westinghouse CE16STD fuel, and decreases when transitioning from Westinghouse CE16NGF fuel. [[[redacted]]]. Therefore, the active heat transfer coolant flow will not be adversely impacted by the introduction of Framatome CE16HTP fuel.

5.7.3. Crossflow Velocity Analysis

The Inter-Assembly Crossflow velocities affecting the Framatome CE16HTP fuel design were analyzed to assure satisfactory performance during a transition. This data is generated as an input for mechanical calculations. Different core configurations were considered in the analysis, ranging between bounding configurations with Framatome assemblies with one or both Westinghouse assembly designs. Although other geometries and operating conditions may result in different crossflow velocity profiles, the analyzed scenario provides representative crossflow velocities to cover core configurations associated with a fuel transition. The results are representative of anticipated operating conditions and are used to develop bounding inputs for mechanical analyses.

5.7.4. Reactor Coolant System Flow Rate Analysis

An analysis was performed to assess the change in primary system loop flow attributed to a fuel transition. The analysis indicates [[[redacted]]].

The RCS loop flow for the Framatome CE16HTP fuel design falls between the RCS loop flow for the Westinghouse CE16STD and CE16NGF fuel designs. The change in the Reactor Coolant System (RCS) loop flow will not impact the Technical Specification minimum loop flow rate.

5.7.5. CEA Drop (Scram) Time Analysis

An assessment was performed to validate that the Technical Specification requirement for the control element assembly (CEA) drop time is not challenged because of a fuel transition. The CEA drop time is primarily dependent on the number, size, and location of the guide tube weep holes, as well as the inner diameter and height of the guide tube dashpot region.

Due to the similarities between the Westinghouse and Framatome guide tube designs, the CEA drop times will not be significantly impacted by a fuel transition and will remain below the minimum Technical Specification requirement.

5.7.6. Fuel Rod Bow Analysis

The impact of rod bowing on the minimum DNBR and local power peaking was evaluated for Framatome CE16HTP fuel using the rod bow methodology described in Reference 5.2. The objective was to determine the threshold burnup level at which a rod bow penalty must be applied.

Based upon the rod bow analysis, no DNBR penalty is required until [[[redacted]]]. The DNBR penalty is based upon the burnup point at which the [[[redacted]]]. The LHGR penalty is based on the [[[redacted]]]. Thus, [[[redacted]]] is applied.

Refer to Section 5.6 for additional discussion related to application of fuel rod bow penalties.

5.7.7. Guide Tube Heating Analysis

Boiling of coolant within the guide tubes has the potential to increase corrosion rates and be detrimental for neutron moderation. An analysis was performed to demonstrate that boiling will not occur within the guide tubes of the Framatome fuel assemblies. For conservatism, severe operating conditions were used in the analysis.

Guide tube heating is most severe when a neutron absorbing material is inserted into the guide tube. The analysis considered a high-powered assembly with the control rods at Power Dependent Insertion Limit (PDIL) conditions. The analysis demonstrates that for all predicted control rod linear heat generation rates, boiling will be precluded within the guide tube.

5.7.8. T-H Compatibility Conclusions

The Framatome CE16HTP fuel design for Palo Verde is thermal-hydraulically compatible with the co-resident fuel designs at the Palo Verde Units. The Framatome CE16HTP fuel design for Palo Verde has been analyzed in accordance with NRC-approved codes and methods using representative fuel cycle inputs.

5.8. References

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5.13. WCAP-16500-P-A, Supplement 1, Revision 1, *Application of CE Setpoint Methodology for CE 16×16 Next Generation Fuel (NGF)*, December 2010.

5.14. LD-82-054, Enclosure 1-P, *Statistical Combination of Uncertainties, Combination of System Parameter Uncertainties in Thermal Margin Analyses for SYSTEM 80*, May 1982.

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5.17. NRC Information Notice (IN) 2014-01, *Fuel Safety Limit Calculation Inputs were Inconsistent with NRC-Approved Correlation Limit Values*, February 21, 2014.

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5.19. *Guidance for Implementation of 10 CFR 50.59, 'Changes, Tests, and Experiments,'* 81 FR 94275, December 23, 2016 [https://www.gpo.gov/fdsys/pkg/FR-2016-12-23/pdf/2016-30921.pdf].

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6. NON-LOCA TRANSIENT ANALYSIS

6.1. Event Assessment

To quantify the effect of introducing the Framatome CE16HTP fuel design into the PVNGS safety analysis licensing basis, all Updated Final Safety Analysis Report (UFSAR) Chapter 15 Non-LOCA transient analyses were evaluated.

Table 6-1 provides a review of the use of Framatome CE16HTP fuel on the various UFSAR Chapter 15 Non-LOCA transient events. As noted in Table 6-1, the only changes required to account for use of CE16HTP fuel were to Departure from Nucleate Boiling (DNB) propagation (as discussed in Section 6.2), use of statistical convolution to determine fuel failure (as discussed in Section 6.3), and CEA Ejection (as discussed in Section 6.5). No other modifications to the currently approved methodology as discussed in the UFSAR Chapter 15 event sections were required for evaluating non-LOCA transients for CE16HTP fuel.

The modeling of M5® material properties in non-LOCA transient events is specifically addressed in Section 6.4, Summary of Cladding Related Models in the Non-LOCA Transient Evaluation Models.

The limiting infrequent event (an AOO from SAFDL) is presented in Section 6.6 to provide a validation of the currently approved methodology with CE16HTP fuel design including the use of statistic convolution in determining number of fuel failures.

The fuel handling accident is presented in Section 6.7 to address the applicability of the current analyses of record for the CE16HTP fuel design.

Table 6-1: Impact of the Use of Framatome CE16HTP Fuel on UFSAR Chapter 15 Non-LOCA Transient Events

UFSAR Section	Event	Acceptance Criteria	Impact
15.1	Increase in Heat Removal by the Secondary System		
15.1.1	Decrease in Feedwater Temperature (DFWT)	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Peak pressures and fuel performance are bounded by Increased Main Steam Flow (Section 15.1.3). Doses bounded by IOSGADVLOP (15.1.4) and Limiting Infrequent Event (15.E).
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	
		Offsite Doses ≤ a small fraction or 10% of 10 CFR Part 100. Control Room Dose ≤ GDC 19.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.1.2	Increase in (Main) Feedwater Flow (IFWF)	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Peak pressures and fuel performance are bounded by Increased Main Steam Flow (Section 15.1.3). Doses bounded by IOSGADVLOP (15.1.4) and Limiting Infrequent Event (15.E).
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	
		Offsite Doses ≤ a small fraction or 10% of 10 CFR Part 100. Control Room Dose ≤ GDC 19.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.1.3	Increased Main Steam Flow (IMSF)	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Pressure in the RCS and main steam system will be maintained below 110% of the design value. The transient Linear Heat Rate (LHR) will not exceed 21.0 kW/ft. Therefore, the fuel centerline melt temperature will not be exceeded. For the moderate frequency increase in main steam flow event (without an additional single failure), fuel cladding integrity will be maintained. For the infrequent increase in main steam flow event (with an additional single failure), limited fuel cladding degradation may occur. However, offsite and control room radiological dose consequences are bounded by those that may result from an IOSGADVLOP event, and are in compliance with regulatory guidelines (see UFSAR Section 15.1.4).
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	
		Offsite Doses ≤ a small fraction or 10% of 10 CFR Part 100. Control Room Dose ≤ GDC 19.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.1.4	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (IOSGADV)	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Pressure in the RCS and main steam system will be maintained below 110% of the design value. The transient Linear Heat Rate (LHR) will not exceed 21.0 kW/ft. Therefore, the fuel centerline melt temperature will not be exceeded. For the moderate frequency IOSGADV event (without an additional single failure), fuel cladding integrity will be maintained. For the infrequent IOSGADVLOP event (i.e., an IOSGADV event with an additional single failure), limited fuel cladding degradation may occur. However, offsite radiological dose consequences will not exceed a small fraction, or 10%, of 10 CFR Part 100 guideline values. Control room dose consequences will not exceed the limits specified by GDC 19.
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	
		Offsite Doses ≤ a small fraction or 10% of 10 CFR Part 100. Control Room Dose ≤ GDC 19.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.1.5	Steam System Piping Failures Inside and Outside Containment – Operating Modes 1 and 2	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Pressure in the RCS and main steam system will be maintained below 110% of the design value. The transient Linear Heat Rate (LHR) will not exceed 21.0 kW/ft. Therefore, the fuel centerline melt temperature will not be exceeded. If a MSLB results in an accident-Generated Iodine Spike (GIS), offsite radiological dose consequences will not exceed a small fraction, or 10%, of the 10 CFR Part 100 guideline values. If a MSLB results in 1% failed fuel, or if it occurs with a Pre-accident Iodine Spike (PIS), offsite radiological dose consequences will not exceed 10 CFR Part 100 guideline values. Control room dose consequences will not exceed the limits specified by GDC 19.
		Peak Secondary Pressure ≤110% of Design.	
		The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit based on an acceptable correlation. For PVNGS, the DNBR SAFDL for Post- Trip main steam line break (MSLB) is 1.30, based on the Macbeth CHF correlation. If the DNBR falls below this value, fuel failure must be assumed for all rods that do not meet the criterion; unless it can be shown, based on an acceptable fuel damage model, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.	
		The potential for fuel pellet melting is evaluated by confirming that the maximum Linear Heating Rate (LHR) remains below a steady-state value that conservatively bounds fuel centerline melt.	
		For a MSLB with an assumed Pre-accident Iodine Spike (PIS) and for a MSLB with the highest worth control rod stuck out of the core, the calculated doses should not exceed the guideline values of 10 CFR Part 100.	

UFSAR Section	Event	Acceptance Criteria	Impact
		For a MSLB with the equilibrium iodine concentration for continued full power operation in combination with an assumed accident Generated Iodine Spike (GIS), the calculated doses should not exceed a small fraction (10%) of the above guideline values of 10 CFR Part 100.	
15.1.6	Steam System Piping Failures Inside and Outside Containment – Mode 3 Operation	Peak RCS Pressure ≤110% of Design.	No change to methodology required:
		Peak Secondary Pressure ≤110% of Design.	Pressure in the RCS and main steam system will be maintained below 110% of the design value.
		The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit based on an acceptable correlation. For PVNGS, the DNBR SAFDL for Post- Trip MSLB is 1.30, based on the Macbeth correlation. If the DNBR falls below this value, fuel failure must be assumed for all rods that do not meet the criterion; unless it can be shown, based on an acceptable fuel damage model, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.	The transient Linear Heat Rate (LHR) will not exceed 21.0 kW/ft. Therefore, the fuel centerline melt temperature will not be exceeded. If a MSLB results in an accident-Generated Iodine Spike (GIS), offsite radiological dose consequences will not exceed a small fraction, or 10%, of the 10 CFR Part 100 guideline values.
		The potential for fuel pellet melting is evaluated by confirming that the maximum Linear Heating Rate (LHR) remains below a steady-state value that conservatively bounds fuel centerline melt.	If a MSLB results in 1% failed fuel, or if it occurs with a Pre-accident Iodine Spike (PIS), offsite radiological dose consequences will not exceed 10 CFR Part 100 guideline values.
		For a MSLB with an assumed Pre-accident Iodine Spike (PIS) and for a MSLB with the highest worth control rod stuck out of the core, the calculated doses should not exceed the guideline values of 10 CFR Part 100.	Control room dose consequences will not exceed the limits specified by GDC 19.

UFSAR Section	Event	Acceptance Criteria	Impact
		For a MSLB with the equilibrium iodine concentration for continued full power operation in combination with an assumed accident Generated Iodine Spike (GIS), the calculated doses should not exceed a small fraction (10%) of the above guideline values of 10 CFR Part 100.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.2	Decrease in Heat Removal by the Secondary System		
15.2.1	Loss of External Load (LOL)	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Bounded by Loss of Condenser Vacuum (Section 15.2.3). This event with a loss of offsite power results in an event similar to the Loss of Flow (LOF) event (Section 15.3.1).
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.2.2	Turbine Trip (TT)	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Bounded by Loss of Condenser Vacuum (Section 15.2.3). This event with a loss of offsite power results in an event similar to the Loss of Flow (LOF) event (Section 15.3.1).
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.2.3	Loss of Condenser Vacuum (LOCV)	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Pressure in the RCS and main steam system will be maintained below 110% of the design value. The transient Linear Heat Rate (LHR) will not exceed 21.0 kW/ft. Therefore, the fuel centerline melt temperature will not be exceeded. For the moderate frequency LOCV event (without an additional single failure), fuel cladding integrity will be maintained. This event with a loss of offsite power results in an event similar to the Loss of Flow (LOF) event (Section 15.3.1).
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	
15.2.4	Main Steam Isolation Valve Closure	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Bounded by Loss of Condenser Vacuum (Section 15.2.3). This event with a loss of offsite power results in an event similar to the Loss of Flow (LOF) event (Section 15.3.1).
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	

UFSAR Section	Event	Acceptance Criteria	Impact
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	
15.2.5	Steam Pressure Regulator Failure	N/A	This event does not apply to the Palo Verde System 80 design.
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries (LOAC)	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Bounded by Loss of Flow (LOF) event (Section 15.3.1). Peak pressures are bounded by Loss of Condenser Vacuum (Section 15.2.3).
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.2.7	Loss of Normal Feedwater (LFW)	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Bounded by Loss of Condenser Vacuum (Section 15.2.3). This event with a loss of offsite power results in an event similar to the Loss of Flow (LOF) event (Section 15.3.1).
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	
15.2.8	Feedwater System Pipe Breaks (FWLB)	For low probability events, such as small FWLBs with a limiting single failure (SF) and offsite power available, primary and secondary system pressures must not exceed 110% of their respective design pressures.	No change to methodology required: Pressure in the RCS and main steam system will be maintained below 110% of the design value for low probability events. Pressure in the RCS and main steam system will be maintained below 120% of the design value for very low
		For very low probability events, including large FWLBs and small FWLBs with a concurrent loss of offsite power (LOP), primary and secondary system pressures must not exceed 120% of their respective design pressures.	

UFSAR Section	Event	Acceptance Criteria	Impact
		The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit based on an acceptable correlation. If the DNBR falls below this value, fuel failure must be assumed for all rods that do not meet the criterion; unless it can be shown, based on an acceptable fuel damage model, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.	probability events. If a FWLB results in an accident-Generated Iodine Spike (GIS), offsite radiological dose consequences will not exceed a small fraction, or 10%, of the 10 CFR Part 100 guideline values.
		The dose consequences for FWLBs are limited to a small fraction of 10 CFR part 100 limits for TS primary and secondary coolant activities and to within 10 CFR 100 limits for a Pre-Existing Iodine Spike.	If a FWLB occurs with a Pre-accident Iodine Spike (PIS), offsite radiological dose consequences will be within 10 CFR Part 100 guideline values. Control room dose consequences will not exceed the limits specified by GDC 19.
		Auxiliary Feedwater (AFW) System is capable of supplying adequate feedwater flow to the unaffected SG during the accident to ensure an orderly shutdown.	The AFW System will be demonstrated capable of RCS decay heat removal during the FWLB.
		Maximum pressurizer volume for the limiting Chapter 15 transient shall remain below the Pressurizer Safety Valve (PSV) nozzles when the PSVs are required to open.	It will be demonstrated that the maximum two-phase pressurizer volume and level will remain below the elevation where water entrainments is predicted to occur and only steam is released, so that PSV operability is not challenged.
		Maximum pressurizer pressure shall remain below 2697 psia.	The peak pressurizer pressure will be maintained lower than 2697 psia so that PSV operability is not challenged

UFSAR Section	Event	Acceptance Criteria	Impact
15.3	Decrease in Reactor Coolant Flow Rate		
15.3.1	Total Loss of Forced Reactor Coolant Flow (LOF)	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Event specific DNB will be maintained above the 95/95 DNBR Limit, thus fuel cladding integrity will be maintained. Pressure in the RCS and main steam system will be maintained below 110% of the design value. The loss of offsite power event plus a single failure will not result in a lower DNBR than that calculated for the loss of offsite power event alone. For the total loss of RCS flow event, the minimum DNBR occurs during the first few seconds of the transient and the reactor is tripped by the CPCs on low RCP shaft speed. Therefore, any single failure that would result in a lower DNBR during the transient would have to occur during the first few seconds of the event. None of the single failures listed in UFSAR Table 15.0-0 will have any effect on the transient minimum DNBR during this period.
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.3.2	Flow Controller Malfunction Causing a Flow Coastdown	N/A	This event does not apply to the Palo Verde System 80 design.
15.3.3.	Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power (SR)	Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.	No change to methodology required: Bounded by Single Reactor Coolant Pump Shaft Break with Loss of Offsite Power (Section 15.3.4).
		The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations. If the DNBR or falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model, which includes the potential adverse effects of hydraulic instabilities that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.	
		Any activity release must be such that the calculated doses at the site boundary (EAB) and low population zone (LPZ) area within the 10 CFR Part 100.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.3.4	Single Reactor Coolant Pump Shaft Break with Loss of Offsite Power (SS)	Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations. If the DNBR or falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model, which includes the potential adverse effects of hydraulic instabilities that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability. Any activity release must be such that the calculated doses at the site boundary (EAB) and low population zone (LPZ) area within the 10 CFR Part 100.	No change to methodology required: he maximum RCS and secondary side pressures due to a single RCP shaft break in combination with a LOP resulting from turbine trip will be maintained less than 110% of their design values. In the event of a single RCP shaft break or rotor seizure event, the radiological dose exposure at the site boundary and Low population zone will remain within 10 CFR 100 limits. During the first few seconds of the transient, the combination of decreasing flow rate and increasing RCS temperature results in a decrease in the DNBR of the fuel pins. The transient minimum DNBR does go below the Specified Acceptable Fuel Design Limit (SAFDL) for DNBR. The amount of predicted failed fuel is determined with the statistical convolution technique (see UFSAR Section 15.4.8.3.C). DNB Propagation is evaluated by verifying that the bounding fuel clad strain evaluation is still applicable.

UFSAR Section	Event	Acceptance Criteria	Impact
15.4	Reactivity and Power Distribution Anomalies		
15.4.1	Uncontrolled CEA Withdrawal from a Subcritical or Low (Hot Zero) Power Condition	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Event specific DNB will be maintained above the 95/95 DNBR Limit, thus fuel cladding integrity will be maintained. While the maximum LHGR exceeds 21 kw/ft for short time intervals, however, this is acceptable as the analysis verifies that the total integrated energy deposited in the fuel does not result in peak centerline fuel temperature to exceed the TS 2.1.1.2 limit. While no specific criteria are mentioned in the SRPs for peak pressure criteria, the pressure in the RCS and main steam system will be maintained below 110% of the design value.
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		Fuel centerline temperature (as specified in TS 2.1.1.2) is not exceeded.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.4.2	Uncontrolled CEA Withdrawal at Power (CEAW)	Peak RCS Pressure ≤110% of Design.	No change to methodology required: The pressure in the RCS and main steam system will be maintained below 110% of the design value. Fuel cladding integrity will be maintained by continuing to ensure the 95/95 DNBR limit is met. The maximum LHGR will remain below the value that causes peak centerline melt temperature (TS 2.1.1.2 limit). Fuel cladding degradation is not anticipated and there are no radiological consequences resulting from the event. This event would not result in any releases of radioactive material above that of a normal reactor trip.
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		Fuel centerline temperature (as specified in TS 2.1.1.2) is not exceeded.	
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.4.3	Single Full-Strength Control Element Assembly Drop	Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	No change to methodology required: Fuel cladding integrity will be maintained by continuing to ensure the 95/95 DNBR limit is met.
		Fuel centerline temperatures, as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point.	The maximum LHGR will remain below the value that causes peak centerline melt temperature (TS 2.1.1.2 limit). No unusual cladding strain is expected to occur for this event
		Uniform cladding strain as specified in SRP Section 4.2, subsection II.A.2(b) do not exceed 1%.	Fuel cladding degradation is not anticipated and there are no radiological consequences resulting from the event.

UFSAR Section	Event	Acceptance Criteria	Impact
15.4.4	Startup of an Inactive Reactor Coolant Pump (SIRCP)	Peak RCS Pressure ≤110% of Design.	No change to methodology required: The pressure increase during the event remains below LTOP limits during LTOP conditions and remains below 110% of design pressure during conditions above LTOP. The SIRCP event does not result in a return to criticality. Therefore, there is no significant increase in heat flux and therefore no decrease in minimum DNBR or increase in PCT (peak clad temperature). The SIRCP event does not generate a more serious plant condition during its sequence of events. Since DNBR and PCT SAFDLs are maintained, there is no fuel failure from this event and there are no dose consequences.
		Peak Secondary Pressure ≤110% of Design.	
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	
15.4.5	Flow Controller Malfunction Causing an Increase in BWR Core Flow	N/A	This event does not apply to the Palo Verde System 80 design.

UFSAR Section	Event	Acceptance Criteria	Impact
15.4.6	Inadvertent Deboration (ID)	Peak RCS Pressure ≤110% of Design.	No change to methodology required:
		Peak Secondary Pressure ≤110% of Design.	The peak RCS pressure and main steam system pressure is indirectly demonstrated to be satisfied by ensuring that an uncontrolled criticality will not occur while using the specified times for operator action.
		Fuel cladding integrity must be maintained so the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for pressurized-water reactors (PWRs) based on acceptable correlations.	The fuel SAFDLs are indirectly demonstrated to be satisfied by ensuring that an uncontrolled criticality will not occur while using the specified times for operator action.
		An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.	
		The operators must be alerted to an ongoing Boron Dilution at least 15 minutes prior to the loss of Shutdown Margin (SDM) for operation in Modes 1 (power operation) through 5 (cold shutdown) and 30 minutes for Mode 6 (refueling).	The radiological dose criteria are indirectly demonstrated to be satisfied by ensuring that an uncontrolled criticality will not occur while using the specified times for operator action.
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	The event is analyzed such that there is sufficient time available for the operator to detect and to terminate an ID event if it occurs.

UFSAR Section	Event	Acceptance Criteria	Impact
15.4.7	Inadvertent Loading of a Fuel Assembly into the Improper Position	To meet the requirements of GDC 13, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel-loading errors after fueling operations.	No change to methodology required: The plants Low Power Physics Testing (LPPT) procedures and the Power Ascension Testing (PAT) procedures capture the provision to use reactor instrumentation to search for potential fuel-loading errors. The predicted doses for the misload event were determined to be lower than or the same as those predicted for the AOO from the SAFDL event (remain below 10 percent of the 10 CFR Part 100 limits) and therefore the applicable dose limit is satisfied.
		In the event the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10 CFR Part 100 criteria. A small fraction is interpreted to be less than 10% of the 10 CFR Part 100 reference values. For the purpose of this review, the radiological consequences of any fuel-loading error should include consideration of the containment, confinement, and filtering systems. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies.	

UFSAR Section	Event	Acceptance Criteria	Impact
15.4.8	Control Element Assembly Ejection	The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed “Service Limit C” as defined in the ASME Code. For PVNGS this equates to 120% of design pressure or 3000 psia.	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. Refer to Section 6.5 for details on the methodology changes. The maximum reactor pressure during any portion of the assumed excursion will be maintained less than the value that will cause stresses to exceed “Service Limit C” as defined in the ASME Code. For PVNGS this equates to 120% of design pressure or 3000 psia. If an accident occurs, offsite radiological dose consequences will not exceed the10 CFR Part 100 guideline. Control room dose consequences will not exceed the limits specified by GDC 19.
		Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/ gm at any axial location in any fuel rod. Implicit in this limit is a requirement to stay below a fuel centerline temperature, which ensures fuel melting will not occur.	
		The fission product inventory in the fuel rods calculated to experience a departure from nucleate boiling condition is an input to the radiological evaluation. Offsite doses remain below 100% of the 10 CFR100 limits namely, 300 REM thyroid and/or 30 REM thyroid Control Room dose (whichever is limiting).	

UFSAR Section	Event	Acceptance Criteria	Impact
15.5	Increase in RCS Inventory		
15.5.1	Inadvertent Operation of the ECCS	Peak RCS Pressure ≤110% of Design.	No change to methodology required:
		Peak Secondary Pressure ≤110% of Design.	Pressure in the RCS and main steam system will be maintained below 110% of the design value.
		The pressure-temperature limits (PTLR) for brittle fracture of the RCS are not violated by this transient.	The shutdown cooling relief valves (LTOPs) will mitigate the pressure transient so that the temperature-pressure limits are not exceeded thus preventing brittle facture RCS.
15.5.2	CVCS Malfunction-Pressurizer Level Control System Malfunction with Loss of Offsite Power	Peak RCS Pressure ≤110% of Design.	No change to methodology required:
		Peak Secondary Pressure ≤110% of Design.	Pressure in the RCS and main steam system will be maintained below 110% of the design value.
		Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.	Fuel cladding integrity will be maintained by continuing to ensure the 95/95 DNBR limit is met.
		An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently.	For the moderate frequency (without an additional single failure), fuel cladding integrity will be maintained.
		An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. There shall be no loss of function of any fission product barrier other than the fuel cladding.	The overall DNB degradation experienced during a Pressurizer Level Control System (PLCS) malfunction event with a loss of offsite power results in an event similar to the Loss of Flow (LOF) event (Section 15.3.1).

UFSAR Section	Event	Acceptance Criteria	Impact
15.6	Decrease in Reactor Coolant System Inventory		
15.6.1	Inadvertent Opening of a Pressurizer Safety / Relief Valve	N/A	The inadvertent opening of a pressurizer safety valve event as described in NRC Standard Review Plan 15.6.1 is evaluated in the emergency core cooling systems analyses (Section 6.3).
15.6.2	Double-Ended Break of a Letdown Line Outside Containment (DBLLOCUS)	Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations. If the DNBR falls below this value, fuel failure must be assumed for all rods that do not meet this criterion; and the additional fission product activity from the fuel failure during the transient in the primary coolant shall be included in the dose consequences analysis. The radiological consequences must be within the SRP guidelines for a small fraction (less than 10%) of 10 CFR Part 100 exposure guidelines, and within 10 CFR Part 100 guidelines for the cases of a pre-accident iodine spike or one rod held out of the core.	No change to methodology required: If an accident occurs, offsite radiological dose consequences will not exceed a small fraction 10 CFR Part 100 guideline values or less. If an accident occurs, offsite radiological dose consequences will be within 10 CFR Part 100 guidelines for the cases of a pre-accident iodine spike or one rod held out of the core.

UFSAR Section	Event	Acceptance Criteria	Impact
15.6.3	Steam Generator Tube Rupture (SGTR)		
	Steam Generator Tube Rupture without a Loss of Offsite Power	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Pressure in the RCS and main steam system will be maintained below 110% of the design value. If an accident occurs, offsite radiological dose consequences will not exceed 10 CFR Part 100 guidelines for the cases of a pre-accident iodine spike. If an accident occurs, offsite radiological dose consequences will not exceed a small fraction of 10 CFR Part 100 guidelines for the cases of an accident generated iodine spike. Control room dose consequences will not exceed the limits specified by GDC 19.
		Peak Secondary Pressure ≤110% of Design.	
		For the SGTR event with a pre-accident iodine spike, the calculated dose should not exceed the 10 CFR 100 limits.	
		For the SGTR event with an accident generated iodine spike, the calculated dose should not exceed a small fraction of 10 CFR 100 limits.	
	Steam Generator Tube Rupture With a Loss of Offsite Power and a Single Failure	Peak RCS Pressure ≤110% of Design.	No change to methodology required: Pressure in the RCS and main steam system will be maintained below 110% of the design value. If an accident occurs, offsite
		Peak Secondary Pressure ≤110% of Design.	
		For the SGTR event with a pre-accident iodine spike, the calculated dose should not exceed the 10 CFR 100 limits.	

UFSAR Section	Event	Acceptance Criteria	Impact
	(SGTRLOP+SF)	For the SGTR event with an accident generated iodine spike, the calculated dose should not exceed the 10 CFR 100 limits. Control Room Dose ≤ GDC 19.	radiological dose consequences will not exceed the10 CFR Part 100 guideline for the cases of a pre-accident iodine spike. If an accident occurs, offsite radiological dose consequences will not exceed 10 CFR Part 100 guidelines for the cases of an accident generated iodine spike. Control room dose consequences will not exceed the limits specified by GDC 19.
15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	N/A	This event does not apply to the Palo Verde System 80 design.
15.6.5	Loss-Of-Coolant Accidents (LOCA)	This Event is addressed separately in Section 7. Palo Verde UFSAR Chapter 6.3 addresses the 10CFR 50.46 event	

UFSAR Section	Event	Acceptance Criteria	Impact
15.7	Radioactive Release from a Subsystem or Component		
15.7.1	Waste Gas System Failure	Offsite doses due to a GRS decay tank rupture are to be well within the guideline values of 10CFR100. Control Room Dose ≤ GDC 19.	No change to methodology required: If an accident occurs, offsite radiological dose consequences will not exceed “well within” 10 CFR Part 100 guideline values. Control room dose consequences will not exceed the limits specified by GDC 19.
15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	The plant is considered adequately designed against a radioactive liquid waste system leak or failure if the conservatively calculated exposures resulting from the release of radioactive gases from the system are small fractions of the 10 CFR Part 100 guideline values.	Bounded by Postulated Radioactive Release Due to Liquid-Containing Tank Failures (Section 15.7.3).
15.7.3	Postulated Radioactive Release Due to Liquid-Containing Tank Failures	The plant is considered adequately designed against a radioactive liquid waste system leak or failure if the conservatively calculated exposures resulting from the release of radioactive gases from the system are small fractions of the 10 CFR Part 100 guideline values.	No change to methodology required: If an accident occurs, offsite radiological dose consequences will not exceed a small fraction 10 CFR Part 100 guideline values. Control room dose consequences will not exceed the limits specified by GDC 19.

UFSAR Section	Event	Acceptance Criteria	Impact
15.7.4	Radiological Consequences of Fuel Handling Accident	The dose consequences are limited to “well within” 10 CFR part 100 limits. Control Room Doses ≤ GDC 19.	No change to methodology required: Section 6.7 addresses the fuel handling accident event. If an accident occurs, offsite radiological dose consequences will not exceed “well within” 10 CFR Part 100 guideline values. Control room dose consequences will not exceed the limits specified by GDC 19.
15.7.5	Spent Fuel Cask Drop Accidents	N/A	The probability of fuel handling accidents in the fuel building that result from dropping a Transportable Storage Canister/Transfer Cask (TSC/TFR) containing spent fuel or other heavy load from the single failure proof Cask Handling Crane is sufficiently small that they are not credible events, and therefore do not require analysis.

UFSAR Section	Event	Acceptance Criteria	Impact
15.E	Limiting Infrequent Event		
15.E	Limiting Infrequent Event	An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations are acceptable.	No change to methodology required: Section 6.6 provides a validation of the currently approved methodology with Framatome CE16HTP fuel design including the use of statistical convolution of uncertainties in determining number of fuel failures. If an accident occurs, offsite radiological dose consequences will not exceed a small fraction (10%) of the10 CFR Part 100 guideline. Control room dose consequences will not exceed the limits specified by GDC 19.
		Offsite radiological dose consequences are limited to a small fraction, or 10%, of 10 CFR Part 100 guideline values. Additionally, radiation exposures for control room personnel are subject to the limits specified in General Design Criterion 19 of 10 CFR 50 Appendix A.	

6.2. DNB Propagation

DNB propagation is a concern when the internal pin pressure is greater than the RCS pressure. Under these conditions, degraded heat removal from the fuel pin (due to DNBR < DNBR SAFDL) may cause the fuel pin to balloon, resulting in flow blockage. This flow blockage could then cause additional fuel pins to enter DNBR conditions and fail.

Reference 6.3 (Appendix A, Table 3-3) determined that the maximum cladding strain for CE 14X14 fuel during DNB conditions did not exceed [[[redacted]]]. The CE 14X14 fuel represents a limiting case for DNB propagation resulting from cladding strain and has become the bounding values of strain for fuel in CE designed NSSS. Thus, the [[[redacted]]] limit is conservative for PVNGS 16X16 assemblies.

For CE16HTP fuel with M5® cladding, an analysis was performed to develop criteria that assures that DNB propagation does not occur for M5® clad fuel pins [[[redacted]]].

[[[redacted]]].

Figure 6-1: [[[redacted]]]

[[

[redacted]

]]

Figure 6-2: [[[redacted]]]

[[

[redacted]

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[[[redacted]]].

6.3. Fuel Failure Prediction Using DNB Statistical Convolution

The NRC Safety Evaluation for topical report CENPD-183-A, C-E Methods for Loss of Flow Analysis (Reference 6.1) establishes a method for determining fuel failure using DNB statistical convolution. For this determination, CENPD-183-A establishes a link between fuel assembly design, computer codes used for analysis, the Critical Heat Flux (CHF) correlation, and the DNB probability distribution function (pdf).

Regarding the Framatome fuel design, [[[redacted]]]. This will result in a different pdf than those previously approved by the NRC and shown in Table 2 of CENPD-183-A.

The key to the use of the DNB statistical convolution methodology is developing the pdf of exceeding DNB with respect to DNBR. Per the NRC Safety Evaluation approving CENPD-183-A:

> *"Since experimental evidence indicates that fuel cladding failure is not necessarily coincident with a short duration of DNB, we conclude that the statistical convolution technique is conservative and acceptable provided that the probability distribution for DNB is acceptable."*

[[[redacted]]].

Section 6.6 presents the AOO from SAFDL event, which demonstrates use of DNB statistical convolution to determine fuel failure for Framatome CE16HTP fuel.

6.4. Summary of Cladding Related Models in the Non-LOCA Transient Evaluation Models

As discussed in the M5® Topical (Reference 6.5, Section 4.1) and the [[[redacted]]], the cladding material properties potentially impacted in Non-LOCA transient analysis system response computer codes listed in this section are cladding thermal conductivity and specific heat. [[[redacted]]].

6.4.1. Cladding Thermal Conductivity

As described in Section 6.5, for use in CEA ejection, [[[redacted]]].

6.4.2. Cladding Specific Heat

As evaluated in Section 6.5 for use in CEA ejection, the M5® specific heat is [[redacted]]] for specific heat.

6.4.3. Fuel-to-Clad Gap Coefficient of Conductance

Per Section 4.2 of this Attachment, the current values utilized in UFSAR Chapter 15 remain applicable for use with the CE16HTP fuel design, when the COPERNIC generated fuel-to-clad gap coefficient of conductance (Hgap) is modeled in conjunction with the conservative methods and models in the current licensing basis.

6.4.4. CENTS Code

The CENTS computer code (Reference 6.7) is an interactive, faster than real time computer code for the simulation of the NSSS and related systems. It is capable of calculating the behavior of a PWR for both normal and abnormal conditions, including accidents. The CENTS code is approved for use for transient analyses; refer to UFSAR (Reference 6.8) Section 15.0.3.1.3.2.

A review of CENTS indicated that the cladding material properties employed are cladding thermal conductivity and specific heat. As discussed in the preceding paragraphs, [[[redacted]]]. This approach is consistent with [[[redacted]]] (Reference 6.6, Section 7.2). Consequently, [[[redacted]]].

6.4.5. HERMITE Code

HERMITE (Reference 6.9) is a space-time kinetics computer code. HERMITE was developed for the analysis of design and off-design transients in PWRs by means of a numerical solution to the multi-dimensional, few-group, time dependent neutron diffusion equation including feedback effects of fuel temperature, coolant temperature, coolant density and control rod motion. The heat conduction equation in the fuel pellet, gap and clad is solved by a finite difference method. Continuity and energy conservation equations are solved for the coolant enthalpy and density. HERMITE code is approved for use for Loss of Flow transient analyses; refer to UFSAR (Reference 6.8) Section 15D.2.4.

A review of HERMITE indicated that the cladding material properties employed are cladding thermal conductivity and specific heat. As discussed in the preceding paragraphs, [[[redacted]]]. This approach is consistent with [[[redacted]]] (Reference 6.6, Section 7.2). Consequently, [[[redacted]]] are needed.

6.4.6. STRIKIN-II Code

Refer to Section 6.5.

6.5. CEA Ejection Analysis

UFSAR Section 15.4.8 describes the Control Element Assembly (CEA) Ejection event. The analysis has been updated to determine the impact of the implementation of a transition or full core of Framatome CE16HTP fuel. Each aspect of the CEA Ejection event methodology is evaluated for potential impact by the change to CE16HTP fuel. Differences that are evaluated include:

(1) Change from Zircaloy-4, ZIRLO® or Optimized ZIRLO™ cladding to M5® cladding.

(2) Change from the FATES3B fuel performance code to the COPERNIC fuel performance code. The use of the COPERNIC code explicitly accounts for thermal conductivity degradation (TCD) when evaluating CE16HTP fuel.

(3) Change from the STRIKIN-II code with the CE-1 CHF correlation to the VIPRE code with the BHTP CHF correlation to evaluate fuel failure.

6.5.1. M5® Cladding Impact on CEA Ejection Analysis

The deposited energy acceptance criteria evaluation for this event is performed using the STRIKIN-II code (Reference 6.11) to determine the energy deposited in the fuel rods by an ejected CEA at various plant conditions and power levels. The analysis uses the basic methodology described in CENPD-190-A (Reference 6.12). The methodology has been supplemented as required by CENPD-404-P-A Sections 7.2.1 and 7.3.1 (Reference 6.6) for the use of ZIRLO® and Optimized ZIRLO™ clad fuel rods. [[[redacted]]].

To implement the Framatome M5® cladding material, the material properties of the M5® alloy were reviewed to determine the impact to the CEA Ejection analysis. The thermo-physical, mechanical, and corrosion properties of M5® are discussed in the M5® topical report (Reference 6.13). Consistent with the ZIRLO® topical, [[[redacted]]].

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

[redacted]

[redacted]

[redacted]

[redacted]]].

However, for transients in which the cladding temperature may enter the phase change range the impact of M5® must be considered due to the differences in heat capacities. The M5® topical report identified CEA Ejection as the accident likely to result in cladding temperature that enters the phase change range of the M5® material.

The heat capacities of M5® are modeled in the CEA Ejection analysis via user input to the STRIKIN-II code. [[[redacted]]]. The specific derivation of the M5® alloy's specific heat thermal-mechanical properties are provided below. Section I.5 of Reference 6.13 provides the following correlations for M5® specific heat based on testing data:

[[

[redacted]]].

The cladding specific heat data used in STRIKIN-II is volumetric specific heat (Btu/ft^3-^{0}F) versus temperature (°F). The M5® specific heat inputs for the STRIKIN-II code are developed based on the correlation and the M5® alloy density. Note that the M5® volumetric specific heat data is generated [[[redacted]]] of the cladding material, consistent with the STRIKIN-II topical report (Appendix I). The volumetric specific heat data of the three cladding materials Zircaloy-4, ZIRLO®, and M5® are compared in Table 6-2 and Figure 6-3.

Table 6-2: Volumetric Specific Heat Data for Zircaloy-4, ZIRLO®, and M5®

[[

[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

]]

Figure 6-3: Cladding Volumetric Specific Heat Inputs to STRIKIN-II for Zircaloy-4, ZIRLO®, and M5®

[[

[redacted]

]]

6.5.2. CEA Ejection Transient Analysis Benchmark Cases

The CEA Ejection analysis performed benchmark cases to compare the impact of the COPERNIC-generated fuel performance input versus the FATES3B-generated fuel performance input. The benchmark cases used the maximum fuel rod radial average and maximum incipient centerline melting enthalpies as points of comparison.

The fuel performance data for input to STRIKIN-II was provided by the COPERNIC code in lieu of the FATES3B code (References 6.14, 6.15, and 6.16). The COPERNIC code provides the data to STRIKIN-II in the same interface format as provided by the FATES3B code. The benchmark cases were performed at hot full power at various cladding and burnup points to demonstrate the impact of the generic modeling differences between the FATES3B and COPERNIC codes as well as the impact of the Thermal Conductivity Degradation (TCD) modeled in the COPERNIC code. No changes were made to the actual STRIKIN-II code.

The COPERNIC to FATES3B benchmark demonstrated the following:

1. The COPERNIC-based output data are acceptable for use in the STRIKIN-II code.
2. The modeling differences between the two codes can lead to differences in the centerline temperature at lower burnups, consistent with what has been observed in the industry.

3. At the region of interest to CEA Ejection Analysis (i.e., burnup > 12,000 MWD/MTU), the FATES3B and COPERNIC codes predict temperature in a consistent manner, with the COPERNIC temperature being higher due to TCD.
4. The limiting Hot Rod burnup point for the STRIKIN-II code analysis should represent the burnup point at the knee of the radial falloff curve to fully absorb the effect of TCD.

6.5.3. CEA Ejection Transient Analysis Process

UFSAR Section 15.4.8 describes the CEA Ejection event. The analysis has been updated to determine the impact of the implementation of a transition or a full core of Framatome CE16HTP fuel into the APS core. The methodology is consistent with that used in the current analysis presented in UFSAR Section 15.4.8.2 except for use of the COPERNIC code in lieu of the FATES3B code (as discussed in Section 4), and use of the VIPRE code with the BHTP CHF correlation in lieu of the STRIKIN-II code with the CE-1 CHF correlation to evaluate fuel failure (as discussed in Section 5).

The analysis performed explicit calculations for the maximum radially averaged fuel enthalpy and fuel centerline temperature. The analysis also performed a DNBR evaluation to demonstrate that the transient DNBR and radiological dose consequences of the event will be bounded by the current UFSAR cases. The DNBR analysis was performed using the STRIKIN-II code in conjunction with the VIPRE code. The STRIKIN-II code was used to generate the state parameters during the CEA Ejection event. The limiting state parameters were then used by the VIPRE code with the BHTP CHF correlation to generate the DNBR values during the event.

The CEA ejection event did not require the use of the CENTS computer code (Reference 6.10) cases to reanalyze the system response as system model input changes due to Framatome CE16HTP fuel were small and the impact on the overall transient system response is insignificant. As the overall transient system response remains the same as the current analysis, the peak reactor coolant system pressures reported in UFSAR Section 15.4.8 remain bounding.

6.5.4. CEA Ejection Transient Analysis Results

6.5.4.1. Fuel Enthalpy and Temperature Evaluation Results

Fuel Enthalpy evaluations were performed at Hot Full Power and Hot Zero Power (HZP). The initial power for the HZP cases was set to 20% of full power. Intermediate power levels were evaluated to validate the Power Dependent Insertion Limits (PDILs).

The fuel enthalpy and centerline temperature evaluation results for CE16HTP fuel are shown in Table 6-3. As seen in Table 6-3, the calculated maximum radially averaged fuel enthalpy for the limiting case is less than the acceptance criterion of 280 cal/gm, and the peak fuel centerline temperature is less than the fuel melt temperature for Framatome fuel (which varies as a function of burnup). Table 6-3 also presents the current analysis of record results for CE16STD and CE16NGF.

Table 6-3: CEA Ejection Fuel Enthalpy and Temperature Evaluation Results

Parameter	Acceptance Criteria	CE16STD Analysis Results	CE16NGF Analysis Results	CE16HTP Analysis Results
Hot Full Power Maximum Radially Averaged Fuel Enthalpy, cal/gm	≤ 280	136.22	135.64	141.04
Hot Zero Power Maximum Radially Averaged Fuel Enthalpy, cal/gm	≤ 280	155.85	149.70	157.69
Hot Full Power Peak Fuel Centerline Temperature (°F) [Fuel Specific Limit]	No melt	4800 [4810]	4726 [4730]	4757 [4843]
Hot Zero Power Peak Fuel Centerline Temperature (°F) [Fuel Specific Limit]	No melt	4777 [4810]	4726 [4730]	4806 [4843]

6.5.4.2. Fuel Failure Evaluation Results

A DNBR evaluation was performed to determine the limiting DNBR values during the event. The DNBR values were used to generate the fuel failure percentage because of the system being in DNB during the event.

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.

Since the system responses to transient is unchanged for Framatome fuel (i.e., the duration for the system being in DNB being less than 4.5 seconds), it is also concluded that like the UFSAR cases, the system will not undergo DNBR propagation and coolable geometry is maintained.

6.6. AOO from SAFDL

UFSAR Appendix 15.E describes a composite event that is evaluated to bound all infrequent events, including Anticipated Operational Occurrences (AOOs) in combination with a single active failure, with respect to the degradation in the Departure from Nucleate Boiling Ratio (DNBR). The limiting infrequent event is a loss of flow from a specified acceptable fuel design limit (SAFDL). The Loss of Flow from SAFDL analysis has been updated to evaluate the implementation of Framatome CE16HTP fuel in combination with the BHTP CHF correlation (Reference 6.2).

The methodology is the same as that used in the current analysis presented in UFSAR Section 15.E.2 except for the following: 1) the SAFDL condition at time zero is calculated using the VIPRE code with the BHTP CHF correlation and the BHTP CHF DNBR limit, and 2) the DNBR values during the transient are calculated by the VIPRE code with the BHTP CHF correlation. As discussed in Section 6.4.5, no changes to the HERMITE computer code are needed to accommodate Framatome CE16HTP fuel; therefore, as the transient response is not impacted by fuel type, no HERMITE computer code cases were reanalyzed. The analysis as described in UFSAR Appendix 15.E remains applicable for the implementation of Framatome CE16HTP fuel.

As the overall transient system response remains the same as the current analyses, only the DNB margin analysis is affected. With the implementation of CE16HTP fuel, a minimum BHTP DNBR of 1.008 is calculated for the loss of flow from SAFDL. The fuel failure for this event was calculated based on the DNB statistical convolution methodology discussed in Section 6.3 using the BHTP probability density function. The calculated fuel failure for the event is bounded by the current analysis.

The DNB propagation for this event was evaluated using the current APS methodology discussed in Section 6.2. Since the event was in DNB for only 3.8 seconds, which is less than the 4.5 seconds DNB propagation criterion, it is concluded that the DNB will not propagate for this event.

The results of the loss of flow from a specified acceptable fuel design limit event support both transition and full core implementation of CE16HTP fuel.

6.7. Fuel Handling Accident

The fuel handling accident (FHA) may occur in either the Fuel Building or inside the Containment. Relative to the existing FHA dose analyses of record, the introduction of Framatome CE16HTP fuel could only affect the FHA source term. The FHA source term is dependent on the fuel assembly isotope inventory in the fuel rod gap spaces, the number of damaged fuel rods, and the fuel rod gap release fractions.

Per Section 9 of this Attachment, the Framatome CE16HTP fuel has been found to introduce no changes that would affect the source terms as described in Chapter 15 of the UFSAR. The FHA is currently evaluated assuming damage to all 236 fuel rods in the dropped fuel assembly. An evaluation confirms that damage to all 236 fuel rods in the dropped fuel assembly bounds the potential damage to a dropped CE16HTP fuel assembly.

PVNGS Operating License Amendment 153 (Reference 6.17) approved a deviation from Regulatory Guide (RG) 1.25 (Reference 6.18) to allow use of 'peak assembly average fuel pin pressure is < 1200 psig' in place of 'maximum fuel rod pressurization is 1200 psig'. This approach allows a few fuel rods to exceed the 1200 psig maximum pressurization while still maintaining the conservative iodine decontamination factor (DF) value specified by RG 1.25. Fuel rod pressures for Framatome CE16HTP fuel with M5® cladding and with and without gadolinia, based on 72 hours of in-core hold time, have been calculated using the approved COPERNIC fuel rod design computer code. The peak assembly average fuel pin pressure for Framatome CE16HTP fuel is less than 1200 psig, and consequently the CE16HTP fuel rod gap release fractions are bounded by the CE16STD and CE16NGF fuel rod gap release fractions.

Therefore, a FHA involving the dropping of a Framatome CE16HTP fuel assembly is no more severe than an FHA involving the dropping a CE16STD or CE16NGF fuel assembly.

6.8. Non-LOCA Transient Modeling Review Conclusions

The Palo Verde UFSAR Chapter 15 Non-LOCA transient analyses were reviewed for potential impacts resulting from the unrestricted use of Framatome CE16HTP fuel. Non-LOCA transient events will model [[[redacted]]].

6.9. References

6.1. CENPD-183-A, *Loss of Flow, C-E Methods for Loss of Flow Analysis*, June 1984.

6.2. BAW-10241(P)(A), Revision 1 as amended by Errata 1P-001 (February 2016), *BHTP DNB Correlation Applied with LYNXT*, July 2005.

6.3. License Amendment 205 to PVNGS Units 1, 2 and 3 Operating Licenses to revise Technical Specifications to Support the Implementation of Next Generation Fuel, January 23, 2018 (ADAMS Accession No. ML17319A103).

6.4. CEN-372-P-A Task 617, Revision 0, *Fuel Rod Maximum Allowable Gas Pressure*, May 1990.

6.5. BAW-10227NP-A, Revision 1, *Evaluation of Advanced Cladding and Structural Material (M5®) in PWR Reactor Fuel*, approved by the NRC in June 2003.

6.6. CENPD-404-P-A, Revision 0, *Implementation of ZIRLO® Cladding Material in CE Nuclear Power Fuel Assembly Designs*, November 2001.

6.7. WCAP-15996-P-A, Revision 1 (CENPD-282-P-A, Revision 2), *Technical Description Manual for the CENTS Code*, Volumes 1 through 4, March 2005.

6.8. PVNGS UFSAR, *Updated Final Safety Analysis Report (UFSAR)*, Revision 19 [Change A].

6.9. CENPD-188-A, *A Multi-Dimensional Space-Time Kinetics Code for PWR Transients*, July 1976.

6.10. WCAP-15996-P-A, Revision 1, *Technical Description Manual for the CENTS Code*, November 2005.

6.11. CENPD-135-P, *STRIKIN-II Cylindrical Geometry Fuel Rod Heat Transfer Program*, August 1974.

6.12. CENPD-190-A, *C-E Method for Control Element Assembly Ejection Analysis*, July 30, 1976.

6.13. BAW-10227P-A, *Evaluation of Advanced Cladding and Structural Material (M5®) in PWR Reactor Fuel*, February 2000.

6.14. CENPD-139-P-A, *C-E Fuel Evaluation Model*, July 1974.

6.15. CEN-161(B)-P-A, *Improvements to Fuel Evaluation Model*, August 1989.

6.16. CEN-161(B)-P, Supplement 1-P-A, *Improvements to Fuel Evaluation Model*, January 1992.

6.17. License Amendment 153 to PVNGS Units 1, 2 and 3 Operating Licenses to change Maximum Fuel Pin Pressurization Criteria used in the Evaluation of the Design Basis Fuel Handling Accident, September 27, 2004, ADAMS Accession No. ML042720349.

6.18. NRC Regulatory [Safety] Guide 1.25, Revision 0, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors*, March 1972.

7. ECCS PERFORMANCE LOCA ANALYSIS

The Small Break and Realistic Large Break LOCA analyses to support the introduction of the CE16HTP design for PVNGS are contained within separate Licensing Summary Reports. The licensing analyses are performed in accordance with the NRC-approved methods of References 7.1, 7.2, and 7.3. The detailed evaluation of the margins to 10 CFR 50.46(b) acceptance criteria is reported in separate Framatome Licensing Summary Reports.

Attachments 9 and 11 provide non-proprietary and proprietary versions of these Licensing Summary Reports, respectively.

7.1. Post-LOCA Long-Term Cooling Analysis

Post-LOCA long-term cooling (LTC) compliance is evaluated relative to the introduction of Framatome CE16HTP fuel assemblies. An evaluation of both mixed and full core scenarios was performed.

The License Amendment Request (LAR) supporting implementation of Westinghouse Next Generation Fuel at PVNGS identified how the post-LOCA LTC analyses support current operation with CE16STD fuel and the implementation of CE16NGF. The LTC evaluation model consists of a boric acid precipitation (BAP) analysis and a decay heat removal (DHR) analysis. The precipitation analysis demonstrates that the maximum boric acid concentration remains below the solubility limit, ensuring that precipitation does not occur. The decay heat removal analysis demonstrates that that the decay heat in the fuel can be removed for the long term and that the core remains covered with two-phase liquid, ensuring that temperatures remain acceptably low.

7.1.1. Post-LOCA Boric Acid Precipitation Analysis

Per the Safety Evaluation associated with the issuance of the PVNGS NGF license amendment (Reference 7.4, Section 3.5.8.4), the NRC staff found that an acceptable boric acid precipitation analysis has been performed for PVNGS and concluded that boric acid precipitation can be prevented during a postulated LOCA event at PVNGS for both CE16STD fuel and CE16NGF.

The core mixing volume is a key parameter for the BAP analysis in that the smaller the mixing volume, the shorter the time to reach the BAP solubility limit. The core volume at steady-state conditions is unchanged relative to CE16STD fuel and therefore, the core mixing volume will be unchanged for CE16HTP fuel relative to CE16STD fuel. Accordingly, the time to reach the BAP solubility limit (i.e., the time by which simultaneous hot and cold side injection is required) with CE16HTP fuel will be the same as the CE16STD fuel. Therefore, it can be concluded that the BAP analyses performed for the CE16STD fuel are equally applicable to CE16HTP fuel in either a mixed-core or full-core capacity, and the implementation of CE16HTP fuel at Palo Verde will not require a change to the post-LOCA operator action time for initiating simultaneous hot/cold side injection in the Emergency Operating Procedures.

7.1.2. Post-LOCA Decay Heat Removal

Per the Safety Evaluation associated with the issuance of the PVNGS NGF license amendment (Reference 7.4, Section 3.5.8.4), the NRC staff concluded that the decay heat removal analysis, in combination with the boric acid precipitation analysis, ensures that the 10 CFR 50.46(b)(5) criterion for LTC has been adequately met for PVNGS. The Palo Verde decay heat removal analysis demonstrates that decay heat can be removed in the long-term for any size LOCA, and that plant operators can correctly identify and initiate an appropriate means of long-term decay heat removal, as follows:

- For small break sizes, the RCS will refill and the shutdown cooling system entry temperature will be met, thereby allowing plant operators to use the shutdown cooling system for decay heat removal.
- For large break sizes, plant operators can utilize simultaneous hot/cold side injection to provide sufficient decay heat removal via the break flow.

Each of the important input parameters to the DHR analysis (e.g., RCS, steam generator, and condensate storage parameters) remain applicable for CE16HTP fuel. Therefore, the results presented in the NGF LAR for CE16STD fuel and CE16NGF are equally applicable when Framatome CE16HTP fuel is inserted into the core in either a mixed-core or full-core capacity.

7.2. New Time Requirement for LOCA Mitigation Operator Action

As discussed in Section 4.3 of the Small Break LOCA Licensing Summary Report, for plants such as PVNGS that do not have an automatic RCP trip, a delayed RCP trip can potentially result in a more limiting condition than tripping the RCPs at reactor scram. Continued operation of the RCPs can result in earlier loop seal clearing (LSC) with associated two-phase flow out the break, which would result in less inventory loss out the break early in the transient, but in the longer term could result in more overall inventory loss out the break. It has been postulated that tripping the pumps when the minimum RCS inventory occurs could cause a collapse of voids in the core, thus depressing the core level and provoking a deeper core uncovery, and a potentially higher peak clad temperature. Therefore, the methodology prescribes an RCP trip study for both the cold and hot leg breaks consistent with the plant licensing basis and Emergency Operating Procedures. For Palo Verde, a RCP trip time of 5 minutes following loss of subcooling margin at an assumed pressure of 1471 psia was analyzed to demonstrate 10 CFR 50.46(b)(1-4) criteria. The results of the RCP trip cases indicate that there is at least 5 minutes of expected operator time to trip all four RCPs after NPSH criteria are met with considerable margin to the 10 CFR 50.46(b)(1-4) criteria.

Standard Post-Trip Actions Procedure 40EP-9EO01[1] provides those operator actions, including immediate actions, which must be accomplished following an automatic or manually initiated reactor trip and the Diagnostic Actions necessary to determine a preliminary diagnosis of the

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[1] The Standard Post-Trip Action procedure incorporates the NRC guidance on human actions (HAs) and human factors engineering (HFE) and is maintained in compliance with said guidance.

event(s). The requirement to stop all RCPs when pressurizer pressure drops below the RCP NSPH limits is unchanged from the existing step 5.4 of this Procedure. Since the procedure step currently exists (without a specific time limit), the operators are already trained on and familiar with the procedure step, and the PVNGS simulator is capable of modeling the conditions under which the action would be required. Therefore, actual performance of the step is feasible and reliable. The requirement to stop all Reactor Coolant Pumps within 5 minutes following pressurizer pressure dropping below the RCP NPSH limits during a Small Break LOCA event will be added to Procedure 40DP-9ZZ04, *Time Critical Action (TCA) Program*, which provides a means to document periodic validation of credited action items, and a means to ensure that changes to the plant or to procedures or protocols do not invalidate credited action items.

Thus, the requirement to stop all RCPs within 5 minutes following pressurizer pressure dropping below the RCP NPSH limits during a SBLOCA event is not creating a new required operator action or safety-related operator action, but rather adding a new event limit (amount of time available) onto the existing action. As noted in the preceding paragraph, the operator action is already in the plant operating procedures and has been structured to account for human actions (HAs). The addition of an event limit for the SBLOCA event does not change the HAs but rather with its addition to the TCA program provides a method to validate the HAs and prevent unintended changes to the HA due to future plant modifications or unrelated procedure changes. As this is not a new action, there are no changes to the operator actions, operating procedures, control room displays or controls, or simulator. The commitment to enter the new time requirement into the time critical action program ensures that any needed changes to the operator training program due to the new time requirement are addressed. Further, five operating crews were evaluated on the new event limit, and the average completion time was well under one minute.

No new Operator actions are required. No existing Operator actions are required to change.

7.3. Generic Safety Issue 191

Following a LOCA, a debris mix could collect on the sump screen and potentially create sufficient resistance to recirculating flow that long-term core cooling may be challenged. There is also concern about the consequences of the debris that may pass through the sump screen. This debris could be ingested into the ECCS and flow into the reactor coolant system (RCS). This passed debris may collect on the fuel. These concerns have been broadly grouped under Generic Safety Issue 191 (GSI-191) (Reference 7.5).

Although all PWRs have made significant modifications to enhance long-term cooling (LTC) performance in the presence of post-LOCA debris (e.g., installation of larger ECCS recirculation strainers, modifications to thermal insulation inside containment), many PWR licensees, including APS, have not fully resolved all issues associated with GSI-191.

APS is continuing its efforts to resolve concerns associated with GSI-191 in accordance with the policy outlined by the Commission in its Staff Requirements Memorandum to SECY-12-0093 (References 7.6 and 7.7). APS acknowledges that final resolution of GSI-191 issues is necessary to obtain adequate confidence in the LTC plan in the presence of post-LOCA debris.

Consistent with a prior commitment associated with implementation of the Westinghouse CE16NGF next generation fuel design, APS commits to reflecting the change in Framatome CE16HTP fuel design as part of its final resolution of GSI-191 issues.

7.4. 10 CFR 50.46 Reporting

Fuel supply vendors are responsible for assessing peak clad temperature (PCT) for their fuel so that it may be addressed in the licensee-submitted 10 CFR 50.46 reports. During mixed core operation using fuel supplied by both Framatome and WEC, Framatome will provide a PCT assessment as to how the co-resident Westinghouse CE16STD or CE16NGF fuel affects the Framatome CE16HTP PCT calculations, and Westinghouse will provide a PCT assessment as to how the co-resident Framatome CE16HTP fuel affects the Westinghouse CE16STD or CE16NGF PCT calculations.

7.5. References

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

7.2. EMF-2328(P)(A), Revision 0, *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*, March 2001.

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

7.4. Letter from Siva Lingam (NRC) to Robert Bement (APS) of January 23, 2016, *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Issuance of Amendments to Revise Technical Specifications to Support the Implementation of Next Generation Fuel (CAC NO. MF8076, MF8077, and MF80738; EPOD L-2016-LLA-0005)*, (ADAMS Accession No. ML17319A103).

7.5. Generic Safety Issue 191 (GSI-191), *Assessment of Debris Accumulation on Pressurized Water Reactor (PWR) Sump Performance*.

7.6. U.S. Nuclear Regulatory Commission, *Staff Requirements - SECY-12-0093 – Closure Options for Generic Safety Issue -191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance*, December 14, 2012 (ADAMS Accession No. ML12349A378).

7.7. U.S. Nuclear Regulatory Commission, SECY-12-0093, *Closure Options for Generic Safety Issue -191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance*, July 9, 2012 (ADAMS Accession No. ML121320270).

8. CONTAINMENT RESPONSE ANALYSIS

8.1. Mass and Energy Release Analysis for Loss-of-Coolant Accidents

The Large Break LOCA (LBLOCA) Mass and Energy (M&E) analysis examines three major phases during the event: Blowdown, Reflood/Post-Reflood and Long Term Boiloff. An evaluation shows that there are no changes due to the Framatome CE16HTP fuel assemblies that affect the blowdown evaluation. The changes in core pressure losses are negligible for the LBLOCA blowdown M&E release Analysis of Record (AOR) and will have no significant effect on the transient results. The fuel performance parameters for CE16HTP fuel assemblies are bounded by the AOR fuel performance inputs in the blowdown analysis. Evaluations also show that there are no changes due to the CE16HTP fuel assemblies that affect the reflood/post-reflood and long term boiloff evaluations, and they therefore remain applicable. Therefore, the Westinghouse originated LBLOCA M&E Release AORs and their evaluations remain applicable for PVNGS Units 1, 2 and 3 with a full core of Framatome CE16HTP fuel assemblies.

8.2. Mass and Energy Release Analysis for Main Steam Line Break Accidents

The Main Steam Line Break (MSLB) M&E releases are calculated at 102%, 75% and 0% (i.e., no-load) power levels. Several M&E cases were run to accommodate the evaluation of peak containment pressures, containment Equipment Qualification (EQ) temperatures, outside containment peak pressures and outside containment EQ temperatures. NUREG-0800 Standard Review Plan (SRP) 6.2.1.4 (Reference 8.1) requires that the MSLB M&E releases account for stored energy in the affected steam generator (SG) metal, feedwater line, the steam line, the water in the SG, the feedwater transferred to the affected SG prior to the closing of the main feedwater isolation valve (MFIV), steam from the unaffected SG prior to the closing of the main steam isolation valves (MSIVs), and the energy in the primary coolant. The SGNIII code accounts for all of the SRP 6.2.1.4 requirements.

The change to Framatome CE16HTP fuel does not affect any of the following: the reactor coolant system (RCS), secondary side metal mass, component volumes, plant protection system including closure time of the main feedwater isolation valves and the main steam isolation valves. The SGNIII code calculates the feedwater flow, steam flow and break flow based on the code input. The code input values that generate the core stored energy are the only parameters that need to be evaluated for the fuel change. Evaluations show that the SGNIII input data used in the AOR will continue to equal or exceed the MSLB source energy values related to the Framatome CE16HTP fuel assemblies. Therefore, the Westinghouse originated MSLB M&E Release AORs and their evaluations remain applicable for PVNGS Units 1, 2 and 3 with a full core of Framatome CE16HTP fuel assemblies.

8.3. Mass and Energy Release for Containment Subcompartment Line Breaks

The tributary line breaks 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. Therefore, the Westinghouse originated tributary line break M&E Release AOR and its evaluations remain applicable for PVNGS Units 1, 2 and 3 with Framatome CE16HTP fuel assemblies.

8.4. References

8.1. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (SRP), *Section 6.2.1.4, Mass and Energy Release Analysis for Postulated Sectondary System Pipe Ruptures*, Revision 2, March 2007 (ADAMS Accession No. ML070620010).

9. RADIOLOGICAL SOURCE TERM EVALUATION

Source Terms for evaluating the radiological consequences of postulated accidents are based on methodology as described in Chapter 15 of the UFSAR. Framatome fuel design parameters (e.g., initial uranium mass, burnup, power factors, and operating histories) are essentially equivalent to those for current CE16STD fuel. The change from other zirconium-based cladding to M5® cladding is not only a minor contributor to the source term, but since the elemental compositions of the claddings are similar, the impact of the cladding on the source terms is insignificant and will not change the core isotopic distribution assumed in post-accident conditions. Additionally, system model input changes due to Framatome fuel are small and the impacts to overall transient system responses are insignificant. Based on these facts CE16HTP fuel has been found to introduce no changes that would affect Chapter 15 source terms.

UFSAR Section 11.1 provides Reactor Coolant System specific activities for 1% failed fuel conditions. These activities are significantly more conservative than the Technical Specification LCO 3.4.17 RCS specific activity limits for Dose Equivalent I-131 and Dose Equivalent Xe-133.

10. RADIOLOGICAL ACCIDENTS EVALUATION

As addressed in Section 6 of this Technical Analysis, all limiting offsite and control room dose consequences reported in UFSAR Chapters 15 and 6.4.7 remain bounding and applicable with Framatome fuel.

11. COLSS/CPCS SETPOINTS ANALYSIS

Setpoints analysis uses the Modified Statistical Combination of Uncertainties (MSCU) methodology (Reference 11.1), as the basis for stochastically combining uncertainty terms to calculate Core Operating Limits Supervisory System (COLSS) DNB Power Operating Limit (POL) and Core Protection Calculator System (CPCS) DNBR addressable uncertainty constants.

The MSCU methodology ensures that the COLSS DNB POL calculations and the CPCS DNBR calculations will be conservative to at least a 95% probability and a 95% confidence level. COLSS and CPCS include thermal hydraulic algorithms derived from the CETOP-D T-H code. The Plant COLSS and CPCS Computer Systems utilize the CE-1 critical heat flux (CHF) correlation. The CETOP-D model is benchmarked to the detailed (e.g., TORC or VIPRE) T-H code model to determine factors that correct CETOP-D results. These CETOP-D adjustment factors are input to Digital COLSS/CPCS setpoints analysis.

The setpoint analysis uses the same CETOP-D model that was benchmarked to the detailed T-H code model along with the associated CETOP-D adjustment factors from that benchmarking. The MSCU methodology then compares the results of the CETOP-D model to the results of the COLSS and CPCS Simulator code models to determine the addressable constants that are installed into the plant COLSS and CPCS.

For implementation of Framatome CE16HTP fuel the CETOP-D model with the CE-1 CHF correlation is maintained for use in the setpoint analysis. The CETOP-D model with the CE-1 CHF correlation is benchmarked to a VIPRE model using the BHTP CHF correlation to determine the CETOP-D adjustment factors. Since the adjustment factors from the VIPRE benchmarking are also used in setpoints analysis, the addressable constants that result from the setpoints analysis for installation into the Plant Computers will inherently correct for the COLSS/CPCS use of the CE-1 CHF. Therefore, the MSCU methodology as described in CEN-356(V)-P-A, Revision 01-P-A, *Modified Statistical Combination of Uncertainties,* (Reference 11.1) is applicable to setpoint analysis of the CE16HTP fuel.

An available alternative for implementation of Framatome CE16HTP fuel is to use the MSCU process steps as augmented by WCAP-16500-P-A, Supplement 1, Revision 1, *Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)*, (Reference 11.2). This alternative is the process used for implementation of CE16NGF. Instead of accounting for the difference between the BHTP and CE-1 CHFs in the T-H code benchmarking, the WCAP-16500-P-A (Reference 11.2) augmented process uses the setpoint analysis to account for the difference in CHF between the fuel and COLSS and CPCS.

11.1. References

11.1. CEN-356(V)-P-A, Revision 01-P-A, *Modified Statistical Combination of Uncertainties*, May 1988.

11.2. WCAP-16500-P-A, Supplement 1, Revision 1, *Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)*, December 2010.

12. OTHER ISSUES

12.1. Technical Specification Safety Limit Considerations

12.1.1. TS Safety Limit 2.2.1.1 – DNBR Safety Limit

10 CFR 50.36, Technical Specifications, defines a safety limit as a limit upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. 10 CFR 50.36 also states "A Limiting Safety System Setting is the setting for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

10 CFR Part 50 General Design Criterion (GDC) 10, Reactor Design, requires that specified fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur.

No changes are required of the Technical Specification Safety Limit 2.1.1.1 which addresses the Departure from Nucleate Boiling Ratio (DNBR) safety limit of 1.34. This DNBR safety limit is based on use of the CE-1 critical heat flux (CHF) correlation.

The BHTP CHF correlation may be used in the CE16HTP safety analyses. However, because of existing hardware limitations, the Core Protection Calculator (CPC) algorithm will retain the CE-1 CHF correlation. Since the CPC thermal-hydraulic algorithm retains the CE-1 CHF correlation, any change to the DNBR-Low trip setpoint and Allowable Value would introduce inconsistency between the trip setpoint and the Control Room monitors. To ensure that the Plant Operators have consistency between the trip setpoint and their Control Room monitors (i.e., a human factors concern), the DNBR-Low trip setpoint and Allowable Value will remain set at 1.34.

The TS Bases 2.1.1 will be revised to address the application of the current TS 2.1.1.1 DNBR safety limit of 1.34 for a reactor core loaded with Westinghouse supplied fuel (i.e., CE16STD and CE16NGF) and/or Framatome supplied CE16HTP fuel. The following is proposed text to address TS 2.1.1.1:

> The CPC algorithm uses the CE-1 correlation and the DNBR-Low trip setpoint with an Allowable Value of 1.34. The DNBR limit used in the safety analyses is dependent on fuel type.
>
> 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 may be limiting, the most conservative DNBR analytical limit is used.

12.1.2. TS Safety Limit 2.2.1.2 – Peak Fuel Centerline Temperature Safety Limit

The restrictions of Technical Specification Safety Limit 2.1.1.2 prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. The current TS 2.1.1.2 text is consistent with the Standard Technical Specifications for Combustion Engineering Plants as specified in NUREG-1432 (References 12.1 and 12.2).

Centerline fuel melt analyses for Framatome supplied fuel will be performed with COPERNIC best-estimate predictions and nominal fuel rod design parameters consistent with the best estimate fuel temperature relationship, including uncertainty, defined by Equation 12-3 in approved COPERNIC Topical Report BAW-10231(P)(A) (Reference 12.2).

The Bases for TS 2.1.1.2 will be revised to address the fuel centerline melt temperature for Framatome supplied fuel. For Framatome supplied fuel, the design melting point of new fuel is 4901 Fahrenheit. The melting point is adjusted downward from this temperature depending on the amount of burnup in the fuel. The 13.7 Fahrenheit per 10,000 MWD/MTU adjustment for burnup was accepted by the NRC in Topical Report BAW 10231(P)(A), *COPERNIC Fuel Rod Design Computer Code*, January 2004 (refer to Section 12.3.1 and Equation 12-3 on pages 12-6 and 12-7, and the response to RAI Question 24 on pages 14-61 and 14-62). The values in the fuel melt safety limit are presented in units of degrees Fahrenheit and MWD/MTU burnup, following conversion from Equation 12-3 units of degrees Centigrade and GWd/tU burnup.

Technical Specification Bases 2.1.1 will be revised to include text to address the application of the current peak fuel centerline temperature safety limit for Westinghouse supplied fuel (i.e., CE16STD and CE16NGF) and/or Framatome supplied CE16HTP fuel. The following is proposed text to address TS 2.1.1.2:

> Maintaining the dynamically adjusted peak LHR to ≤ 21 kW/ft or peak fuel centerline temperature below the limit ensures that fuel centerline melt will not occur during normal operating conditions or design AOOs.
>
> For Westinghouse supplied fuel, 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.

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-10231(P)(A), “COPERNIC Fuel Rod Design Computer Code,” January 2004.

12.2. Spent Fuel Criticality Safety Analysis

By letter dated July 28, 2017 (Reference 12.4), the NRC issued Amendment No. 203 to the PVNGS Units 1, 2 and 3 Operating Licenses. This license amendment revised the technical specifications to modify TS requirements to incorporate the results of an updated criticality safety analysis documented in Westinghouse Report WCAP-18030-P (Reference 12.4) for both new and spent fuel storage.

Section 3.1, "Reactor Description," and Section 4.3, "Fuel Design Selection," of WCAP-18030-P provide information on fuel assembly selection for use in the criticality safety analysis. WCAP-18030-P documents analysis of current, future, and all legacy fuel assembly designs used or expected to be used at PVNGS to establish the limiting fuel assembly design. These designs included CE Standard Fuel, CE Value Added Pellet, Framatome lead fuel assemblies that have operated at Palo Verde, and CE Next Generation Fuel. The analysis of the Framatome fuel assembly design included the modeling of gadolinia as a burnable absorber.

WCAP-18030-P documents a fuel assembly reactivity comparison and concluded that the NGF design, as described in Table 3-6, *Palo Verde Operation with IFBA and NGF*, in WCAP-18030-P, would be limiting throughout the life of the fuel assembly as compared to legacy and current fuel designs. The NGF design was then used for the rest of the analysis.

Per the Safety Evaluation for Amendment No. 203, the NRC staff accepted the selection of the NGF design, as described in Table 3-6, as the reference assembly design, and concluded that the assumptions and analytical techniques used were adequately substantiated to conclude at a 95 percent probability, 95 percent confidence level, that the regulatory requirements will be met.

In accordance with the APS response to SNPB RAI-4 associated with the criticality license amendment request (Reference 12.5), to provide confidence that future cycles are bounded by the fuel design and operating parameters assumed in WCAP-18030-P, APS is revising procedures to include verifications against the criteria described in Section 4.4, “Final Depletion Analysis” of WCAP-18030-P.

12.3. References

12.1. NUREG-1432, Revision 4.0, Volume 1, *Standard `Technical Specifications - Combustion Engineering Plants: Specifications*, (ADAMS Accession No. ML12102A165).

12.2. NUREG-1432, Revision 4.0, Volume 2, *Standard Technical Specifications - Combustion Engineering Plants: Bases*, (ADAMS Accession No. ML12102A169).

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

12.4. Letter from (NRC) to Robert S. Bement, *Palo Verde Nuclear Generating Station Units 1, 2, and 3 – Issuance of Amendments to Revise Technical Specifications to Incorporate Updated Criticality Safety Analysis (CAC Nos. MF7138, MF7139, and MF7140)*, July 28, 2017 (ADAMS Accession No. ML17188A412).

12.5. Westinghouse Report WCAP-18030-P, Revision 1, *Criticality Safety Analysis for Palo Verde Nuclear Generating Station Units 1, 2, and 3*, October 2016.

12.6. APS Letter 102-07342 from Maria Lacal (APS) to NRC of October 6, 2016, *Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specifications to Incorporate Updated Criticality Safety Analysis*, (ADAMS Accession Number No. ML16286A240).

13. LIST OF TYPICAL ACRONYMS, ABBREVIATIONS AND TRADEMARKS

Acronym	Meaning
ABB-NV	Westinghouse (ABB) Non-Vane Critical Heat Flux Correlation
ADAMS	NRC Agencywide Document Access Management System
ADV	Atmospheric Dump Valve
AFW	Auxiliary Feedwater
AOO	Anticipated Operational Occurrence
AOR	Analysis of Record
APS	Arizona Public Service
ARO	All Rods Out
ASI	Axial Shape Index
ASME	American Society of Mechanical Engineers
BAP	Boric Acid Precipitation
BHTP	Designation for a Framatome Critical Heat Flux Correlation
BOC	Beginning of Cycle
BOL	Beginning-of-Life
BWR	Boiling Water Reactor
B&W	Babcock and Wilcox
cal/gm	Calories per gram
CBC	Critical Boron Concentration
CE	Combustion Engineering, now Westinghouse
CE-1	Westinghouse (Combustion Engineering) Critical Heat Flux Correlation
CE-16	Combustion Engineering 16x16 (fuel design)
CE16NGF	Westinghouse supplied Next Generation Fuel
CE16HTP	Framatome supplied High Thermal Performance Fuel
CE16STD	Westinghouse supplied Standard fuel
CEA	Control Element Assembly (Control Rod)
CEAW	Control Element Assembly Withdrawal
CETOP	Combustion Engineering Thermal On-Line Program
CFM	Centerline Fuel Melt
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
COLR	Core Operating Limits Report
COLSS	Core Operating Limits Supervisory System
CPC(S)	Core Protection Calculator (System)
CVCS	Chemical Volume Control System
DBLLOCUS	Double-Ended Break of a Letdown Line Outside Containment

Acronym	Meaning
DF	Decontamination Factor
DFWT	Decrease in Feedwater Temperature
DHR	Decay Heat Removal
DLL	Dynamic Link Library
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DNBRL	Design Limit for Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Days
EM	Evaluation Model (or Methodology)
EOC	End of Cycle
EOL	End-of-Life
EPRI	Electric Power Research Institute
Er_2O_3	Erbia (burnable poison)
EQ	Equipment Qualification
F_q, F_Q	Power Distribution Total Peaking Factor
F_r	Power Distribution Integrated Peaking Factor
F_{xy}	Power Distribution Planar Peaking Factor
FCF	Framatome Cogema Fuels
FIV	Flow-Induced Vibration
ft	feet
FWLB	Feedwater Line Break (Feedwater System Pipe Breaks)
Gd_2O_3	Gadolinia (burnable poison)
GDC	General Design Criterion (or Criteria)
GIS	Generated Iodine Spike
GL	Generic Letter
GTRF	Grid-to-Rod Fretting
GWd/mtU	Gigawatt-Day(s) per Metric Ton Uranium
HFP	Hot Full Power
HL	Heated Length
HMP™	High Mechanical Performance
HTP™	High Thermal Performance
HZP	Hot Zero Power
ICI	In-Core Instrumentation
ID	Inadvertent Deboration
ID	Inner Diameter

Acronym	Meaning
IFBA	Integral Fuel Burnable Absorber (zirconium diboride)
IFD	Inlet Flow Distribution
IFM	Intermediate Flow Mixing
IFWF	Increase in Main Feedwater Flow
IMSF	Increased Main Steam Flow
IN	Information Notice
IOSGADV	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve
IOSGADVLOP	IOSGADV with a Loss of Offsite Power
kw/ft	Kilowatt per Foot
LAR	License Amendment Request
LBLOCA	Large Break LOCA
LCO	Limiting Condition of Operation
LFA	Lead Fuel Assembly
LFW	Loss of (Normal) Feedwater
LHGR	Linear Heat Generation Rate
LHR	Linear Heat Rate
LOAC	Loss of (Nonemergency) AC Power (to the Station Auxiliaries)
LOCA	Loss of Coolant Accident
LOCV	Loss of Condenser Vacuum
LOF	Loss of Flow (Total Loss of Reactor Coolant Flow)
LOL	Loss of (External) Load
LOP	Loss of (offsite or AC) Power
LPPT	Low Power Physics Testing
LPZ	Low Population Zone
LSC	Loop Seal Clearing
LTC	Long-Term Cooling
LTOP	Low Temperature Over Pressure
LTP	Lower Tie Plate
M5®	Designation for a Framatome Fuel Rod Cladding Material
M&E	Mass and Energy
MFIV	Main Feedwater Isolation Valve
Mlb, Mlbm	Mega (one million) pound mass
MSCU	Modified Statistical Combination of Uncertainties
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MTU	Metric Ton of Uranium
MWD/MTU, MWd/MTU	Megawatt-Day(s) per Metric Ton Uranium

Acronym	Meaning
MWt, MWth	Megawatt(s) Thermal
N.A.	Not Applicable
NAF	Neutron Absorbing Fuel
NEI	Nuclear Energy Institute
NFM	PVNGS Nuclear Fuel Management
NGF	Next Generation Fuel
NPSH	Net Positive Suction Head
NRC	United States Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OD	Outer Diameter
OE	Operating Experience
P/M	Predicted/Measured
PAT	Power Ascension Testing
PCM	Per Cent Mille (1 part in 100,000)
PCT	Peak Cladding Temperature
pdf or p.d.f. or pdf	fuel failure Probability Distribution Function
PDIL	Power Dependent Insertion Limit
PIE	Post-Irradiation Examination
PIS	Pre-accident Iodine Spike
PLCS	Pressurizer Level Control System
POL	Power Operating Limit
psia	Pounds per Square Inch at Atmosphere
PSV	Pressurizer Safety Valve
PVNGS	Palo Verde Nuclear Generating Station
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
QAPD	Quality Assurance Program Description
RAI, RAIs	Request(s) for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RLBLOCA	Realistic Large Break LOCA
RTP	Rated Thermal Power
SAFDL	Specified Acceptable Fuel Design Limit
SBLOCA	Small Break LOCA
SCU	Statistical Combination of Uncertainties

Acronym	Meaning
SDC	Shutdown Cooling
SE	US NRC Safety Evaluation
SER	US NRC Safety Evaluation Report
SF	Single Failure
SFP	Spent Fuel Pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SGTRLOP+SF	Steam Generator Tube Rupture With a Loss of Offsite Power and a Single Failure
SI	Safety Injection
SIRCP	Startup of an Inactive Reactor Coolant Pump
SIT	Safety Injection Tank
SL	Safety Limit
SLB	Steam Line Break
SR	Seized Rotor (Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power)
SR	Surveillance Requirement
SRP	Standard Review Plan
SS	Sheared Shaft (Single Reactor Coolant Pump Shaft Break with Loss of Offsite Power)
SSC	Systems, Structures and Components
STAR	Startup Test Activity Reduction
STD	Standard Fuel (also known as Value-Added Fuel)
TCD	Thermal Conductivity Degradation
TCS	Transient Cladding Strain
T-H	Thermal Hydraulic
TORC	Thermal-Hydraulics of Reactor Core
TS	Technical Specifications
TSC/TFR	Transportable Storage Canister/Transfer Cask
TSTF	Technical Specification Task Force
TT	Turbine Trip
UFSAR	Updated Final Safety Analysis Report
UGS	Upper Guide Structure
NRC	United States Nuclear Regulatory Commission
UO_2	Uranium Dioxide (fuel)
UTP	Upper Tie Plate
VIPRE	Versatile Internals and Component Program for Reactors; EPRI
WCAP	Westinghouse Commercial Atomic Power

Acronym	Meaning
WEC	Westinghouse Electric Company
WLOP	Westinghouse Low-Pressure Critical Heat Flux Correlation
WPR	Wetted Perimeter Ratio
WSSV	Westinghouse Side Supported Vane Critical Heat Flux Correlation
WSSV-T	Westinghouse Side Supported Vane Critical Heat Flux Correlation for use with the TORC code
wt%	Weight Percent

13.1. Trademark Notes

ATTACHMENT 9

Framatome Licensing Summary Reports

ANP-3639NP Revision 0 “Palo Verde Units 1, 2, and 3 Realistic Large Break LOCA Summary Report”

[NON-PROPRIETARY VERSION]

AND

ANP-3640NP Revision 0 “Palo Verde Units 1, 2, and 3 Small Break LOCA Summary Report”

[NON-PROPRIETARY VERSION]

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

ANP-3639NP
Revision 0

Licensing Report

March 2018

Framatome Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
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Nomenclature

Acronym	**Definition**
ASI	Axial Shape Index
BOCR	Beginning of Core Recovery
CCFL	Counter Current Flow Limiting
CE	Combustion Engineering
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CSAU	Code Scaling, Applicability and Uncertainty
CWO	Core-Wide Oxidation
ECCS	Emergency Core Cooling System
ECR	Equivalent Cladding Reacted
EM	Evaluation Model
EMDAP	Evaluation Model Development and Assessment Process
F_Q	Total Peaking Factor
F_r	Nuclear Enthalpy Rise Factor
FSRR	Fuel Swell Rupture and Relocation
GDC	General Design Criteria
HPSI	High Pressure Safety Injection
LHGR	Linear Heat Generation Rate
LPSI	Low Pressure Safety Injection
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MLO	Maximum Local Oxidation
No-LOOP	No Loss of Offsite Power
NRC	U. S. Nuclear Regulatory Commission
PCT	Peak Clad Temperature
PIRT	Phenomena Identification and Ranking Table
PWR	Pressurized Water Reactor

Acronym	Definition
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RLBLOCA	Realistic Large Break Loss of Coolant Accident
SE	Safety Evaluation
SG	Steam Generator
SIAS	Safety Injection Actuation Signal
SIT	Safety Injection Tank
UTL	Upper Tolerance Limit
w/o	Weight Percent

ABSTRACT

This report describes and provides results from the RLBLOCA analysis for the Palo Verde Nuclear Generating Station (PVNGS) Vendor Qualification Program with the Framatome Advanced CE16 HTP™[1] Fuel Design with M5®[1] cladding. The plant is a PWR Combustion Engineering 2x4-loop design with an analyzed thermal power of 4070 MWt (including measurement uncertainty) and dry atmospheric containment. The loops contain four RCPs, two U-tube steam generators and a pressurizer.

The analysis supports operation of PVNGS Units 1 through 3 with Framatome's 16x16 CE array with HTP™ intermediate grids and a lower HMP™[1] grid. The fuel assembly includes an M5® MONOBLOC™[1] guide tube design, M5® fuel rod design using standard UO_2 fuel with 2%, 4%, 6%, and 8% Gd_2O_3 and FUELGUARD™[1] debris-resistant lower tie-plate design. The analysis performed is the PVNGS-specific implementation of Framatome's RLBLOCA EM in Reference 1. The analysis results confirm that the 10 CFR 50.46(b) paragraph (1) through (3) acceptance criteria (Reference 2) are met and serve as the basis for operation of PVNGS Units 1 through 3 with Framatome Advanced CE16 HTP™ Fuel with M5® cladding.

[1] M5, HTP, HMP, MONOBLOC and FUELGUARD are trademarks or registered trademarks of Framatome or its affiliates, in the USA or other countries.

1.0 INTRODUCTION

This report summarizes the RLBLOCA analysis for Palo Verde Nuclear Generating Station (PVNGS). The purpose of the RLBLOCA analysis is to support the Vendor Qualification Program (VQP) for PVNGS with the Framatome Advanced CE 16 x 16 HTP™ Fuel Design with M5® cladding. This analysis was performed in accordance with the NRC-approved S-RELAP5 methodology described in Reference 1.

PVNGS Units 1, 2, and 3 are 2x4-loop, CE-designed PWRs. The Framatome Advanced CE16 Fuel Design with M5® cladding for PVNGS consists of a 16x16 CE array with HTP™ intermediate grids and a lower HMP™ grid. The fuel assembly will include an M5® MONOBLOC™ guide tube design, M5® fuel rod design and FUELGUARD™ debris-resistant lower tie-plate design. The fuel will be standard UO_2 fuel with 2, 4, 6, and 8 weight percent Gd_2O_3 rods included.

The analysis supports plant operation at a core power level of 4070 MWt (including measurement uncertainty), a peak LHGR of 13.1 kW/ft[1], a radial peaking factor of 1.81 (includes uncertainty), and up to 10% SG tube plugging. This analysis also addresses typical operational ranges or technical specification limits (whichever is applicable) with regard to pressurizer pressure and level; safety injection tank (SIT) pressure, temperature, and level; core inlet temperature; core flow; containment pressure and temperature; and refueling water storage tank temperature. The analysis explicitly analyzes fresh and once-burned fuel assemblies. The parameter specification for this analysis is provided in Table 3-1. The analysis also uses the Fuel Swelling, Rupture, and Relocation (FSRR) model to determine if cladding rupture occurs and evaluate the consequences of FSRR on the transient response.

[1] For some cases, the LHGR exceeded the Palo Verde limit. These cases do not invalidate the 95/95 resulting answer of the analysis and the higher LHGR limit (13.3 kW/ft) utilized in this analysis is conservative.

The UTL results providing 95/95 simultaneous coverage from this evaluation meet the 10 CFR 50.46(b) criteria with a PCT of 1752°F, a maximum local oxidation of 2.37 percent and a total core-wide oxidation of 0.020 percent. The PCT of 1752°F occurred in a fresh UO_2 fuel rod with an assembly burnup of 21.9 GWd/mtU.

2.0 DESCRIPTION OF ANALYSIS

2.1 *Acceptance Criteria*

The purpose of the analysis is to verify the adequacy of the PVNGS ECCS by demonstrating compliance with the following 10 CFR 50.46(b) criteria (Reference 2):

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

The final two criteria, coolable geometry and long-term cooling, are treated separately during plant-specific evaluations.

Note: The original 17% value in the second acceptance criterion for MLO was based on the usage of the Baker-Just correlation. For present reviews on ECCS Evaluation Model (EM) applications, the NRC staff is imposing a limitation specifying that the equivalent cladding reacted (ECR) results calculated using the Cathcart-Pawel correlation are considered acceptable in conformance with 10 CFR 50.46(b)(2) if the ECR value is less than 13%. The limitation is addressed in Table 3-3.

2.2 *Description of LBLOCA Event*

A Large Break LOCA is initiated by a postulated large rupture of the RCS primary piping. Because a pump discharge break more easily discharges all coolant, particularly liquid coolant (including the pressurizer inventory), to the containment and is most likely to discharge the emergency coolant to the break, the worst break location is in the cold leg piping between the reactor coolant pump and the reactor vessel for the RCS loop containing the pressurizer. The plant is assumed to be operating normally at full power prior to the accident. The large cold leg break is assumed to open instantaneously. For this break, a rapid depressurization occurs, along with a core flow stagnation and reversal. This causes the fuel rods to experience departure from nucleate boiling (DNB). Subsequently, the limiting fuel rods are cooled by film and transition boiling heat transfer. The coolant voiding creates a strong negative reactivity effect, and core fission ends. As heat transfer from the rods is reduced, the cladding temperature rises. A reactor trip signal is initiated when the low pressurizer pressure trip setpoint is reached; however, for Framatome RLBLOCA analyses reactor trip is conservatively neglected. The reactor is shut down by coolant voiding in the core.

Coolant in all regions of the RCS begins to flash. At the break plane, the loss of subcooling in the coolant results in substantially reduced break flow. This reduces the depressurization rate, and may also lead to a period of positive core flow or reduced downflow as the reactor coolant pumps in the intact loops continue to supply water to the vessel. Cladding temperatures may be reduced, and some portions of the core may rewet during this period. This positive core flow or reduced downflow period ends as two-phase conditions occur in the reactor coolant pumps, reducing their effectiveness. Once again, the core flow reverses as most of the vessel mass flows out through the broken cold leg.

Mitigation of the LBLOCA begins when the SIAS is tripped. This signal is initiated by either high containment pressure or low pressurizer pressure. Regulations require that a worst single failure be considered for ECCS safety analysis. This single failure has been determined for a CE plant to be the loss of one ECCS train, including one HPSI pump and one LPSI pump. The Framatome RLBLOCA methodology conservatively assumes an on-time start and normal lineups of the containment spray, fan coolers (if present), or other cooling mechanisms. (Note: reducing containment pressure will penalize clad temperatures by increasing RCS voiding and break flow).

When the RCS pressure falls below the SIT pressure, fluid from the SITs is injected into the cold legs. In the early delivery of SIT water, high pressure and high break flow will drive some of this fluid to bypass the core. During this bypass period, core heat transfer remains poor and fuel rod cladding temperatures increase. As RCS and containment pressures equilibrate, ECCS water begins to fill the lower plenum and eventually the lower portions of the core; thus, core heat transfer improves and cladding temperatures decrease.

Eventually, the relatively large volume of SIT water is exhausted and core recovery must rely on pumped SI coolant delivery alone. As the SITs empty, the nitrogen gas used to pressurize the SITs exits through the break. This gas release may result in a short period of improved core heat transfer as the nitrogen gas displaces water in the downcomer. After the nitrogen gas has been expelled, the ECCS temporarily may not be able to sustain full core cooling because of the core decay heat and the higher steam temperatures created by quenching in the lower portions of the core. Fuel rod cladding temperatures may increase for a short period until more energy is removed from the core by the low pressure safety injection and the decay heat continues to fall. Steam generated from fuel rod rewet will entrain liquid and pass through the core, vessel upper plenum, the hot legs, the steam generator, and the reactor coolant pump before it is vented out the break. The resistance of this flow path to the steam flow is balanced by the driving force of water filling the downcomer. This resistance may act to retard the progression of the core reflood and postpone core wide cooling. Eventually (within a few minutes of the accident), the core reflood will progress sufficiently to ensure core wide cooling. Full core quench occurs within a few minutes after core wide cooling. Long term cooling is then sustained with the low pressure safety injection.

2.3 *Description of Analytical Models*

The NRC-approved RLBLOCA methodology is documented in EMF-2103(P)(A) *Realistic Large Break LOCA Methodology for Pressurized Water Reactors* (Reference 1). The methodology follows the Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology (Reference 3) and the requirements of the Evaluation Model Development and Assessment Process (EMDAP) documented in Reference 4. The CSAU method outlines an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifies the uncertainties in a LOCA analysis.

The Framatome S-RELAP5 RLBLOCA methodology evaluation model for event response of the primary and secondary systems and the hot fuel rod used in this analysis is based on the use of two computer codes.

- COPERNIC for computation of the initial fuel stored energy, fission gas release, and the transient fuel-cladding gap conductance.
- S-RELAP5 for the thermal-hydraulic system calculations (includes ICECON for containment response).

The methodology (Reference 1) has been reviewed and approved by the NRC to perform LBLOCA analyses. However, some differences from the current LBLOCA methodology were included in this analysis, as described below.

The governing two-fluid (plus non-condensable) model with conservation equations for mass, energy, and momentum transfer is used. The reactor core is modeled in S-RELAP5 with heat generation rates determined from reactor kinetics equations (point kinetics) with reactivity feedback, and with actinide and decay heat.

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on the other are accounted for by interfacial friction, and heat and mass transfer interaction terms in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ.

The modeling of plant components is performed by following guidelines developed to ensure accurate accounting for physical dimensions and that the dominant phenomena expected during the LBLOCA event are captured. The basic building blocks for modeling are hydraulic volumes for fluid paths and heat structures for heat transfer. In addition, special purpose components exist to represent specific components such as the Reactor Coolant Pumps (RCPs) or the steam generator (SG) separators. All geometries are modeled at the resolution necessary to best resolve the flow field and the phenomena being modeled within practical computational limitations.

The analysis considers blockage effects due to clad swelling and rupture as well as increased heat load due to fuel relocation in the ballooned region of the cladding in the prediction of the hot fuel rod PCT.

A typical calculation using S-RELAP5 begins with the establishment of a steady-state initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are chosen to reflect plant technical specifications or to match measured data. Additionally, the COPERNIC code provides initial conditions for the S-RELAP5 fuel models. Specific parameters are discussed in Section 2.6.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood is computed continuously using S-RELAP5. Containment pressure is calculated by the ICECON module within S-RELAP5.

A detailed assessment of the S-RELAP5 computer code was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the ability of the code to predict key physical phenomena in a PWR LBLOCA. The final step of the best-estimate methodology is to combine all the uncertainties related to the code and plant parameters and estimate the first three criteria of 10 CFR 50.46 with a probability of at least 95 percent with 95 percent confidence. The steps taken to derive the uncertainty estimate are summarized below:

1. Base Plant Input File Development

   First, base COPERNIC and S-RELAP5 input files for the plant (including the containment input file) are developed. The code input development guidelines documented in Appendix A of Reference 1 are applied to ensure that model nodalization is consistent with the model nodalization used in the code validation.

2. Sampled Case Development

   The statistical approach requires that many “sampled” cases be created and processed. For every set of input created, each “key LOCA parameter” is randomly sampled over a range established through code uncertainty assessment or expected operating limits (provided by plant technical specifications or data). Those parameters considered "key LOCA parameters" are listed in Table A-6 of Reference 1. This list includes both parameters related to LOCA phenomena, based on the PIRT provided in Reference 1, and to plant operating parameters. The uncertainty ranges associated with each of the model parameters are provided in Table A-7 of Reference 1.

3. Determination of Adequacy of ECCS

   The RLBLOCA methodology uses a non-parametric statistical approach to determine that the first three criteria of 10 CFR 50.46 are met with a probability higher than 95 percent with 95 percent confidence.

2.4 *GDC-35 Limiting Condition Determination*

GDC-35 requires that a system be designed to provide abundant core cooling with suitable redundancy such that the capability is maintained in either the LOOP or No-LOOP conditions. [

]

2.5 *Overall Statistical Compliance to Criteria*

[

]

2.6 *Plant Description*

The plant analyzed is the Palo Verde Unit (Units 1 through 3 are supported), CE designed PWR, which has 2x4-loop arrangement. All three units at PVNGS are CE-designed PWRs with two hot legs, four cold legs and RCPs, and two vertical U-tube SGs. The RCS includes one pressurizer connected to a hot leg. The ECCS comprises four SITs (one per loop/cold leg), and one full train of LPSI and HPSI (after applying the single failure assumption). One HPSI pump is able to feed all four cold leg injection points (cross connected). The highest HPSI flow is modeled going to the broken loop. One LPSI pump is able to feed two cold leg injection points (in the analysis, cold leg 1A which contains the break and the adjacent cold leg 1B receive LPSI flow). The

RLBLOCA transients are of sufficiently short duration that the switchover to sump cooling water for ECCS pumped injection does not need to be considered.

The S-RELAP5 model explicitly describes the RCS, reactor vessel, pressurizer, and ECCS. The ECCS includes a SIT path and a LPSI/HPSI (LPSI feeds only 2 RCS loops as described above) path per RCS loop. HPSI and LPSI feed into a common header that connects to each cold leg pipe downstream of the RCP discharge. The ECCS pumped injection is modeled as a table of flow versus backpressure. This model also describes the secondary-side steam generator that is instantaneously isolated (closed main steam isolation valve and feedwater trip) at the time of the break. The analysis includes Framatome fuel with M5® cladding and utilizes the COPERNIC code for fuel calculations within S-RELAP5. The primary and secondary coolant systems for PVNGS were nodalized consistent with code input guidelines in Appendix A of Reference 1.

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

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As described in Appendix A of Reference 1, many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges are sampled. A summary of those parameters sampled is given in Table A-6 of Reference 1. The LBLOCA phenomenological uncertainties are provided in Table A-7 of Reference 1. Values for process or operational parameters, including ranges of sampled process parameters, and fuel design parameters used in this analysis are given in Table 3-1. Table 3-2 presents a summary of the uncertainties used in the analysis. Two parameters (refueling water storage tank temperature and diesel start time) are set at conservative bounding values for all calculations. A containment passive heat sink margin was incorporated into the analysis so that future changes to the plant configuration could be controlled through established analytical margins rather than through ΔPCT assessments.

2.7 *SE Limitations*

The RLBLOCA analysis for PVNGS presented herein is consistent with the submitted RLBLOCA methodology documented in EMF-2103(P)(A), Revision 3 (Reference 1). The limitation and conditions from the NRC SE (Reference 1) are addressed in Table 3-3.

3.0 RLBLOCA ANALYSIS

3.1 *RLBLOCA Results*

[

] For a simultaneous coverage/confidence level of 95/95, the UTL values are a PCT of 1752°F, a MLO of 2.37 percent, and a CWO of 0.020 percent. The fraction of total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total core wide percent oxidation, which is well below the 1 percent limit.

Table 3-5 is a summary of the major input parameters for the demonstration case. The sequence of event times for the demonstration case is provided in Table 3-6. The heat transfer parameter ranges for the demonstration case are provided in Table 3-7. [

]

The analysis scatter plots for the case set are shown in Figure 3-1 through Figure 3-5. Figure 3-1 shows linear scatter plots of the key parameters sampled for all cases. Parameter labels appear to the left of each individual plot. These figures illustrate the parameter ranges used in the analysis. Visual examination of the linear scatter plots demonstrates that the spread and coverage of all of the values used is appropriate and within the uncertainty ranges listed in Table 3-2. Appendix A provides a listing of all the sampled input values for each case. Key results such as the PCT and event timings are also listed for the case set.

Figure 3-2 and Figure 3-3 show PCT scatter plots versus the time of PCT and versus break size, respectively. The scatter plots for the maximum local oxidation and total core-wide oxidation are shown in Figure 3-4 and Figure 3-5, respectively.

Figure 3-2 shows a general decreasing trend of PCT with increasing PCT time with a distinctive cluster of high PCTs (>1300°F) occurring during blowdown. The time of PCT for this cluster is less than 15 seconds. Blowdown PCT cases are dominated by rapid RCS depressurization and stored energy content. Early reflood PCT cases are dominated by decay heat removal capacity, which is highly dependent on SIT liquid volume and pressure setpoint. As shown in Figure 3-3, there is a strong correlation of PCT to break size. From all sampled parameters, the break size is a dominant effect on PCT because of its high influence on the rate of primary depressurization. As such, the high PCT clusters correlate with the larger end of the break sizes.

In general, for this plant design, larger breaks result in a PCT predominantly occurring in the early blowdown phase. There are some notable exceptions to the PCT occurring at early blowdown. There are several cases including the limiting case, that have relatively high PCTs that occur during reflood. These cases have larger break sizes which allows for a quicker RCS depressurization and an earlier SIT actuation. During the reflood phase after the SITs have discharged, the impact of steam binding plays a significant role in the PCT. As the mixture level moves up the core, steam is generated and liquid is entrained. This entrained liquid passes into the steam generators and vaporizes, causing steam binding to occur. As the entrained liquid evaporates in the steam generator tubes the RCS pressure drop increases, creating a resistance that acts to retard the progression of the core reflood. Most of these cases have relatively high peaking which contributes to higher steam production and steam binding. These cases have top skewed axial power profiles which contribute to PCTs occurring at higher elevations in the core. The higher cladding temperatures at the higher elevations take longer to quench. Steam binding reduces the effectiveness of the pumped safety injection by increasing the necessary hydraulic head that is required in the downcomer to allow fluid to penetrate the core. Inspection of these cases shows that the upper plenum pressure is higher than the downcomer pressure for significant periods of the reflood phase which reduces the effectiveness of the pumped safety injection and contributes to the PCT occurring during the reflood phase.

The demonstration case is a reflood peak case with a PCT timing of 260.6 seconds. Figure 3-6 through Figure 3-17 show key parameters from the S-RELAP5 calculations for the demonstration case. The transient progression for the demonstration case follows that described in Section 2.2.

Figure 3-4 shows a general increasing trend of MLO with PCT. Since the MLO includes the pre-transient oxidation, the MLO is not only a function of cladding temperature but of time in cycle (burnup), which explains the scatter of the points. A stronger correlation of the CWO to PCT is demonstrated in Figure 3-5 as higher PCT cases would have a higher oxidation throughout the core.

Figure 3-18 compares the Beginning of Core Recovery (BOCR) times calculated by S-RELAP5 to the BOCR times predicted using the Counter Current Flow Limiting (CCFL) correlation developed by MPR Associates. Note that Figure 3-18 uses the total break area, while previous plots use break area per side.

[ ]

3.2 *Conclusions*

This report describes and provides results from the RLBLOCA analysis for the PVNGS VQP. The plant is a PWR Combustion Engineering 2x4-loop design with an analyzed thermal power of 4070 MWt (including measurement uncertainty) and dry atmospheric containment. The loops contain four RCPs, two U-tube steam generators and a pressurizer. The base model and the design inputs used are representative of the PVNGS Units 1 through 3. The application of the Framatome RLBLOCA methodology involves developing input decks, executing the simulations that comprise the uncertainty analysis, retrieving PCT, MLO, and CWO information and determining the simultaneous UTL results for the criteria. [ ] The UTL results providing a 95/95 simultaneous coverage/confidence level from this evaluation meet the 10 CFR 50.46(b) criteria with a PCT of 1752°F, a MLO of 2.37 percent and a CWO of 0.020 percent.

Table 3-1
RLBLOCA Analysis - Plant Parameter Values and Ranges

	Plant Parameter	Parameter Value
1.0	**Plant Physical Description**	
	1.1 Fuel	
	a) Cladding outside diameter	0.382 in.
	b) Cladding inside diameter	0.332 in.
	c) Cladding thickness	0.025 in.
	d) Pellet outside diameter	0.3255 in.
	e) Initial Pellet density	96 percent of theoretical
	f) Active fuel length	150 in.
	g) Gd_2O_3 concentrations	2, 4, 6, 8 w/o
	1.2 RCS	
	a) Flow resistance	Analysis
	b) Pressurizer location	[ ]
	c) Hot assembly location	Anywhere in core
	d) Hot assembly type	16x16
	e) SG tube plugging	10 percent
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Analyzed reactor power	4070 MWt
	b) F_Q	2.33[1,2]
	c) F_r	1.81[2]
	2.2 Fluid Conditions	
	a) Loop flow	155.8 Mlbm/hr $\leq$ M $\leq$ 190.3 Mlbm/hr
	b) RCS cold leg temperature	548°F $\leq$ T $\leq$ 556°F
	c) Upper head temperature	~Thot Temperature[3]
	d) Pressurizer pressure	2100 psia $\leq$ P $\leq$ 2325 psia
	e) Pressurizer level	24 percent $\leq$ L $\leq$ 59 percent
	f) SIT pressure	602 psia $\leq$ P $\leq$ 652 psia
	g) SIT liquid volume	1750 $ft^3 \leq V \leq$ 1950 ft^3
	h) SIT temperature	50°F $\leq$ T $\leq$ 120°F (coupled with containment temperature)
	i) SIT resistance fL/D	As-built piping configuration
	j) SIT boron	2300 ppm

1 The value used for F_Q is derived from the LHGR Technical Specification value

2 Includes measurement uncertainty.

3 Upper head temperature will change based on sampling of RCS temperature.

Table 3-1
RLBLOCA Analysis - Plant Parameter Values and Ranges
(Continued)

	Plant Parameter	Parameter Value
3.0	**Accident Boundary Conditions**	
	a) Break location	Cold leg pump discharge
	b) Break type	Double-ended guillotine or split
	c) Break size (each side, relative to cold leg pipe area)	[ ]
	d) ECCS pumped injection temperature	130°F
	e) HPSI pump delay	30 s (No-LOOP) 30 s (LOOP)
	f) LPSI pump delay	30 s (No-LOOP) 30 s (LOOP)
	g) Initial containment pressure	14.2 psia
	h) Initial containment temperature	50°F ≤ T ≤ 120°F
	i) Containment sprays delay	30 s
	j) Containment spray water temperature	50°F
	k) LPSI Flow	

RCS Cold Leg Pressure (psia)	Broken Loop Flow 1A (gpm)	Intact Loop Flow 1B (gpm)	Intact Loop Flow 2A (gpm)	Intact Loop Flow 2B (gpm)
14.2	1872	1872	0	0
25.2	1801	1801	0	0
54.2	1603	1603	0	0
64.2	1510	1510	0	0
77.2	1403	1403	0	0
96.2	1215	1215	0	0
114.2	1012.5	1012.5	0	0
117.2	1000	1000	0	0
126.2	750	750	0	0
138.2	500	500	0	0
144.2	333.5	333.5	0	0
147.2	250	250	0	0
156.2	0	0	0	0

Table 3-1
RLBLOCA Analysis - Plant Parameter Values and Ranges
(Continued)

Plant Parameter	Parameter Value

l) HPSI Flow

RCS Cold Leg Pressure (psia)	Broken Loop Flow 1A (gpm)	Intact Loop Flow 1B (gpm)	Intact Loop Flow 2A (gpm)	Intact Loop Flow 2B (gpm)
14.2	256.5	231.2	231.2	231.2
64.2	252.7	227.8	227.8	227.8
114.2	248.7	224.1	224.1	224.1
144.2	246.2	221.9	221.9	221.9
214.2	240.6	216.8	216.8	216.8
324.2	231.1	208.3	208.3	208.3
619.2	204.1	184.0	184.0	184.0
796.2	185.8	167.4	167.4	167.4
1007.2	161.2	145.3	145.3	145.3
1213.2	133.4	120.2	120.2	120.2
1363.2	109.4	98.6	98.6	98.6
1497.2	82.9	74.7	74.7	74.7
1595.2	55.4	49.9	49.9	49.9
1714.2	0.5	0.5	0.5	0.5
1715.0	0.0	0.0	0.0	0.0

Table 3-2
Statistical Distribution Used for Process Parameters

Parameter	Operational Uncertainty Distribution	Parameter Range	Measurement Uncertainty Distribution	Standard Deviation
Pressurizer Pressure (psia)	[ ]	2100 - 2325	[ ]	[ ]
Pressurizer Level (%)	[ ]	24 - 59	[ ]	[ ]
SIT Volume (ft^3)	[ ]	1750 - 1950	[ ]	[ ]
SIT Pressure (psia)	[ ]	602 - 652	[ ]	[ ]
Containment/SIT Temperature (°F)	[ ]	50 - 120	[ ]	[ ]
Containment Volume ($x10^6$ ft^3)	[ ]	2.62 – 3.01	[ ]	[ ]
Initial Flow Rate (Mlbm/hr)	[ ]	155.8 – 190.3	[ ]	[ ]
Initial Operating Temperature (°F)	[ ]	548 - 566	[ ]	[ ]

Table 3-3
SE Limitations Evaluation

Limitations (Sub-sections of Section 4.0 in Ref. 1)		Response
1	This EM was specifically reviewed in accordance with statements in EMF-2103, Revision 3. The NRC staff determined that the EM is acceptable for determining whether plant-specific results comply with the acceptance criteria set forth in 10 CFR 50.46(b), paragraphs (1) through (3). AREVA did not request, and the NRC staff did not consider, whether this EM would be considered applicable if used to determine whether the requirements of 10 CFR 50.46(b)(4), regarding coolable geometry, or (b)(5), regarding long-term core cooling, are satisfied. Thus, this approval does not apply to the use of SRELAP5-based methods of evaluating the effects of grid deformation due to seismic of LOCA blowdown loads, or for evaluating the effects of reactor coolant system boric acid transport. Such evaluations would be considered separate methods.	This analysis applies only to the acceptance criteria set forth in 10 CFR 50.46(b), paragraphs (1) through (3).
2	EMF-2103, Revision 3, approval is limited to application for 3-loop and 4-loop Westinghouse-designed nuclear steam supply systems (NSSSs), and to Combustion Engineering-designed NSSSs with cold leg ECCS injection, only. The NRC staff did not consider model applicability to other NSSS designs in its review.	Palo Verde is a CE-designed NSSS with cold leg ECCS injection.
3	The EM is approved based on models that are specific to AREVA proprietary M5® fuel cladding. The application of the model to other cladding types has not been reviewed.	The analysis supports operation with M5® cladding.

	Limitations (Sub-sections of Section 4.0 in Ref. 1)	Response
4	Plant-specific applications will generally be considered acceptable if they follow the modeling guidelines contained in Appendix A to EMF 2103, Revision 3. Plant-specific licensing actions referencing EMF 2103, Revision 3, analyses should include a statement summarizing the extent to which the guidelines were followed, and justification for any departures.	Except where described below, the modeling guidelines contained in Appendix A of EMF-2103(P)(A), Revision 3 (Reference 1) were followed completely for the analysis described in this notebook. [ ]

Limitations (Sub-sections of Section 4.0 in Ref. 1)		Response
		[ ]
5	The response to RAI 15 indicates that the fuel pellet relocation packing factor is derived from data that extend to currently licensed fuel burnup limits (i.e., rod average burnup of [ ]). Thus, the approval of this method is limited to fuel burnup below this value. Extension beyond rod average burnup of [ ] would require a revision or supplement to EMF-2103, Revision 3, or plant-specific justification.	The analysis supports operation with M5® cladding, which has a licensed limit of [ ]
6	The response to RAI 15 indicates that the fuel pellet relocation packing factor is derived from currently available data. Should new data become available to suggest that fuel pellet fragmentation behavior is other than that suggested by the currently available database, the NRC may request AREVA to update its model to reflect such new data.	The analysis uses the approved EMF-2103(P)(A), Revision 3 (Reference 1) relocation packing factor application. [ ]

Limitations (Sub-sections of Section 4.0 in Ref. 1)		Response
7	The regulatory limit contained in 10 CFR 50.46(b)(2), requiring cladding oxidation not to exceed 17 percent of the initial cladding thickness prior to oxidation, is based on the use of the Baker-Just oxidation correlation. To account for the use of the Cathcart-Pawel correlation, this limit shall be reduced to 13 percent, inclusive of pre-transient oxide layer thickness.	The MLO UTL is less than 13% (Table 3-4). [ ]
8	In conjunction with Limitation 8 above, Cathcart-Pawel oxidation results will be considered acceptable, provided plant-specific [ ] If second-cycle fuel is identified in a plant-specific analysis, whose [ ] the NRC staff reviewing the plant-specific analysis may request technical justification or quantitative assessment, demonstrating that [ ]	All second cycle fuel rod [ ]
9	The response to RAI 13 states that all operating ranges used in a plant-specific analysis are supplied for review by the NRC in a table like Table B-8 of EMF-2103, Revision 3. In plant-specific reviews, the uncertainty treatment for plant parameters will be considered acceptable if plant parameters are [ ] as appropriate . Alternative approaches may be used, provided they are supported with appropriate justification.	[ ]

	Limitations (Sub-sections of Section 4.0 in Ref. 1)	Response
10	[ ]	[ ] were not used in this analysis.
11	Any plant submittal to the NRC using EMF-2103, Revision 3, which is not based on the first statistical calculation intended to be the analysis of record must state that a re-analysis has been performed and must identify the changes that were made to the evaluation model and/or input in order to obtain the results in the submitted analysis.	This is the first statistical calculation for this plant application.

Table 3-4
Compliance with 10 CFR 50.46(b)

UTL for 95/95 Simultaneous Coverage/Confidence		
Parameter	**Value**	**Case Number**
PCT, °F	1752	131
MLO, %	2.37	216
CWO, %	0.020	191
Characteristics of Case Setting the PCT UTL		
PCT, °F		1752
PCT Rod Type		Fresh UO_2 Rod
Time of PCT, s		260.6
Elevation within Core, ft		11.11
Local Maximum Oxidation, %		7.68
Total Core-Wide Oxidation, %		0.032
PCT Rod Rupture Time, s		111.15
Rod Rupture Elevation within Core, ft		11.11

Table 3-5
Summary of Major Parameters for the Demonstration Case

Parameter	Value
Core Power (MWt)	4070
Time in Cycle (hrs)	23222
Limiting Rod Assembly Burnup (GWd/mtU)	21.9
Limiting Rod LHGR (kW/ft)	12.80
Limiting Rod Equivalent F_Q	2.23
Limiting Rod Radial Peak, F_r	1.81
Limiting Rod Axial Shape Index	-0.1028
Break Type	Split
Break Size (ft^2/side)	4.1981
[ ]	[ ]

Table 3-6
Calculated Event Times for the Demonstration Case

Event	Time (sec)
Break Opens	0.0
RCP Trip	0.0
SIAS Issued	0.6
Start of Broken Loop SIT Injection	12.2
Start of Intact Loop SIT Injection (Loop 2,3 and 4 respectively)	13.9, 13.9 and 13.9
Beginning of Core Recovery (Beginning of Reflood)	23.6
HPSI Available	30.6
Broken Loop HPSI Delivery Began	30.6
Intact Loops HPSI Delivery Began	30.6, 30.6 and 30.6
LPSI Available	30.6
Broken Loop LPSI Delivery Began	30.6
Intact Loops LPSI Delivery Began	30.6, N/A and N/A
Intact Loop SIT Emptied (Loop 2, 3 and 4 respectively)	53.6, 53.4 and 53.4
Broken Loop SIT Emptied	53.4
PCT Occurred	260.6
Transient Calculation Terminated	1067.9

Table 3-7
Heat Transfer Parameters for the Demonstration Case

Time (s)	[					]
LOCA Phase	Early Blowdown	Blowdown[1]	Refill	Reflood	Quench	Long Term Cooling[2]
Heat Transfer Mode	[					
Heat Transfer Correlations						
Maximum LHGR kW/ft						
Pressure (psia)						
Core Inlet Mass Flux (lb/s-ft^2)						
Vapor[4] Reynolds Number						
Liquid Reynolds Number						
Vapor Prandtl Number						
Liquid Prandtl Number						
Vapor[5] Superheat (°F)						]

[1] End of blowdown considered as beginning of refill.

[2] Quench to End of Transient.

[ ]

[4] Not important in pre-CHF heat transfer.

[5] Vapor superheat is meaningless during blowdown and system depressurization.

Table 3-8
Fuel Rod Rupture Ranges of Parameters

Figure 3-1
Scatter Plot Key Parameters

Figure 3-1
Scatter Plot Key Parameters (continued)

[

]

Figure 3-2
PCT versus PCT Time Scatter Plot

[ ]

Figure 3-3
PCT versus Break Size Scatter Plot

[

]

Figure 3-4
Maximum Local Oxidation versus PCT Scatter Plot

[

]

Figure 3-5
Total Core Wide Oxidation versus PCT Scatter Plot

Figure 3-6
Peak Cladding Temperature (Independent of Elevation) for the Demonstration Case

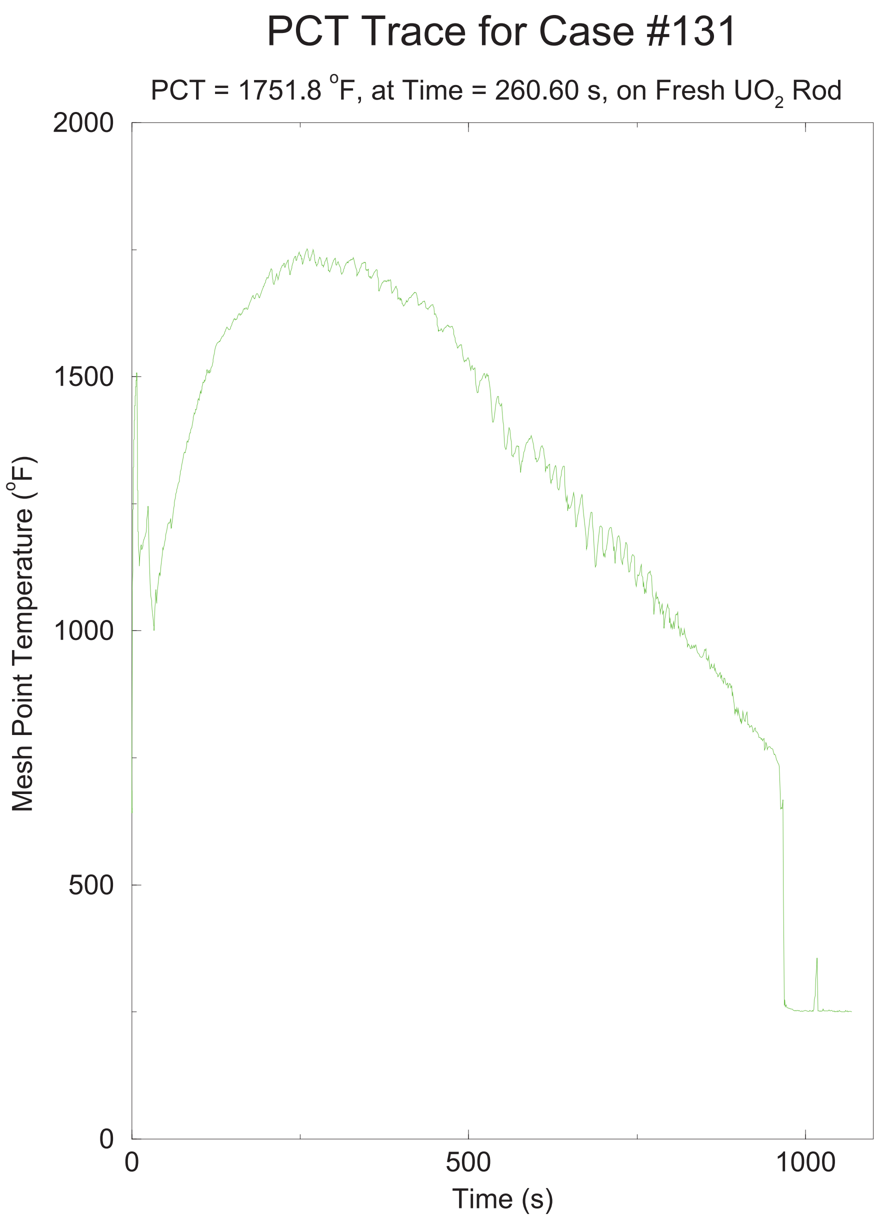

Figure 3-7
Break Flow for the Demonstration Case

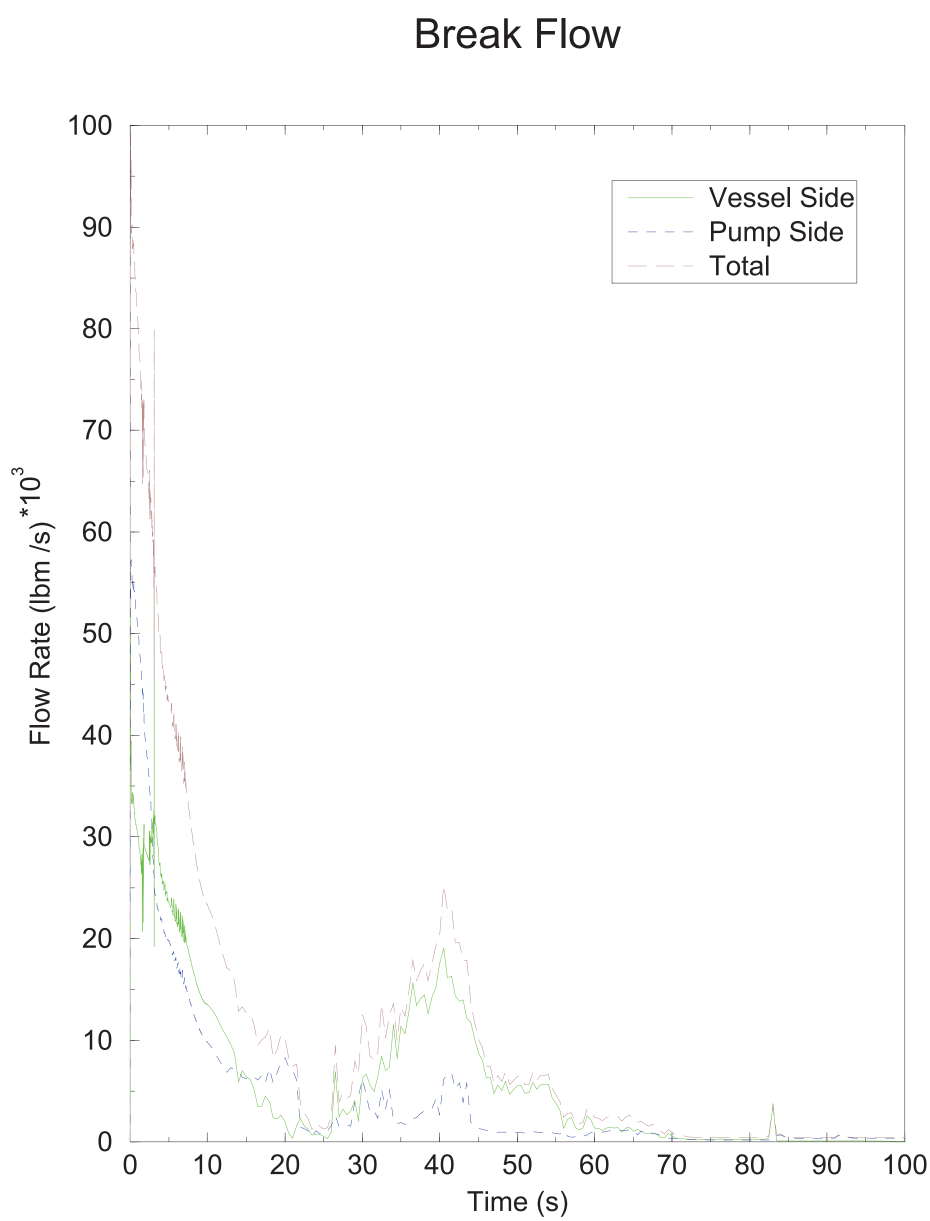

Figure 3-8
Core Inlet Mass Flux for the Demonstration Case

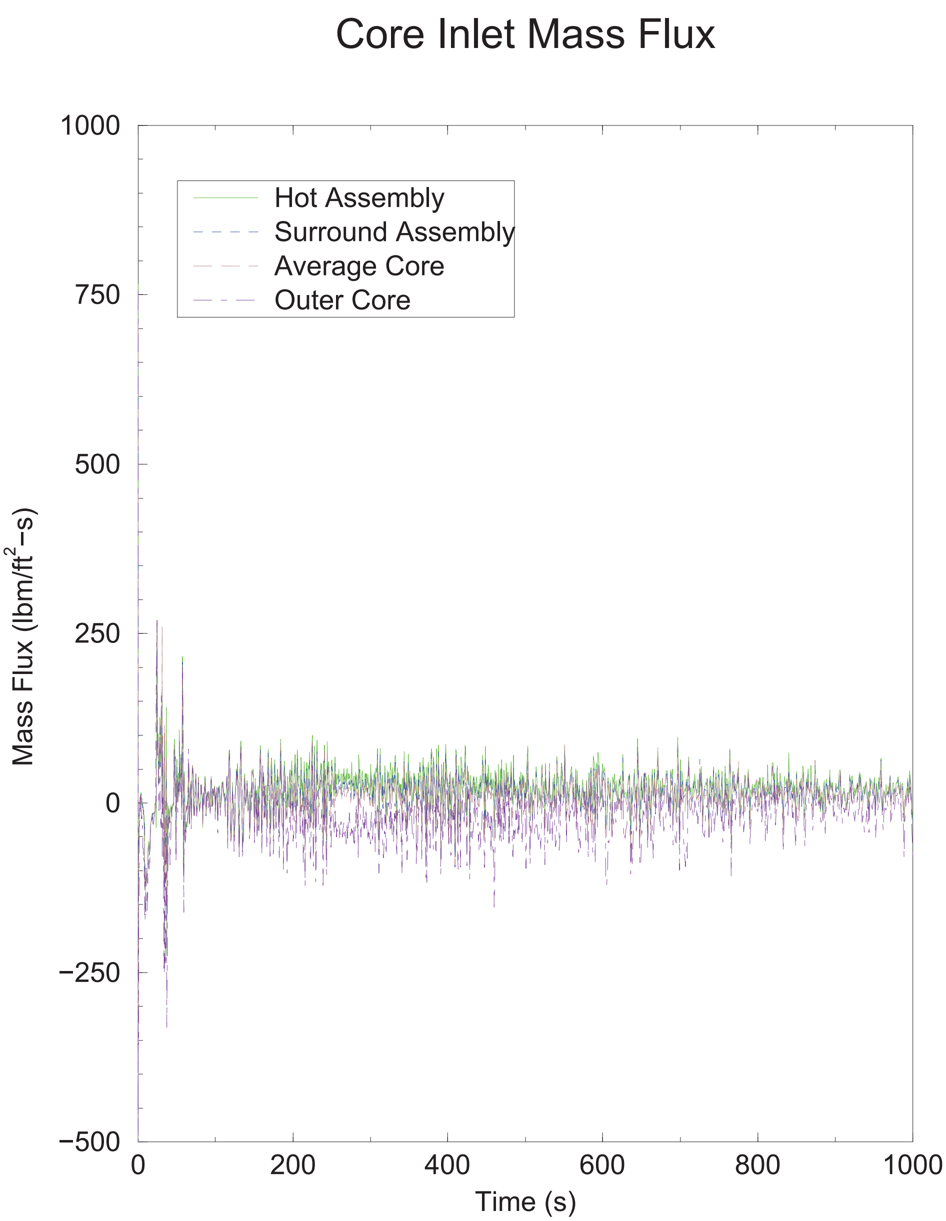

Figure 3-9
Core Outlet Mass Flux for the Demonstration Case

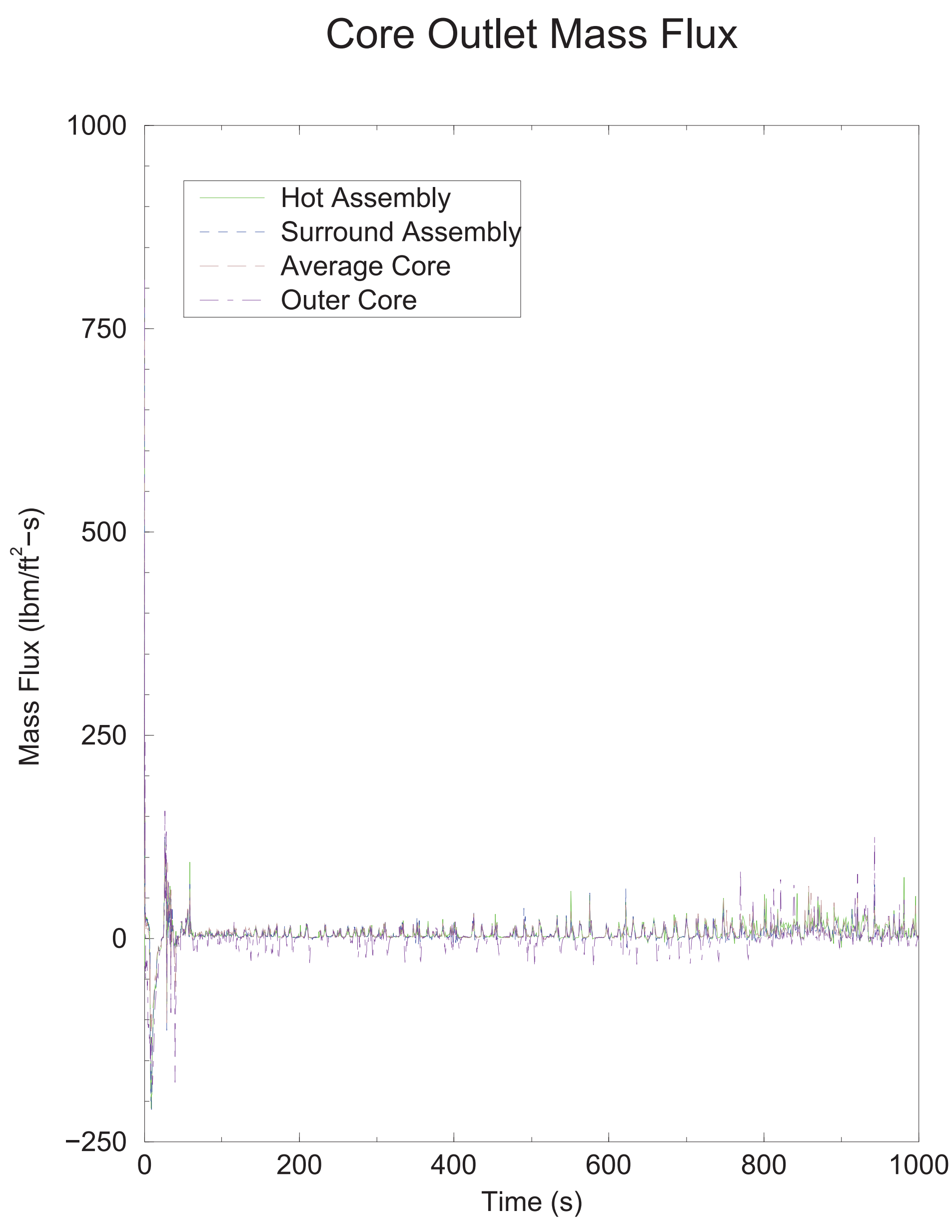

Figure 3-10
Void Fraction at RCS Pumps for the Demonstration Case

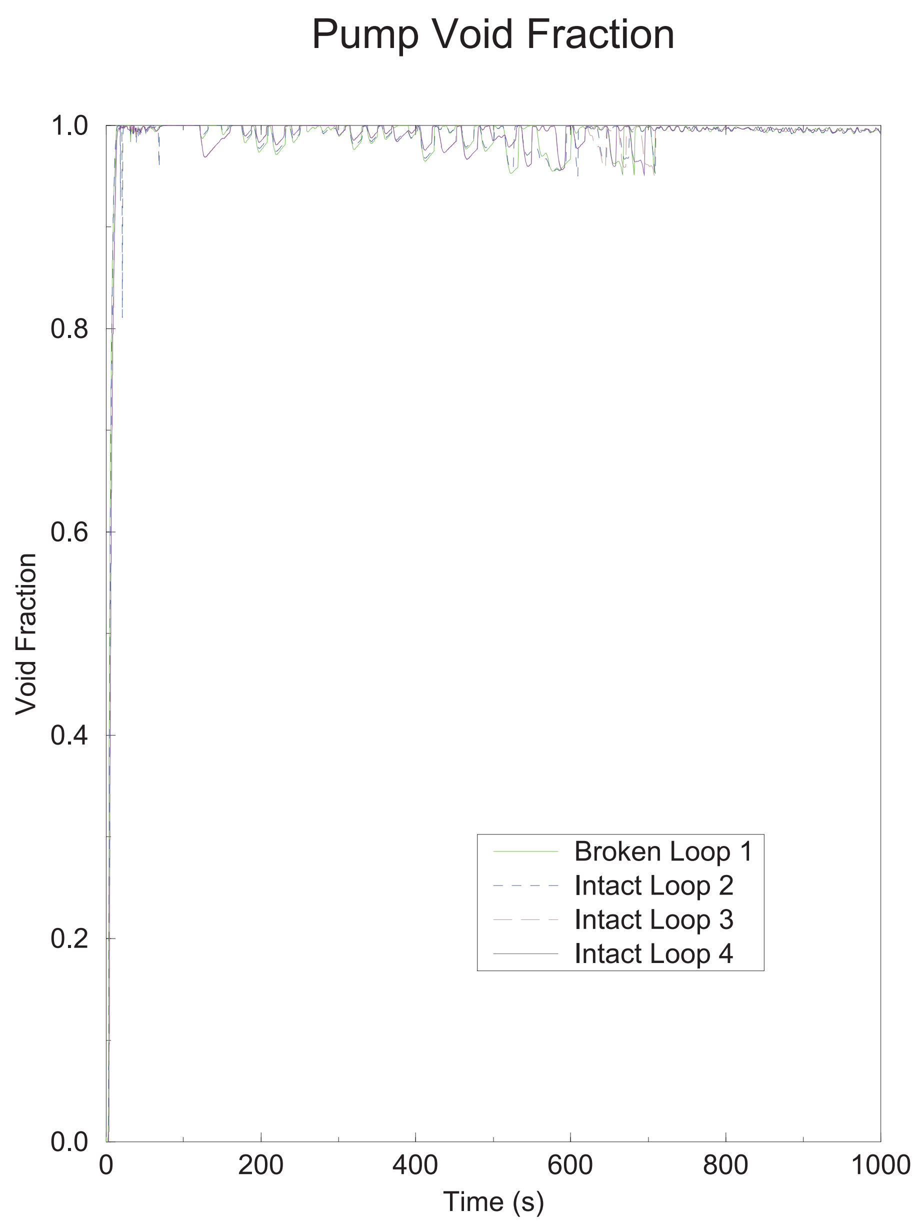

Figure 3-11
ECCS Flows (Includes SIT, HPSI and LPSI) for the Demonstration Case

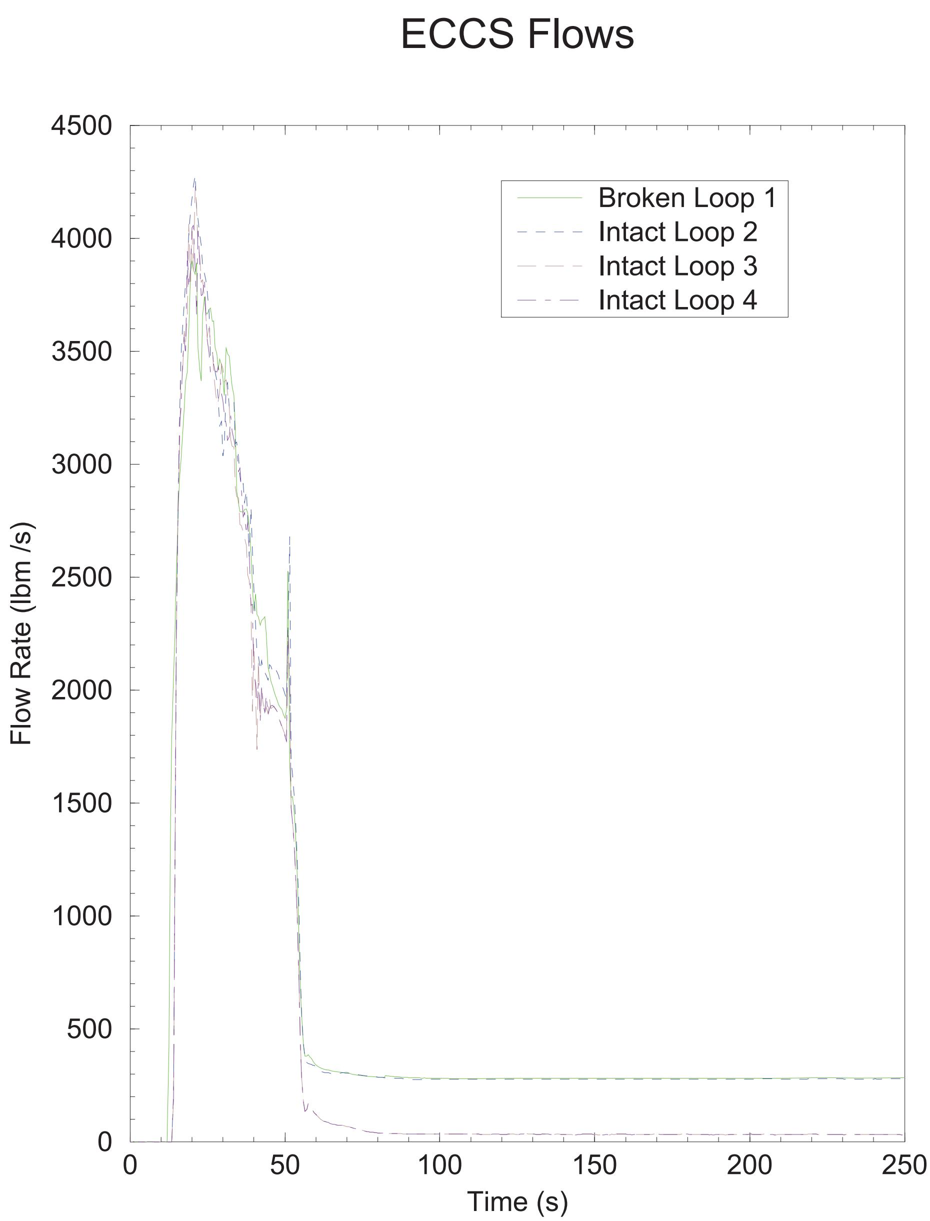

Figure 3-12
Upper Plenum Pressure for the Demonstration Case

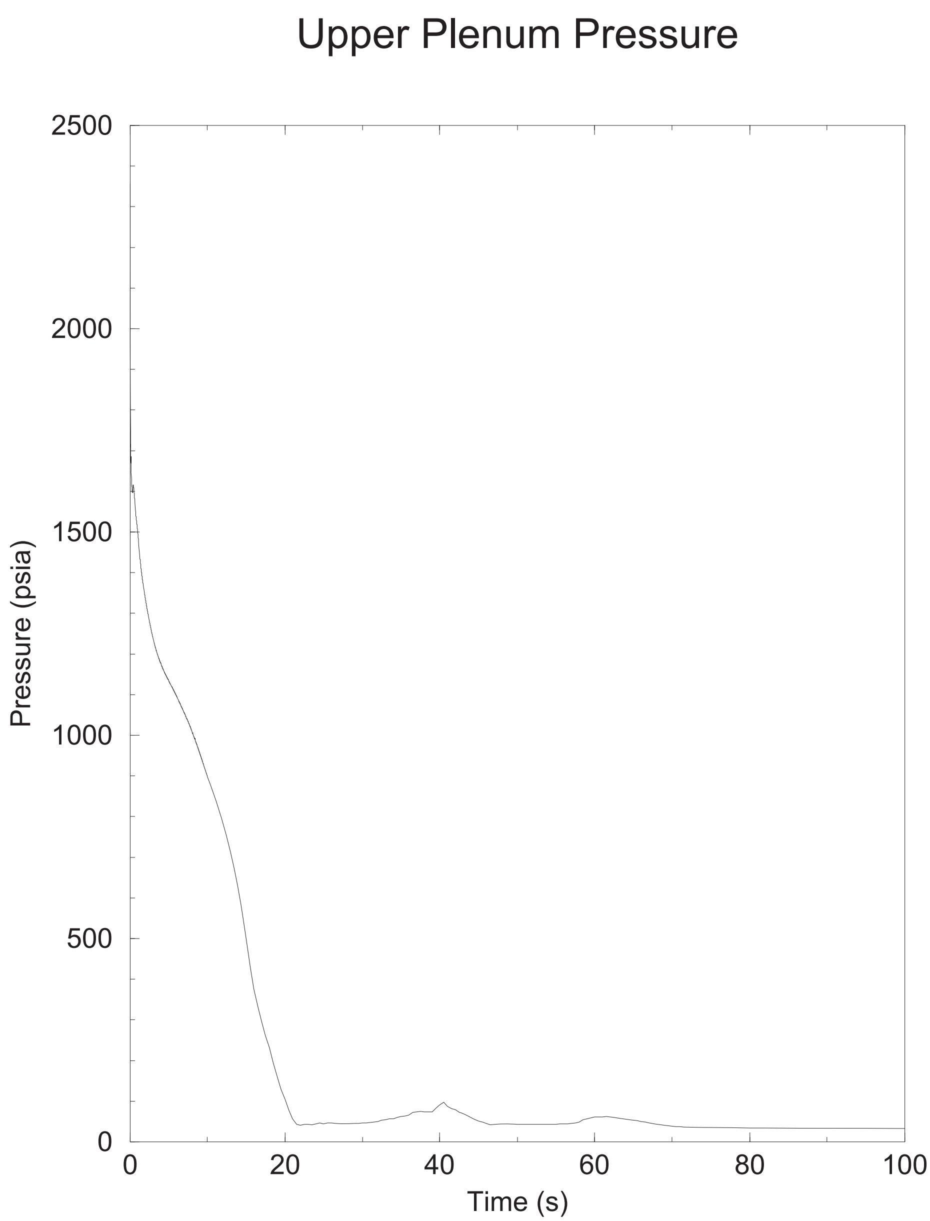

Figure 3-13
Collapsed Liquid Level in the Downcomer for the Demonstration Case

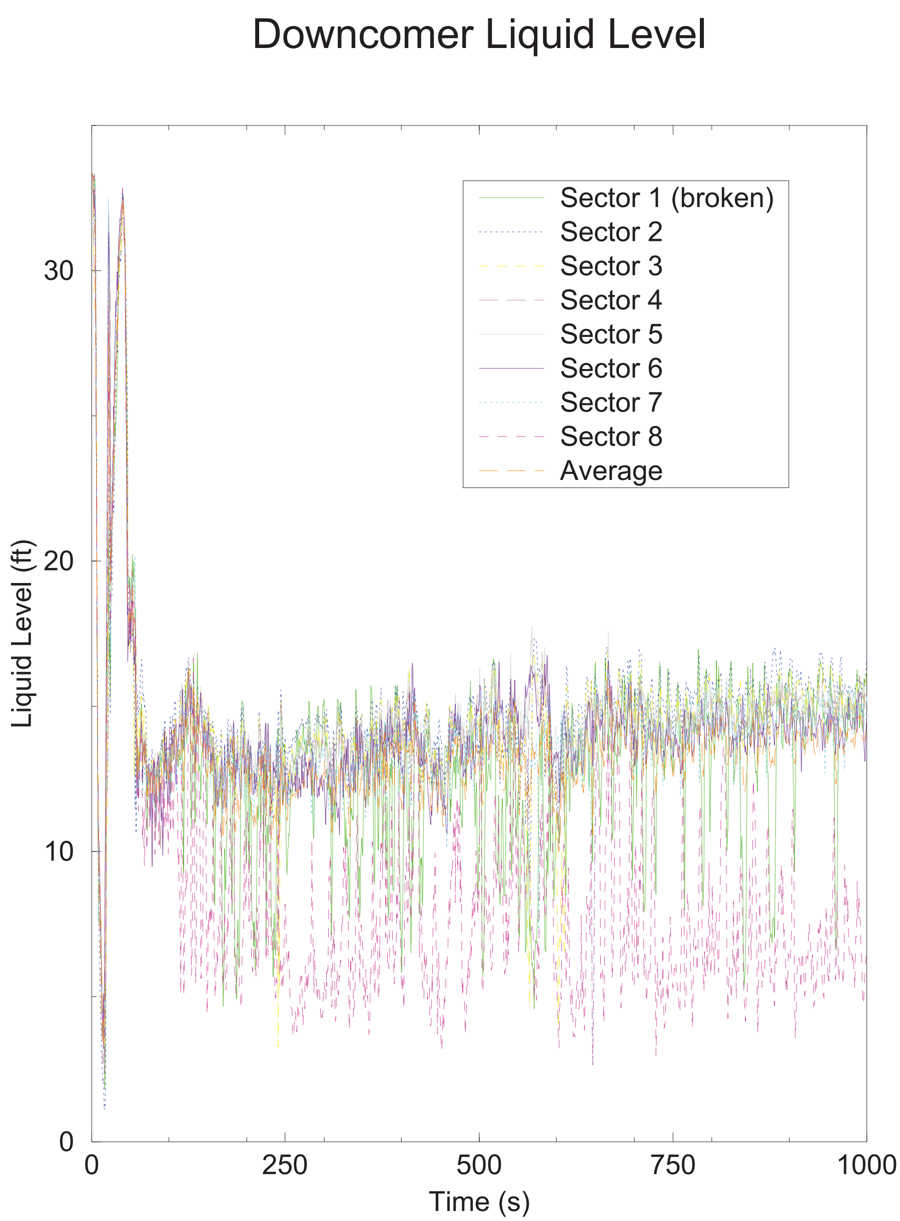

Figure 3-14
Collapsed Liquid Level in the Lower Plenum for the Demonstration Case

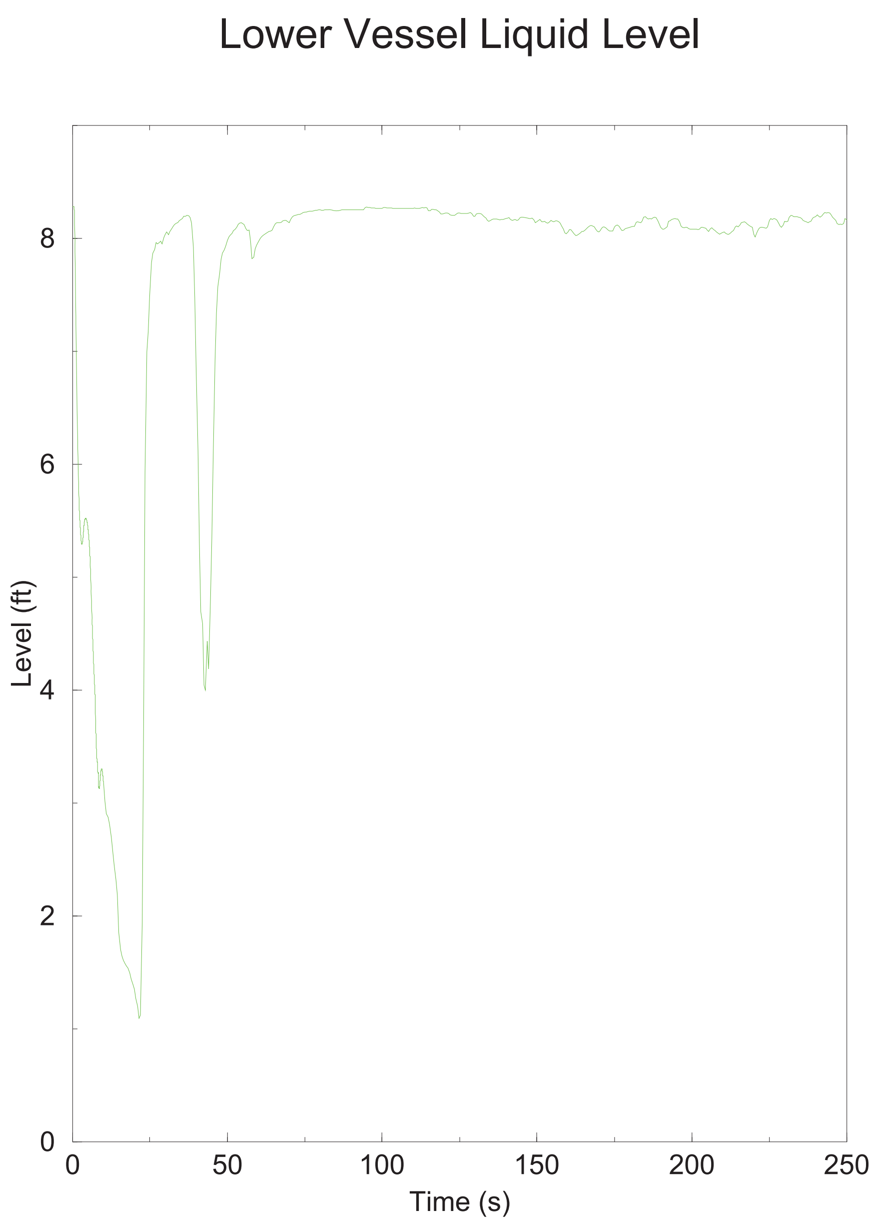

Figure 3-15
Collapsed Liquid Level in the Core for the Demonstration Case

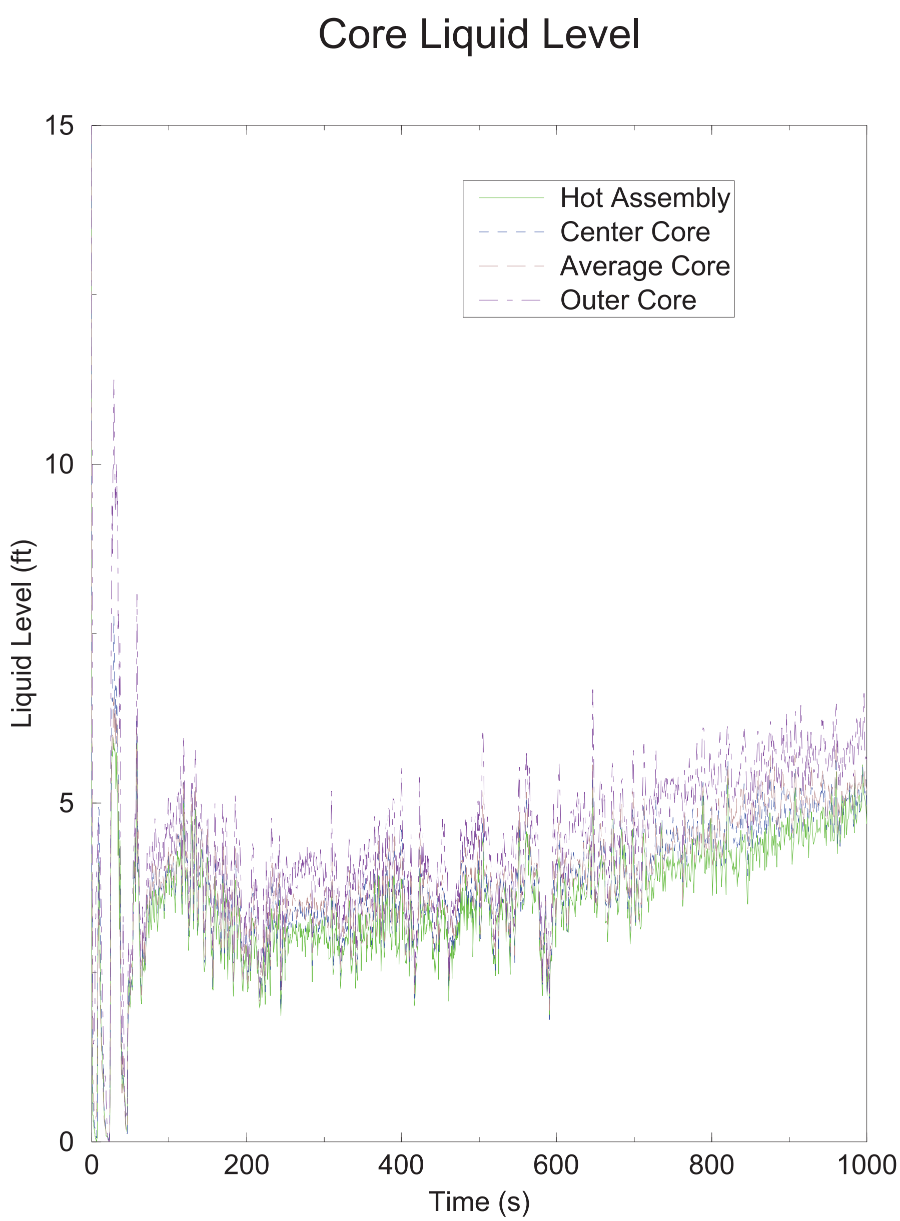

Figure 3-16
Containment and Loop Pressures for the Demonstration Case

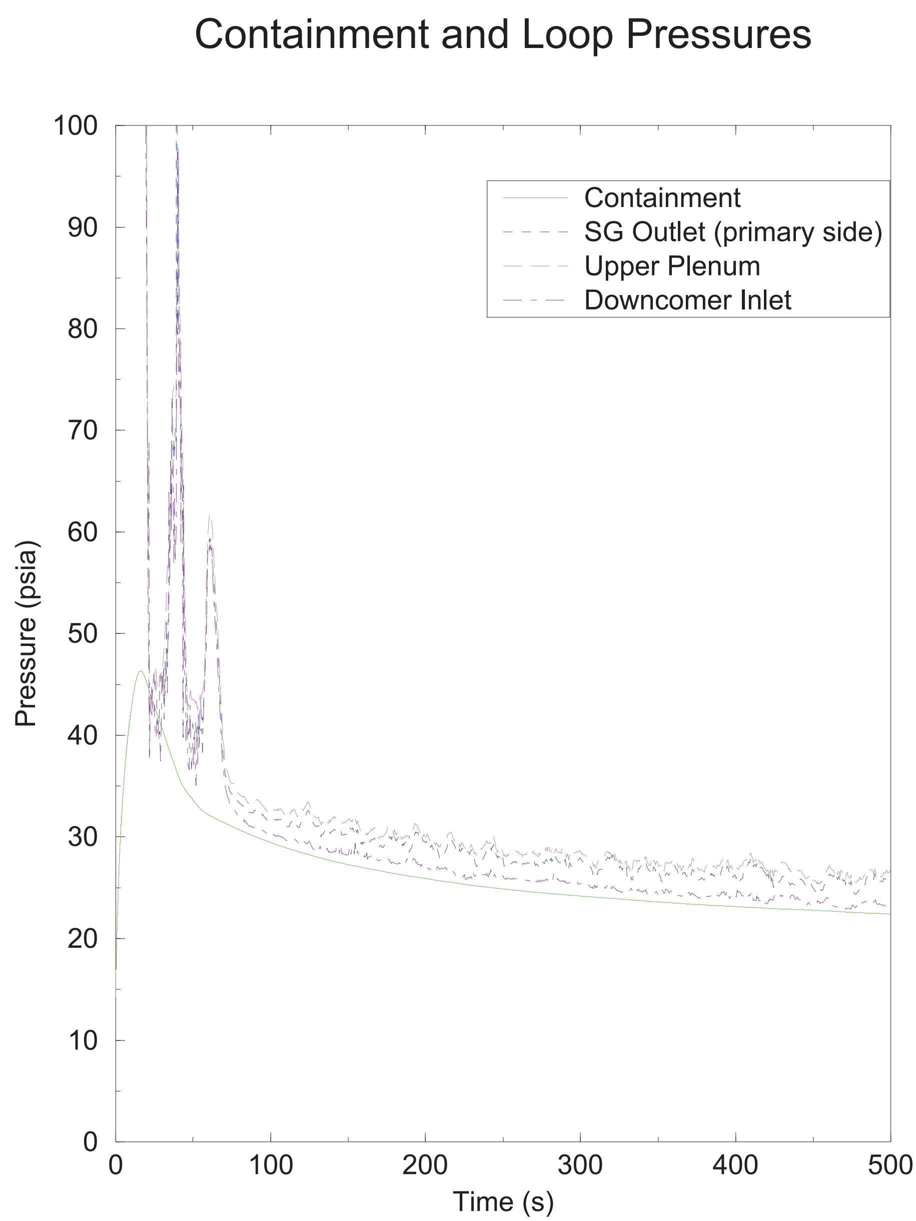

Figure 3-17
Pressure Differences between Upper Plenum and Downcomer for the Demonstration Case

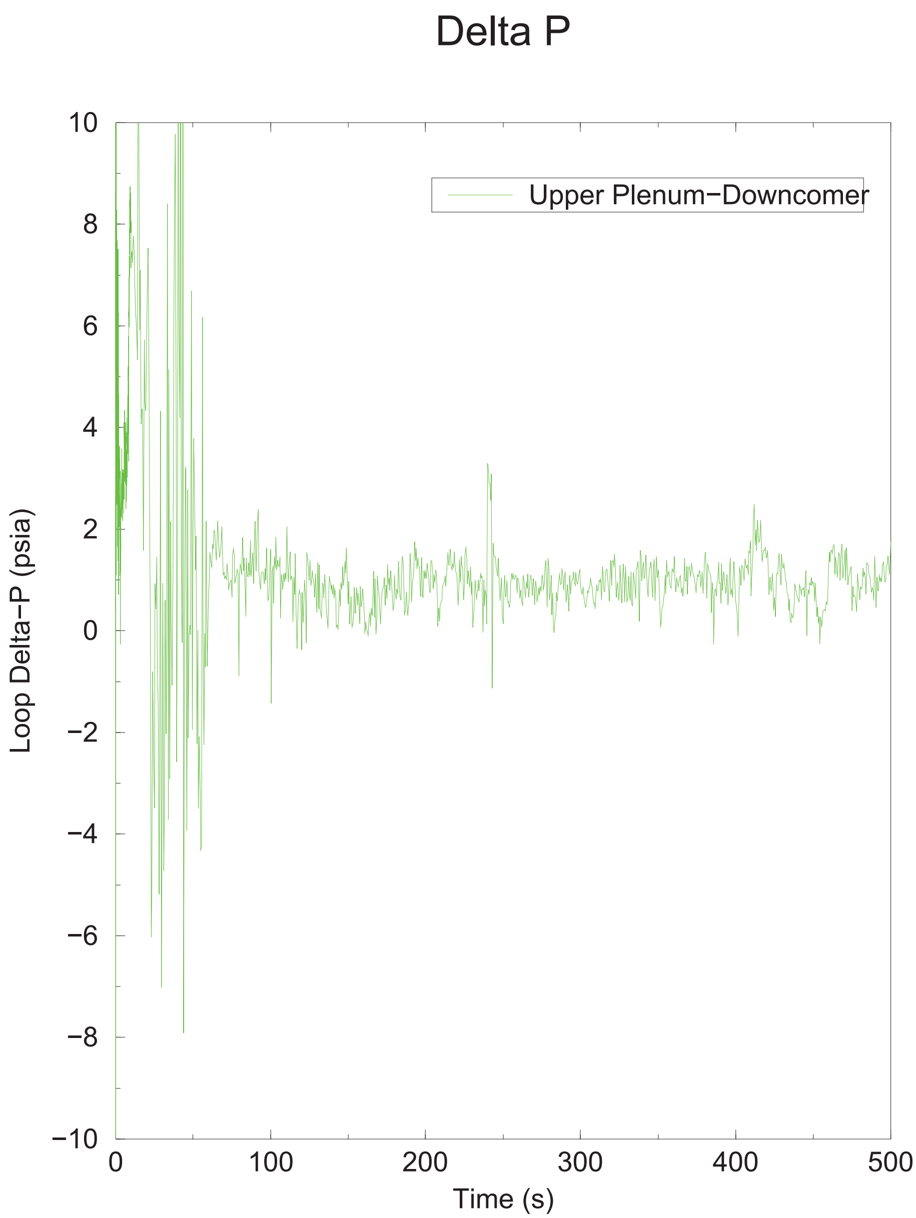

Figure 3-18
Validation of BOCR Time using MPR CCFL Correlation

Figure 3-19

[ ]

[ ]

4.0 REFERENCES

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

2. Code of Federal Regulations, Title 10, Part 50, Section 46, Acceptance Criteria For Emergency Core Cooling Systems For Light-Water Nuclear Power Reactors, August 2007.

3. NUREG/CR-5249, "Quantifying Reactor Safety Margins, Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large Break, Loss-of-Coolant Accident," U.S. NRC, December 1989.

4. Regulatory Guide 1.203, "Transient and Accident Analysis Methods" U.S. NRC, December 2005.

APPENDIX A SUMMARY OF KEY INPUT AND OUTPUT PARAMETERS

The following tables contain the sampled input values for all cases analyzed. Key results are also included in columns 2 through 6 in Table A-1 for the case set. In all cases, the core power is 4070 MWth with a [ ].

Table A-1
Summary of Key Input and Output Parameters, Part 1

[ ]

[ ]

Table A-2
Summary of Key Input and Output Parameters, Part 2

[ ]

Table A-3
Summary of Key Input and Output Parameters, Part 3

0414-12-F04 (Rev. 002, 01/15/2018)

framatome

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

ANP-3640NP
Revision 0

Licensing Report

March 2018

Framatome Inc.

Nature of Changes

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

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Nomenclature

Acronym	Definition
AFW	Auxiliary Feedwater
BOC	Beginning-of-Cycle
CE	Combustion Engineering
CEA	Control Element Assembly
CFR	Code of Federal Regulations
CWO	Core-Wide Oxidation
DC	Downcomer
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOC	End-of-Cycle
HMP™	High Mechanical Performance Spacer Grid
HTP™	High Thermal Performance Spacer Grid
HPSI	High Pressure Safety Injection
IOPSV	Inadvertent Opening of a Pressurizer Safety/Relief Valve
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LPSI	Low Pressure Safety Injection
LSC	Loop Seal Clearing
MFW	Main Feedwater
MLO	Maximum Local Oxidation
MSIV	Main Steam Isolation Valve
MSSV	Main Steam Safety Valve
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PCT	Peak Cladding Temperature
PVNGS	Palo Verde Nuclear Generating Station
PWR	Pressurized Water Reactor
PZR	Pressurizer
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protection System
RV	Reactor Vessel
SBLOCA	Small Break Loss-of-Coolant-Accident
SE	Safety Evaluation
SG	Steam Generator
SI	Safety Injection
SIAS	Safety Injection Actuation Signal

Acronym	**Definition**
SIT	Safety Injection Tank
TOAF	Top of Active Fuel
TS	Technical Specification
VQP	Vendor Qualification Program

1.0 INTRODUCTION

This report summarizes the small break LOCA (SBLOCA) analysis for Palo Verde Nuclear Generating Station (PVNGS). The purpose of the SBLOCA analysis is to support the Vendor Qualification Program (VQP) for PVNGS with the Framatome Advanced CE 16 x 16 HTP™[1] Fuel Design with M5®[1] cladding. This analysis was performed in accordance with the NRC-approved S-RELAP5 methodology described in Reference 1 and as modified by Reference 2. Reference 3 discusses the incorporation of M5® properties into the SBLOCA methodology.

PVNGS Units 1, 2, and 3 are 2x4-loop, CE-designed PWRs. The Framatome Advanced CE16 Fuel Design with M5® cladding for PVNGS consists of a 16x16 CE array with HTP™ intermediate grids and a lower HMP™[1] grid. The fuel assembly will include an M5® MONOBLOC™[1] guide tube design, M5® fuel rod design and FUELGUARD™[1] debris-resistant lower tie-plate design.

A complete spectrum of cold leg break sizes was considered, ranging from 1.00 to 9.49 inches in diameter. Other supporting analyses prescribed by the methodology were performed which consider a delayed RCP trip, attached piping break, and the sensitivity to reduced SI temperature. Two additional evaluations, outside of those prescribed by the methodology, were performed which considered inadvertent opening of a pressurizer safety/relief valve (IOPSV) and RV instrument tube rupture accidents.

The SBLOCA analyses supports plant operation at a core power level of 4070 MWt (including measurement uncertainty), a peak linear heat generation rate (LHGR) of 13.1 kW/ft, a radial peaking factor of 1.65, and up to 10% steam generator tube plugging.

[1] M5, HTP, HMP, MONOBLOC and FUELGUARD are trademarks or registered trademarks of Framatome or its affiliates, in the USA or other countries.

2.0 SUMMARY OF RESULTS

A SBLOCA break spectrum analysis was performed for PVNGS using the NRC-approved Framatome method (Reference 1) as modified by Reference 2. The analyses are performed to demonstrate that the following acceptance criteria for ECCS, as stated in 10 CFR 50.46(b)(1-4) (Reference 4), have been met.

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

The limiting peak cladding temperature (PCT) is 1620°F for a 9.10 inch diameter cold leg pump discharge break. The 8.80 inch diameter cold leg break produced the limiting maximum local oxidation (MLO) value. The limiting total MLO and limiting core wide oxidation (CWO) values for the spectrum are 2.96% and 0.006%, respectively. The total MLO value includes [ ]. The results of the analysis demonstrate the adequacy of the ECCS to support the 10 CFR 50.46(b) (1-4) criteria (Reference 4).

In addition to the cold leg pump discharge break spectrum analysis, three studies were performed to consider a delayed RCP trip, break in an attached pipe and sensitivity to reduced ECCS temperature. The results of the delayed RCP trip study demonstrated that there is at least 5 minutes for operators to trip all four RCPs after NPSH criteria are met. The attached piping study involved an 11.19 inch diameter break in the safety injection tank (SIT) line which connects to the cold leg. The ECCS temperature sensitivity study analyzed the effect of SI temperatures reduced from those used in the break spectrum analysis. The conclusions of these studies support the break spectrum analysis as the licensing basis.

Two additional analyses outside of those prescribed by the methodology were performed. The events included an IOPSV accident and an RV instrument tube rupture. Both analyses showed that the core remained covered during the transient and experienced no significant heatup. The conclusions of these additional studies support the break spectrum analysis as the licensing basis.

3.0 DESCRIPTION OF ANALYSIS

Section 3.1 of this report provides a brief description of the postulated SBLOCA event. Section 3.2 describes the analytical models used in the analysis. Section 3.3 presents a description of the PVNGS plant parameters and outlines the system parameters used in the SBLOCA analysis. Section 3.4 describes compliance with the NRC Safety Evaluation (SE) of the methodology.

3.1 *Description of SBLOCA Event*

The postulated SBLOCA is defined as a break in the RCS pressure boundary with an area less than or equal to 10% of the cold leg pipe area. The most limiting break location is in the cold leg pipe on the discharge side of the RCP. This break location results in the largest amount of RCS inventory loss, the largest fraction of ECCS fluid ejected out through the break, and the largest pressure drop between the core exit and the top of the downcomer (DC). This produces the greatest degree of core uncovery, the longest fuel rod heatup time, and consequently, the greatest challenge to the 10 CFR 50.46(b)(1-4) criteria (Reference 4).

The SBLOCA event progression develops in the following distinct phases: (1) subcooled depressurization (also known as blowdown), (2) natural circulation, (3) loop seal clearing, (4) core boil-off (5) core recovery and long-term cooling. The duration of each of these phases is break size and system dependent.

Following the break, the RCS rapidly depressurizes to the saturation pressure of the hot leg fluid. During the initial depressurization phase, a reactor trip is generated on low pressurizer pressure; the turbine is tripped on the reactor trip. The assumption of a loss-of-offsite-power concurrent with the reactor scram results in RCP trip.

In the second phase of the transient, the RCS transitions to a quasi-equilibrium condition in which the core decay heat, leak flow, SG heat removal, and system hydrostatic head balance combine to control the core inventory. During this period, the RCPs are coasting down and the system drains top down with voids beginning to form at the top of the SG tubes and continuing to form in the RV upper head and at the top of the RV upper plenum region. Also, the loop seals remain plugged during this phase, trapping vapor generated by the core in the RCS, resulting in a low quality flow at the break.

The third phase in the transient is characterized by loop seal clearing (LSC). During this phase, the loop seal, with liquid trapped in the RCP suction piping, can prevent steam from venting via the break. The maximum pressure difference between the RV upper head and DC is reached when the liquid level on the downhill side of the SG is depressed to the elevation of the horizontal loop seal piping. When this point is reached, loop seal upflow is pushed, clearing the loop seal, and the trapped steam can be vented to the break. For a small break, the transient develops slowly, and liquid level in the reactor coolant system may drop to the loop seal level prior to establishing a steam vent. The core can become temporarily uncovered in this LSC process. Following LSC, the break flow transitions to primarily steam and the core recovers to approximately the cold leg elevation, as pressure imbalances throughout the RCS are relieved.

The fourth phase is characterized as core boil-off. With the loop seal cleared, the venting of steam through the break causes a rapid RCS depressurization below the secondary pressure. As boiling increases in the core, the core mixture level decreases. The core mixture level will reach a minimum, in some cases resulting in deep core uncovery. The boil-off period of the transient ends when the core liquid level reaches this minimum. At this time, the RCS has depressurized to the point where ECCS flow into the RV matches the rate of boil-off from the core.

The last phase of the transient is characterized as core recovery. The core recovery period extends from the time at which the core mixture level reaches a minimum in the core boil-off phase until all parts of the core are quenched and covered by a low quality mixture. Core recovery is provided by pumped injection and passive SIT injection when the RCS pressure decreases below the SIT pressure. Generally, PCT occurs at the beginning of the core recovery phase before the mixture level has risen high enough to provide enhanced cooling to the PCT location on the hot rod.

The SBLOCA transient progression is dependent on the size of the break and is typically broken into three different break size ranges. For break sizes towards the larger end of the break spectrum, significant RCS inventory loss results in larger RCS depressurization to the SIT actuation pressure. SIT flow provides sufficient inventory early in the transient to limit the core uncovery and clad heatup, meaning that hot rod heatup is typically not limiting. For break sizes in the middle of the spectrum, the rate of inventory loss from the RCS is such that the HPSI pumps cannot preclude significant core uncovery. The RCS depressurization rate is slow, extending the time required to reach the SIT injection pressure, if SIT injection pressure is reached at all, or to recover core liquid level on HPSI flow. This tends to maximize the heatup time of the hot rod which produces the maximum PCT and local cladding oxidation. The limiting break case will either exhibit core recovery with the HPSI pumped injection alone while the RCS pressure remains barely above the SIT injection setpoint, or core recovery from SIT injection after an extended period of uncovery. For very small break sizes, the RCS pressure does not reach the SIT injection pressure. However, RCS inventory loss is not significant and typically within the means of HPSI makeup capacity such that core uncovery is minimal if not precluded.

3.2 *Analytical Methods*

The Framatome S-RELAP5 SBLOCA evaluation model for event response of the primary and secondary systems and the hot fuel rod used in this analysis is based on the use of two computer codes. The appropriate conservatisms, as prescribed by Appendix K of 10 CFR 50 (Reference 7), are incorporated. This analysis was performed in accordance with the NRC-approved S-RELAP5 methodology described in Reference 1 and as modified by Reference 2.

The two Framatome computer codes used in this analysis are:

1. The RODEX2-2A code (References 5 and 6) was used to determine the burnup-dependent initial fuel rod conditions for the system calculations.
2. The S-RELAP5 code was used to predict the thermal-hydraulic response of the primary and secondary sides of the reactor system and the hot rod response. The code version used addressed all known CRs and modeling issues at the time of analysis.

[ ]

The methodology (Reference 1, Reference 2) has been reviewed and approved by the NRC to perform SBLOCA analyses. However, several modeling differences from the current SBLOCA methodology (Reference 1, Reference 2) were included in this analysis, as described below.

The system nodalization is shown in Figure 3-2 through Figure 3-4. Note that Figure 3-2 (RCS) and Figure 3-3 (Secondary System) show representative system nodalization and minor variations for PVNGS specific details are not shown. For example, the charging system is not simulated in the SBLOCA analysis; therefore, the charging system noding diagram shown in Figure 3-2 is not used. Figure 3-4 (Reactor Vessel) is specific to PVNGS.

Figure 3-1

[ ]

[ ]

Figure 3-2
S-RELAP5 SBLOCA Reactor Coolant System Nodalization

Figure 3-3
S-RELAP5 SBLOCA Secondary System Nodalization

[

]

Figure 3-4
S-RELAP5 SBLOCA Reactor Vessel Nodalization

3.3 *Plant Description and Summary of Analysis Parameters*

All three units at PVNGS are CE-designed PWRs with two hot legs, four cold legs, and two vertical U-tube SGs. The reactor has a core power of 4070 MWt (including measurement uncertainty). The reactor vessel contains a DC, upper and lower plenums, and a reactor core containing 241 fuel assemblies. The hot legs connect the reactor vessel to with the vertical U-tube steam generators. Main feedwater (MFW) is injected into the DC of each SG. There are two safety-grade auxiliary feedwater (AFW) pumps per unit, one motor-driven and one turbine (steam)-driven. The ECCS contains two HPSI pumps, two LPSI pumps, and four SITs.

The RCS was nodalized in the S-RELAP5 model with control volumes interconnected by flow paths or "junctions." The model includes four SITs, a pressurizer, and two SGs with both primary and secondary sides modeled. All of the loops were modeled explicitly to provide an accurate representation of the plant. A SG tube plugging level of 10% was modeled in each SG. Important system parameters and initial conditions used in the analysis are given in Table 3-1. The heat generation rate in the S-RELAP5 reactor core model was determined from reactor kinetics equations with actinide and decay heating as prescribed by Appendix K.

The analysis assumed a loss-of-offsite power concurrent with reactor scram, which is based on the low pressurizer pressure reactor trip and includes delays for RPS actuation and control element assembly (CEA) coil delay. The assumption of loss-of-offsite power concurrent with reactor scram results in an RCP trip.

The RCPs are tripped at the time of reactor scram, instead of the opening of the break (time zero). This is considered to be conservative, since continued RCP operation will delay LSC. This delay in LSC will result in additional RCS inventory loss since the break flow is mostly liquid until the time of LSC. After LSC, a path for steam venting is established and the break flow transitions from liquid to steam, lowering the break mass flow rate.

The single failure criterion required by 10 CFR 50 Appendix K (Reference 7) was satisfied by assuming the loss of one EDG. Thus, this results in the loss of one HPSI pump, one LPSI pump and one motor-driven AFW pump. The initiation of the HPSI and LPSI systems was delayed by 30 seconds following safety injection actuation signal (SIAS) activation.

Table 3-2 and Table 3-3 show the minimum ECCS flow rates with one EDG failure for HPSI and LPSI, respectively. The HPSI system was modeled to deliver the highest SI flow to the broken leg (Loop 2B). The LPSI system was modeled to deliver the SI flow to the loop containing the broken leg (Loop 2B). Although the charging system is considered safety grade, it was not modeled in the analysis.

The disabling of a motor-driven AFW pump due to assumed EDG failure leaves the turbine-driven pump available. This allows 100% flow to be sent to either SG or 50% flow to be sent to both. The input model included the main steam lines between their respective SGs and the turbine control valve, including the connected MSSV inlet piping. The MSSVs were set to open at their nominal setpoints plus 3% tolerance.

The axial power shapes for this analysis are shown in Figure 3-5. Figure 3-5 shows the input axial power shape and the axial power shape after being adjusted so that it is consistent with the Technical Specification peaking and radial peaking factors.

For the IOPSV and RV instrument tube rupture analyses, symmetric HPSI flow splits are applied. These are shown in Table 3-5. The IOPSV and RV instrument tube breaks are located far from the cold leg injection points, and therefore there is no need to conservatively bias more ECCS flow to the broken leg.

3.4 *SE Compliance*

The NRC-approved supplemented EMF-2328 method (Reference 1 and Reference 2) contains no restrictions. The analysis was performed in accordance with the approved methodology except as indicated in Section 3.2.

Table 3-1
System Parameters and Initial Conditions

Parameter	Analysis Value
Core Thermal Power, (MWt)	4070[2]
Peak LHGR, (kW/ft)	13.1
Radial Peaking (Fr)	1.65
Axial Power Shape	Figure 3-5
RCS Flow Rate (gpm)	424311
RCS Operating Temperature (°F)	566
Pressurizer Pressure (psia)	2250
Pressurizer Level (%)	52.6
Steam Generator Pressure (psia)	1039.6
MFW temperature (°F)	448
RPS Low Pressurizer Pressure Trip Safety Analysis Setpoint (psia)	1670
RPS Low Pressurizer Pressure Trip Delay Time (seconds)	1.15
RPS Scram Delay (seconds)	0.6
RCP Trip Criteria – Break Spectrum Analysis	All operating initially, trip on reactor/turbine trip
SIAS Actuation - Pressurizer Low Pressure Setpoint (psia)	1670
AFW Flow Rate, single pump (gpm)	See Table 3-4
AFW Temperature (°F)	120
AFW SG Low Level setpoint (%WR)	20
AFW flow initiation delay after pump start (seconds)	46
SIT Pressure (psia)	602
SIT Water Volume (ft^3)	1850
SIT Fluid Temperature (°F)	120
HPSI and LPSI fluid temperature (°F)	130
HPSI and LPSI delay time (sec)	30
SG tube plugging (%)	10

[2] Includes measurement uncertainty.

Parameter	Analysis Value
MSSV Lift Pressures (psig)	1315 1315 1315 1290 1250

Table 3-2
High Pressure Safety Injection Flow Rates for Cold Leg Breaks

RCS Pressure (psia)	Flow Rate (gpm)			
	Loop 1A	Loop 1B	Loop 2A	Loop 2B (broken)
14.2	231.17	231.17	231.17	256.50
64.2	227.76	227.76	227.76	252.72
114.2	224.11	224.11	224.11	248.67
144.2	221.92	221.92	221.92	246.24
214.2	216.81	216.81	216.81	240.57
324.2	208.29	208.29	208.29	231.12
619.2	183.96	183.96	183.96	204.12
796.2	167.41	167.41	167.41	185.76
1007.2	145.27	145.27	145.27	161.19
1213.2	120.21	120.21	120.21	133.38
1363.2	98.55	98.55	98.55	109.35
1497.2	74.70	74.70	74.70	82.89
1595.2	49.88	49.88	49.88	55.35
1714.2	0.49	0.49	0.49	0.54

Table 3-3
Low Pressure Injection Flow Rates for Cold Leg Breaks

RCS Pressure (psia)	**Flow Rate (gpm)**			
	Loop 1A	Loop 1B	Loop 2A	Loop 2B (broken)
14.2	0.0	0.0	1872.0	1872.0
25.2	0.0	0.0	1801.0	1801.0
54.2	0.0	0.0	1603.0	1603.0
64.2	0.0	0.0	1510.0	1510.0
77.2	0.0	0.0	1403.0	1403.0
96.2	0.0	0.0	1215.0	1215.0
114.2	0.0	0.0	1012.5	1012.5
117.2	0.0	0.0	1000.0	1000.0
126.2	0.0	0.0	750.0	750.0
138.2	0.0	0.0	500.0	500.0
144.2	0.0	0.0	333.5	333.5
147.2	0.0	0.0	250.0	250.0
156.2	0.0	0.0	0.0	0.0

Table 3-4
Auxiliary Feedwater Flow Rate, Single Pump

SG Pressure (psia)	Volumetric Flow Rate (gpm)
736	1203.23
900	1057.09
1041	918.14
1270	647.82
1333	556.73

Figure 3-5
Axial Power Distribution Comparison

[ ]

Table 3-5
High Pressure Safety Injection Flow Rates for IOPSV and RV Instrument Tube Breaks

RCS Pressure (psia)	Flow per Loop, Intact (gpm)	Flow to Broken Loop (gpm)
14.2	237.50	237.50
64.2	234.00	234.00
114.2	230.25	230.25
144.2	228.00	228.00
214.2	222.75	222.75
324.2	214.00	214.00
619.2	189.00	189.00
796.2	172.00	172.00
1007.2	149.25	149.25
1213.2	123.50	123.50
1363.2	101.25	101.25
1497.2	76.75	76.75
1595.2	51.25	51.25
1714.2	0.50	0.50
1715.0	0.00	0.00

4.0 ANALYTICAL RESULTS

The analysis results demonstrate the adequacy of the ECCS to support the criteria given in 10 CFR 50.46(b)(1-4) for PVNGS Units 1, 2, and 3 operating with Framatome supplied Advanced CE16 Fuel with M5® cladding.

Section 4-1 describes the SBLOCA break spectrum for the cold leg break. Section 4.2 describes the event for the limiting break size. Section 4.3 discusses the delayed RCP trip study. Section 4.4 discusses the attached piping break study. Section 4.5 discusses the SI low fluid temperature sensitivity study. Section 4-5 discusses the IOPSV and RV instrument tube rupture event analyses.

4.1 *Results for Break Spectrum*

The PVNGS break spectrum analysis for SBLOCA includes breaks of varying diameter up to 10% of the flow area for the cold leg. The spectrum includes a break size range from 1.0 to 9.49 inches in diameter, which is wide enough to establish a PCT trend. Additional break sizes are analyzed with a smaller break interval once the potential limiting break size is determined to confirm the limiting break size. Figure 4-1 shows the calculated PCTs for these breaks. For the break spectrum analysis, RCP trip is assumed to occur on reactor scram.

The results of the cold leg SBLOCA break spectrum analysis are presented in Table 4-1. The predicted event times for the break spectrum are provided in Table 4-2. The limiting PCT break size was determined to be 9.10 inches in diameter (0.45166 ft^2), resulting in a PCT of 1620°F. The 8.80 inch break size yielded the highest transient MLO from the spectrum. The limiting total MLO and limiting CWO values for the spectrum are 2.96% and 0.006%, respectively.

4.2 *Discussion of Transient for Limiting PCT Break*

The limiting PCT break spectrum case is a 9.10 inch diameter cold leg break. The PCT of this case is 1620°F. The break opens at t=0 seconds and initiates a subcooled depressurization of the RCS. The low pressurizer pressure trip setpoint is reached at 12 seconds and at 14 seconds the reactor is scrammed, coincident with the RCP, MFW, and turbine trips (Figure 4-2, Figure 4-10, Figure 4-11, and Table 4-2). The pressure in the secondary side begins to rise but does not reach the MSSV set points, which remain closed for the duration of the transient (Figure 4-12).

The SIAS is issued at 12 seconds. Following the EDG delay and associated valve delays, the HPSI begins to inject at 42 seconds (Figure 4-17). However, HPSI does not provide sufficient inventory to offset the large amounts lost out the break at this time (Figure 4-20). Therefore, the core begins to uncover at 55 seconds, with effective cooling of the majority of the hot assembly lost in a short period of time (Figure 4-21, Figure 4-22).

All four loop seals clear before PCT, with the broken loop clearing first after 89 seconds, two additional loop seals clearing 5 seconds later, and the final loop seal clearing at 142 seconds (Figure 4-6, Table 4-2). The first three LSCs and the last LSC produce two temporary increases in core level at approximately 100 and 150 seconds, respectively (Figure 4-21, Figure 4-22). However, mixture level remains well below the hot upper regions of the core during both of the increases, resulting in continued poor cooling in the upper regions of the core and allowing the clad temperature excursion to proceed (Figure 4-23).

The SITs inject at 169 seconds (Figure 4-19). The minimum RV mass occurs at 180 seconds (Figure 4-9). There is a time delay from the SIT injection to the mixture level reaching sufficient levels to cool the upper locations in the core. The delay results in a rupture of the hot rod after 185 seconds (Table 4-1). The rupture allows for interior metal-water reaction, thereby increasing the local oxidation at the rupture node.

The cladding temperature excursion is terminated at 186 seconds with a PCT of 1620ºF (Figure 4-23). The core is quenched at approximately 210 seconds with SIT injection ending about 100 seconds later. At this point enough decay heat is being removed and adequate mixture level is sustained by mainly HPSI flow injection (Figure 4-17). LPSI activates two times, at 243 seconds and then again at around 750 seconds. However, since LPSI actuation begins well after the time of PCT, the effects of LPSI on the transient mitigation are considered minimal (Figure 4-18).

4.3 *Delayed RCP Trip Study*

The delayed RCP trip study is performed in accordance with the NRC-approved supplement to the EMF-2328 methodology (Reference 2). For plants such as PVNGS that do not have an automatic RCP trip, a delayed RCP trip can potentially result in a more limiting condition than tripping the RCPs at reactor scram. Continued operation of the RCPs can result in earlier LSC with associated two-phase flow out the break, which would result in less inventory loss out the break early in the transient, but in the longer term could result in more overall inventory loss out the break. It has been postulated that tripping the pumps when the minimum RCS inventory occurs could cause a collapse of voids in the core, thus depressing the core level and provoking a deeper core uncovery, and a potentially higher PCT. Therefore, the methodology prescribes an RCP trip study for both the cold and hot leg breaks consistent with the plant licensing basis and Emergency Operating Procedures. For Palo Verde, a delayed RCP trip time of 5 minutes following loss of subcooling margin at an assumed pressure of 1471 psia was analyzed to demonstrate 10 CFR 50.46(b)(1-4) criteria (Reference 4).

The spectrum of cold and hot leg breaks in this study includes break sizes from 3.00 to 9.49 inches. Based on the break spectrum results, it was determined that break sizes smaller than 3.00 inches would not present a challenge to the criteria since the pump will trip before the time the break uncovers. [

]

The results of the delayed RCP trip cases indicate that there is at least 5 minutes of expected operator time to trip all four RCPs after NPSH criteria are met with considerable margin to the 10 CFR 50.46(b)(1-4) criteria.

4.4 *Attached Piping Break Study*

The ECCS must cope with ruptures of the main RCS piping and breaks in attached piping. To demonstrate this, as prescribed by the NRC-approved supplement to EMF-2328 (Reference 2), an analysis of the ruptures in attached piping that compromise the ability to inject emergency coolant into the RCS is performed. The size of the rupture and the portion of ECCS lost directly to containment are dependent on the plant design. For PVNGS, the limiting break location and size for attached piping is considered a double-ended guillotine break of 11.19 inches in diameter in the SIT line connecting to the cold leg of loop 2B.

The SIT line break resulted in a PCT of 1367°F, transient MLO of 0.07%, and CWO of 0.001%, which are bounded by the results of the break spectrum analysis. The HPSI and LPSI flow rates modeled were sufficient to prevent a subsequent heatup after the initial quench from the SIT discharge.

4.5 *ECCS Temperature Sensitivity Study*

[ ]

4.6 *Additional Events Resulting in Decreased RCS Inventory*

Two additional events outside of those prescribed by the methodology, which result in a decrease in RCS inventory, were analyzed to support the PVNGS VQP. The results of the IOPSV accident and RV instrument tube rupture are discussed in the following sections. Note that both of the events used symmetric HPSI flows as shown in Table 3-5. Due to the differences in break location and system response compared to the break spectrum and its associated studies, a detailed description of the transient will be provided for each event.

4.6.1 IOPSV Accident Results

The IOPSV break size has been defined as a 0.03 ft^2 break in the pressurizer. The sequence of events are provided in Table 4-4. The break flow is steam-prevalent, but the pressurizer level (Figure 4-24) increases to the extent that some liquid passes through the break and break flow is initially quite high as a result. A single train of HPSI initiates at 100.1 seconds (Figure 4-25) and the flow increases as the RCS pressure drops (Figure 4-26).

The reactor, RCPs, and turbine trip after 71.9 seconds on low RCS pressure. Per SBLOCA methodology, a loss of offsite power is assumed to occur coincident with reactor scram. As a result, MFW pumps (Figure 4-28) and RCPs consequently coast down. After reactor/turbine trip occurs, the core heat production is reduced (Figure 4-32) and the core heat removal is mainly accomplished through the break flow and heat transfer to the SGs. AFW is initiated on low SG level (Figure 4-30, Figure 4-31) just before 1000 seconds and MSSVs lift periodically until approximately 2000 seconds (Figure 4-27). A quasi-steady plateau is established as a result of this heat balance with the primary pressure somewhat greater than the secondary steam pressure.

Due to the net loss of system mass to the break (Figure 4-33) combined with RCS pressure reduction, steam is produced at the highest elevations of the system (pressurizer, SG tubes, and RV upper head and plenum). A pressure difference builds between the core exit and the RCS loop seal, and the loop seals clear in loop 2A and briefly in loop 2B (Figure 4-34). Note that the IOPSV model has no artificial biasing of the loop seals. For the IOPSV, steam flows from the core directly through the broken hot leg/pressurizer and the location and number of loop seals cleared is therefore unimportant to the transient progression of this analysis. After LSC, the break flow void fractions are higher, decay heat is reduced, and most of the core heat is removed by the pressurizer safety valve.

The RCS pressure begins to drop again about 3000 seconds, increasing the HPSI flow rate. Towards the end of the transient, the HPSI flow rate exceeds that of the break (Figure 4-25) and RV and RCS mass begin to increase slowly (Figure 4-33) indicating a steady re-fill of the system. Over the duration of the transient, the pressure is never reduced to the point where the SIT or LPSI actuate.

Although the core coolant is partially voided during the event (Figure 4-35), it remained covered by a two-phase mixture (Figure 4-36). Therefore, no coolant or cladding temperature excursion occurs (Figure 4-37, Figure 4-38). The PCT of 686°F occurs at transient initiation. The transient MLO and CWO reached were less than 0.01% and 0.001%, respectively. The analysis results are provided in Table 4-3. The occurrence of an IOPSV accident is bounded in consequence by the SBLOCA break spectrum analyses.

4.6.2 RV Instrument Tube Break Results

In addition to the events discussed above, the rupture of an in-core instrument tube was also considered. The following constitutes a qualitative assessment of the results of the case. A break, equal in size to a completely severed instrument tube (0.003 ft^2), was postulated to occur in the RV bottom head. Long-term cooling is implemented one hour following the break, so the instrument tube rupture case is assessed for this amount of time.

Following break initiation, the RCS pressure drops until it reaches the reactor trip and SIAS low pressure setpoints. Turbine trip occurs on reactor trip. Per SBLOCA methodology, a loss of offsite power occurs coincident with reactor scram. As a result, MFW pumps and RCPs coast down. A single failure is assumed so that only one EDG is started and, after a delay, the generator provides electrical power to one HPSI and one AFW pump. LPSI and passive injection via SITs are available, but for such a small break RCS pressure remains too high for them to be of use.

The RCS pressure continues to drop but, at this point steam flow to the turbine has been isolated. The secondary system pressure increases to the MSSV actuation setpoint and AFW initiates on low SG level. Therefore, the core heat is mainly removed by heat transfer to the SGs. A near-steady heat balance is then attained with the primary temperature somewhat above the temperature of the secondary. The break is located at the bottom of the RV and never relieves steam, limiting the RCS pressure response. Because both the primary and secondary systems are at saturated fluid conditions the primary pressure stalls at a value slightly greater than the secondary pressure.

At the end of the transient, HPSI mass flow rates have not quite risen to match break flow. However, primary system fluid inventory is controlled to the extent that the core remains covered. Therefore, there is no significant clad or core coolant temperature excursion for this event. The occurrence of an instrument tube rupture in the bottom of the RV is bounded in consequence by the SBLOCA break spectrum analyses.

Table 4-1
Summary of SBLOCA Break Spectrum Transient Results

Table 4-1 Notes:

(a) There is no significant transient heat up and therefore the PCT is equal to the initialized temperature

(b) [ ]

(c) [ ]

Table 4-2
Sequence of Events for Break Spectrum (seconds)

[ ]

Table 4-2
Sequence of Events for Break Spectrum (seconds) (cont.)

Table 4-2
Sequence of Events for Break Spectrum (seconds) (cont.)

Table 4-2
Sequence of Events for Break Spectrum (seconds) (cont.)

[ ]

Table 4-2 Notes:

(a) There is no significant transient heat up and therefore the PCT is equal to the initialized temperature

(b) [ ]

(c) [ ]

(d) [ ]

(e) First switch to HEM break model occurs after the time of PCT.

Figure 4-1
Peak Cladding Temperature vs. Break Size (SBLOCA Break Spectrum)

Figure 4-2
Reactor Power – 9.10 inch Break

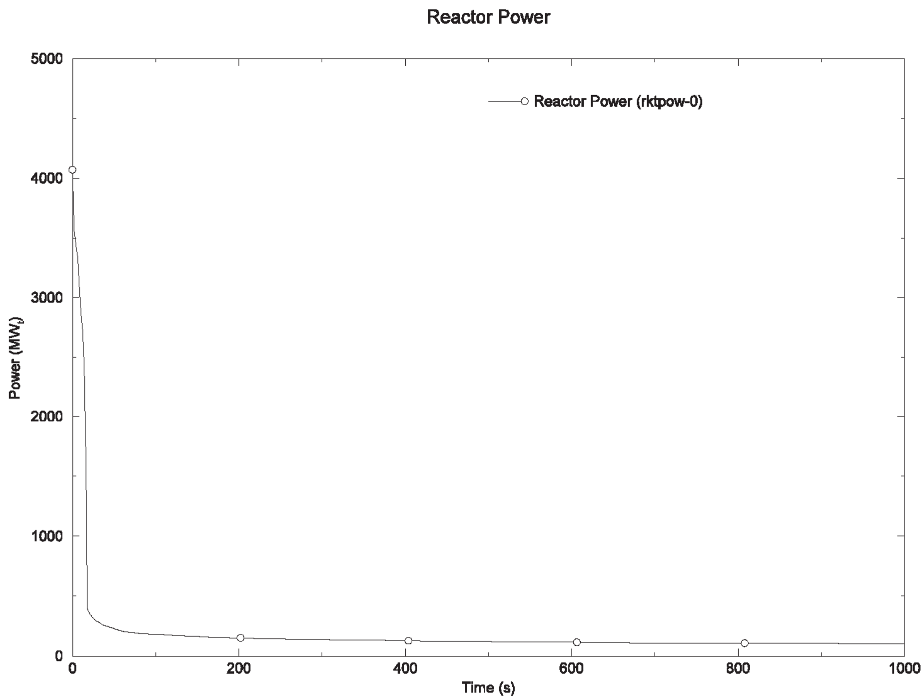


Figure 4-3
Primary and Secondary System Pressures – 9.10 inch Break

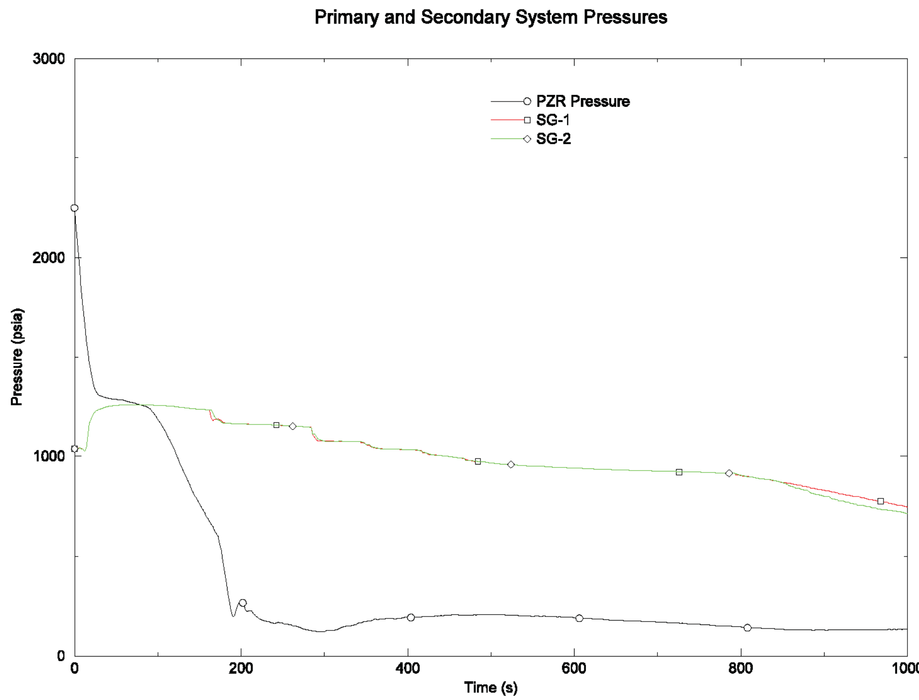


Figure 4-4
Break Mass Flow Rate – 9.10 inch Break

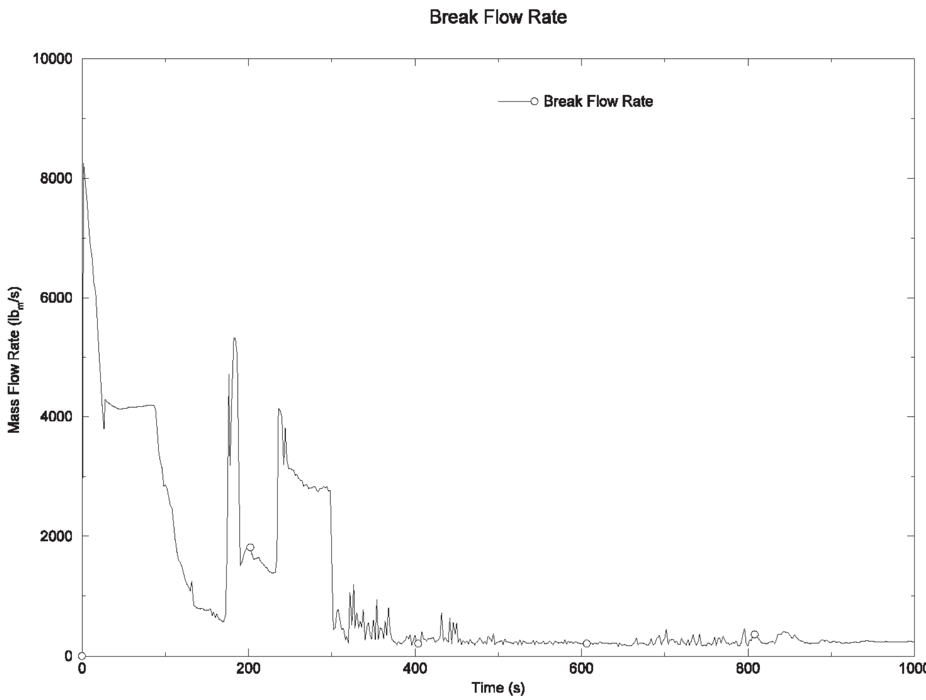


Figure 4-5
Break Vapor Void Fraction – 9.10 inch Break

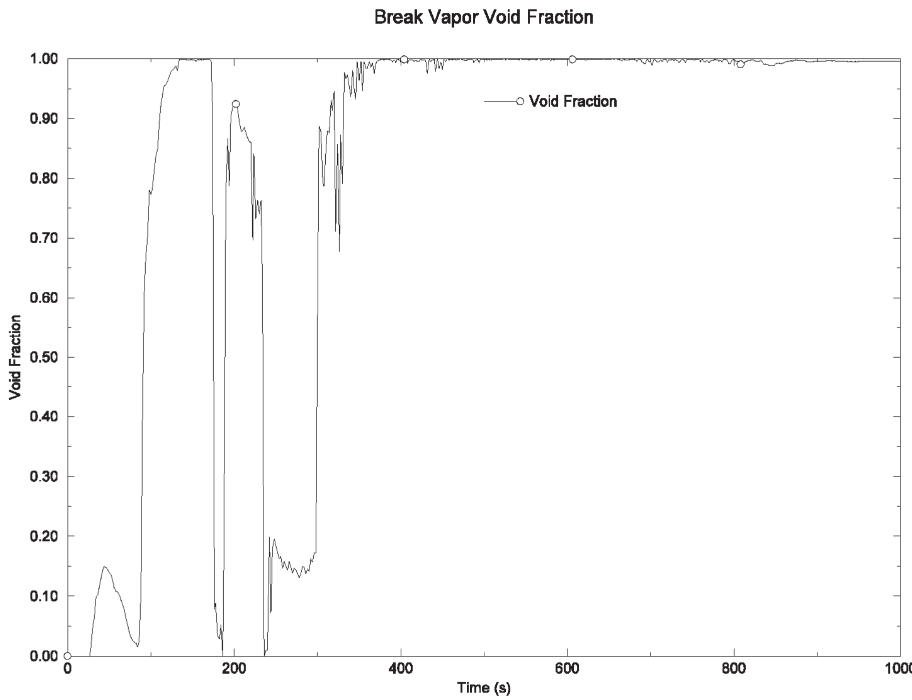


Figure 4-6
Loop Seal Upside Collapsed Levels – 9.10 inch Break

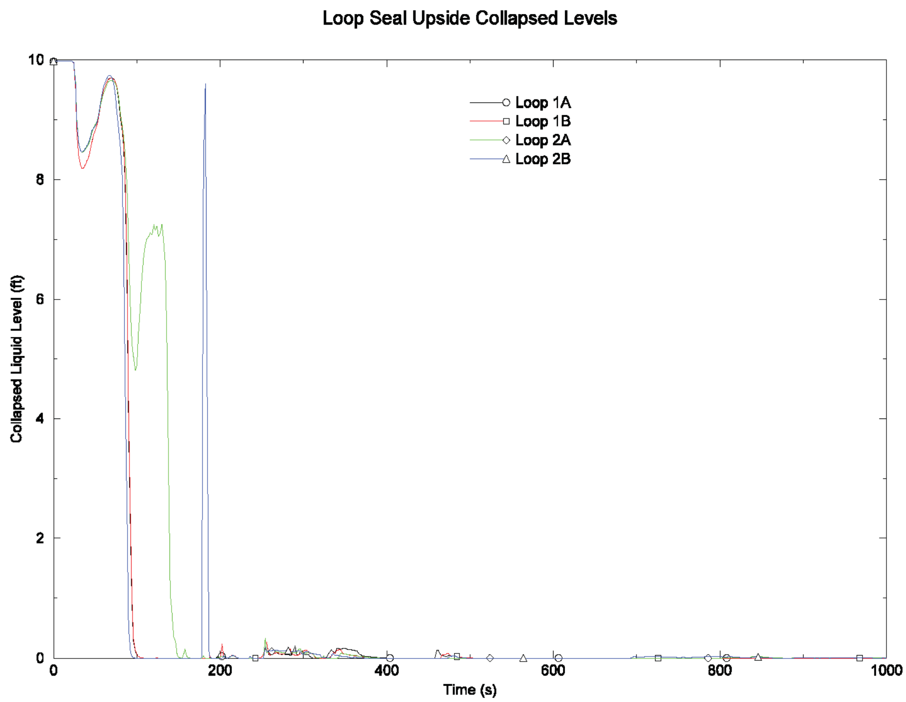


Figure 4-7
Total Core Inlet Mass Flow Rate – 9.10 inch Break

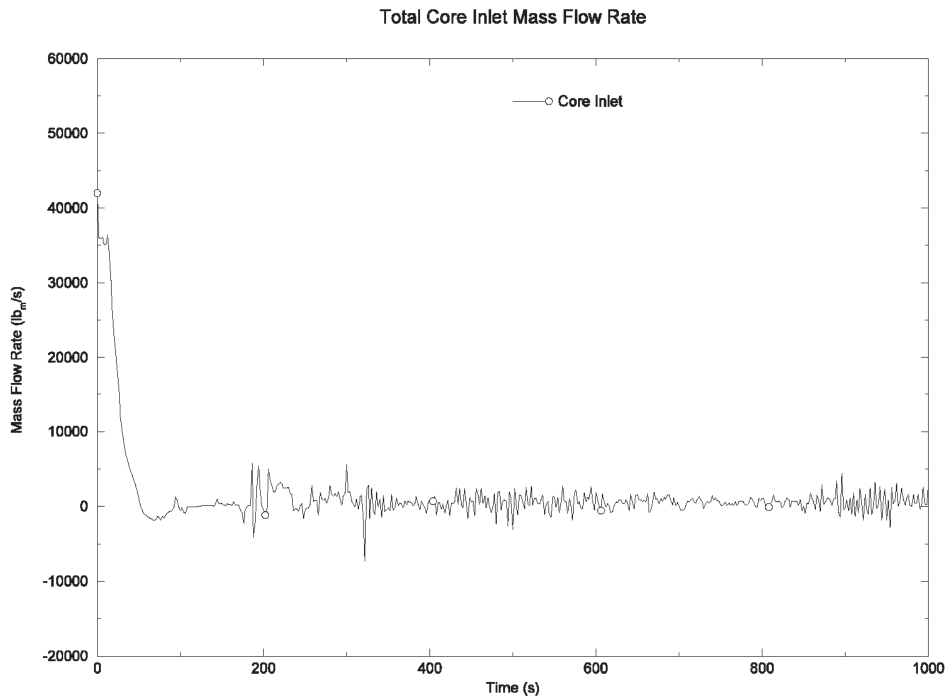


Figure 4-8
Downcomer Collapsed Liquid Level – 9.10 inch Break

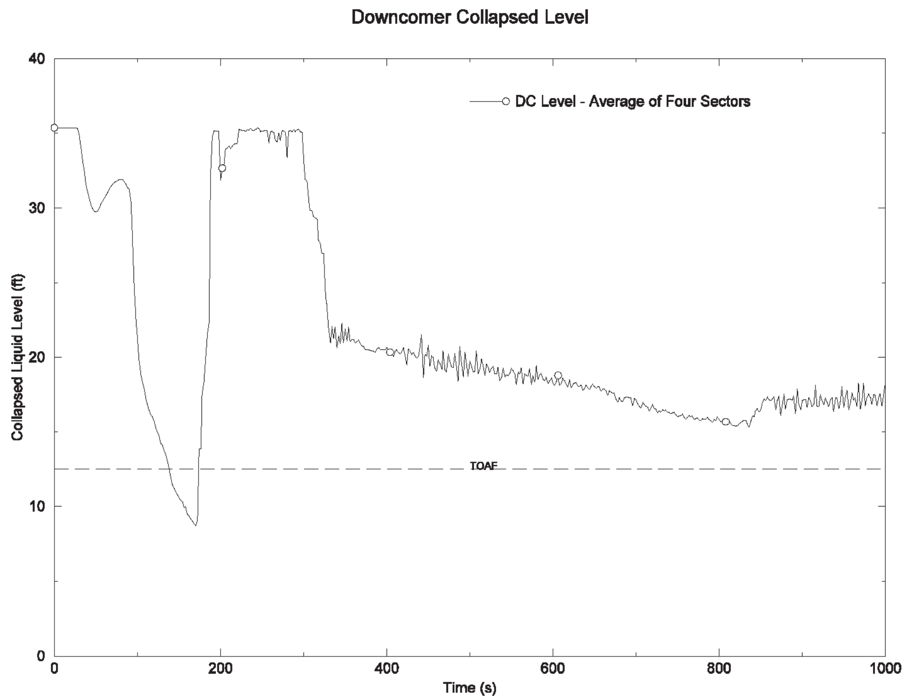


Figure 4-9
Primary System Masses – 9.10 inch Break

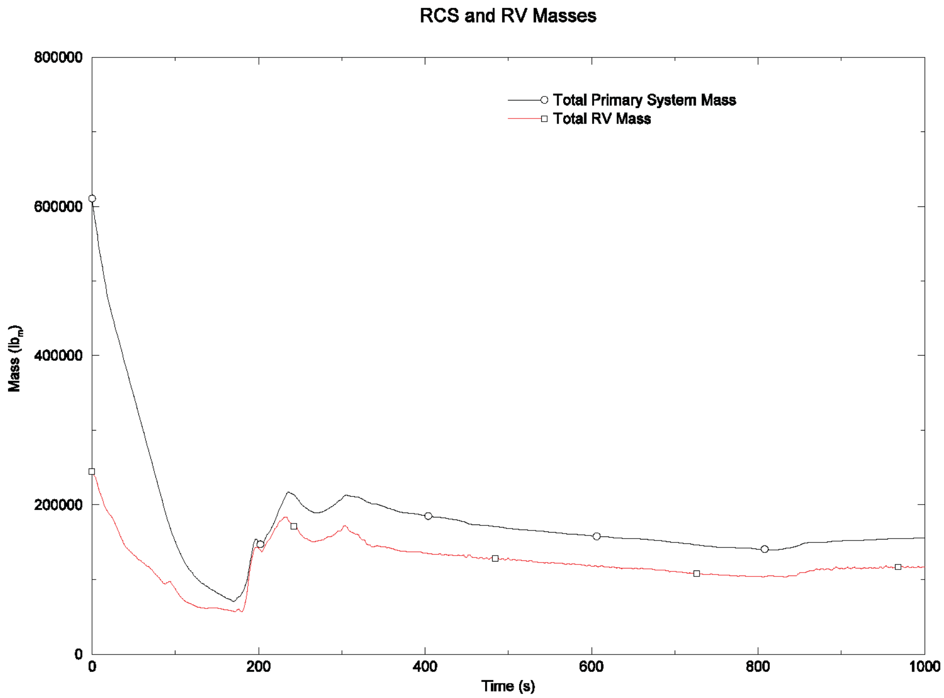


Figure 4-10
RCS Loop Mass Flow Rates – 9.10 inch Break

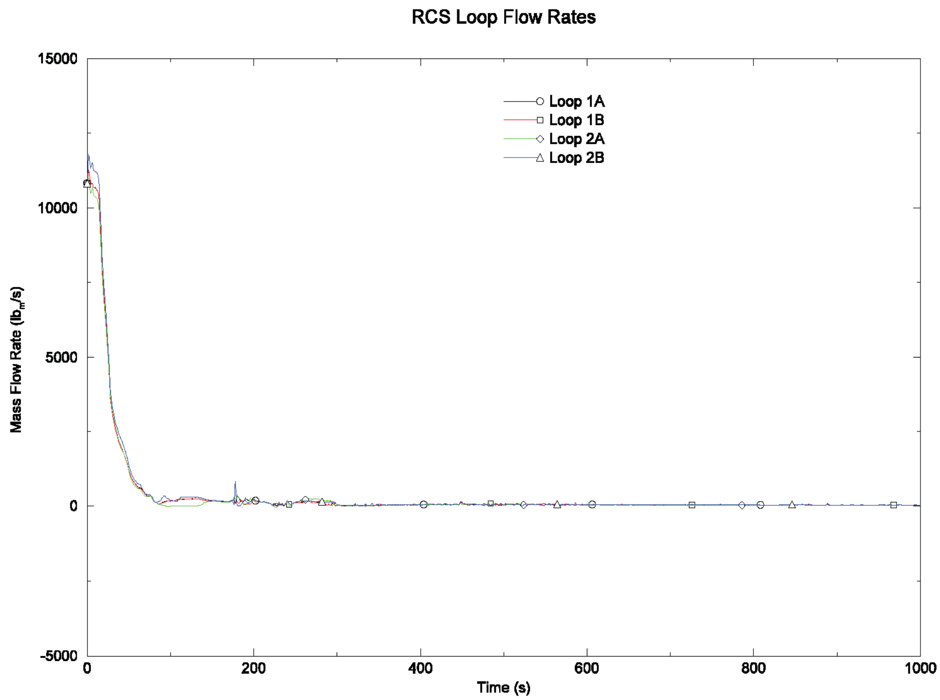


Figure 4-11
Steam Generator Main Feedwater Mass Flow Rates – 9.10 inch Break

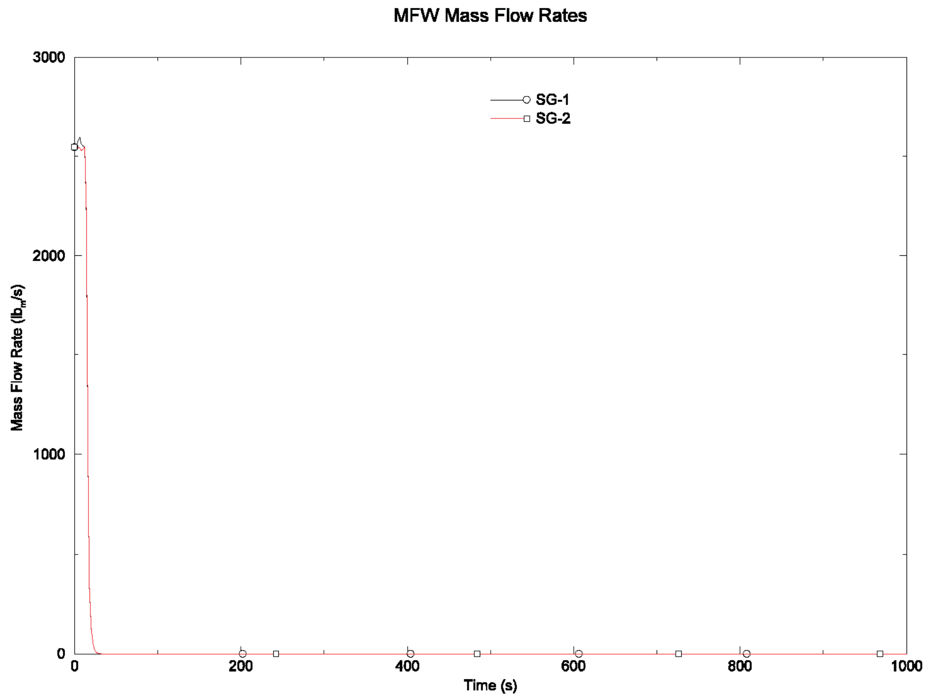


Figure 4-12
SG MSSV Mass Flow Rates – 9.10 inch Break

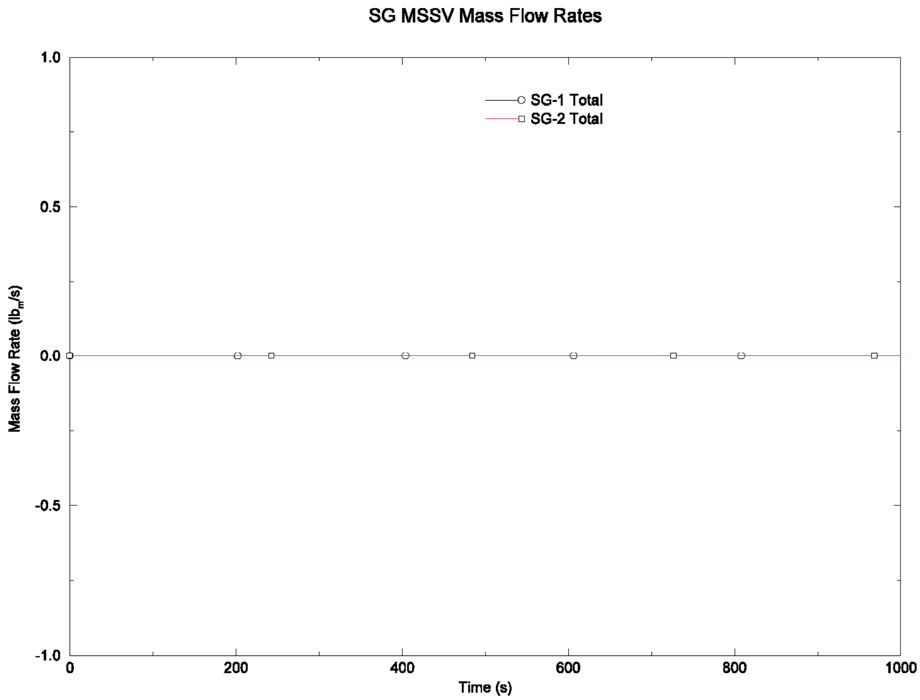


Figure 4-13
Steam Generator Auxiliary Feedwater Mass Flow Rates – 9.10 inch Break

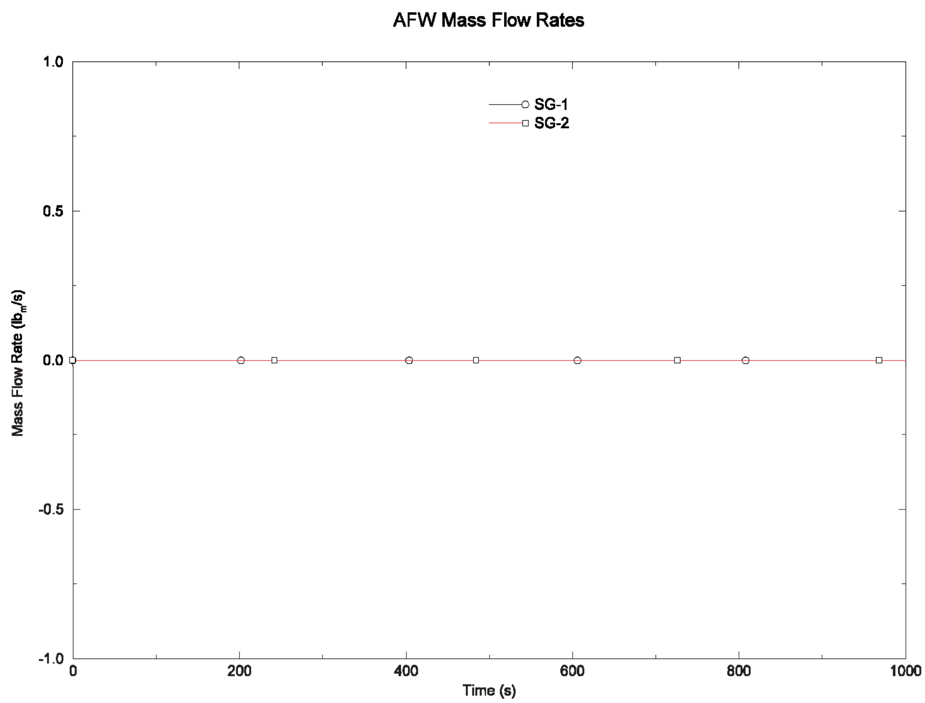


Figure 4-14
Steam Generator Total Mass – 9.10 inch Break

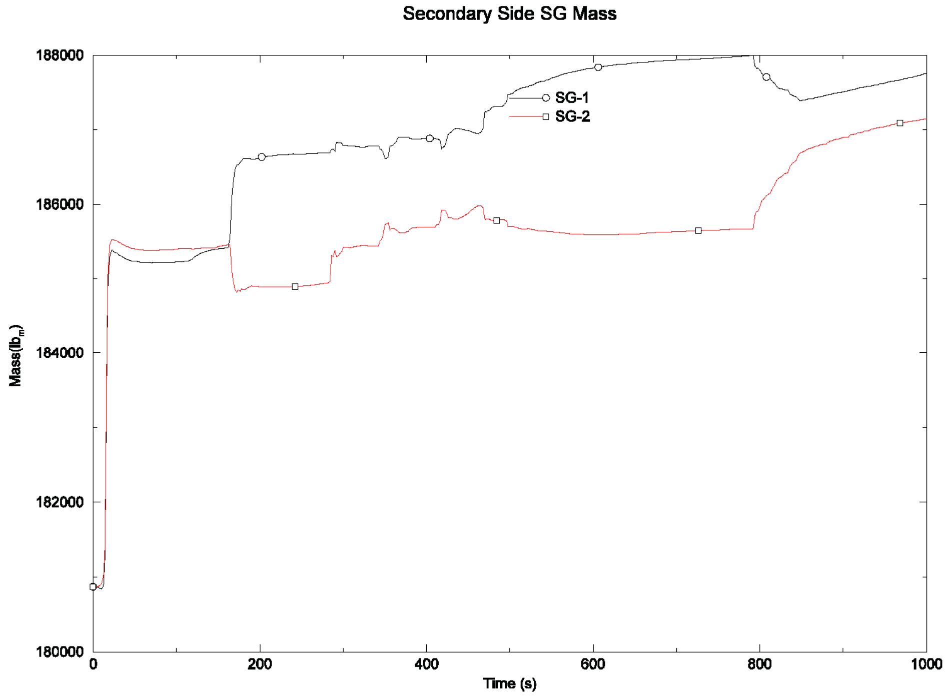


Figure 4-15
Steam Generator Narrow Range Level % – 9.10 inch Break

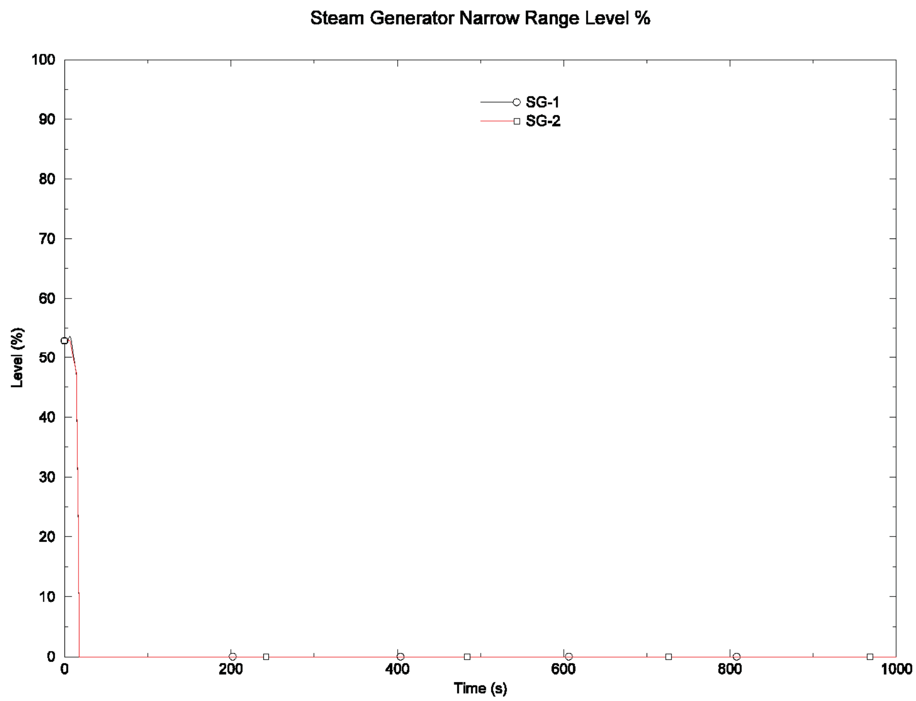


Figure 4-16
Steam Generator Wide Range Level % – 9.10 inch Break

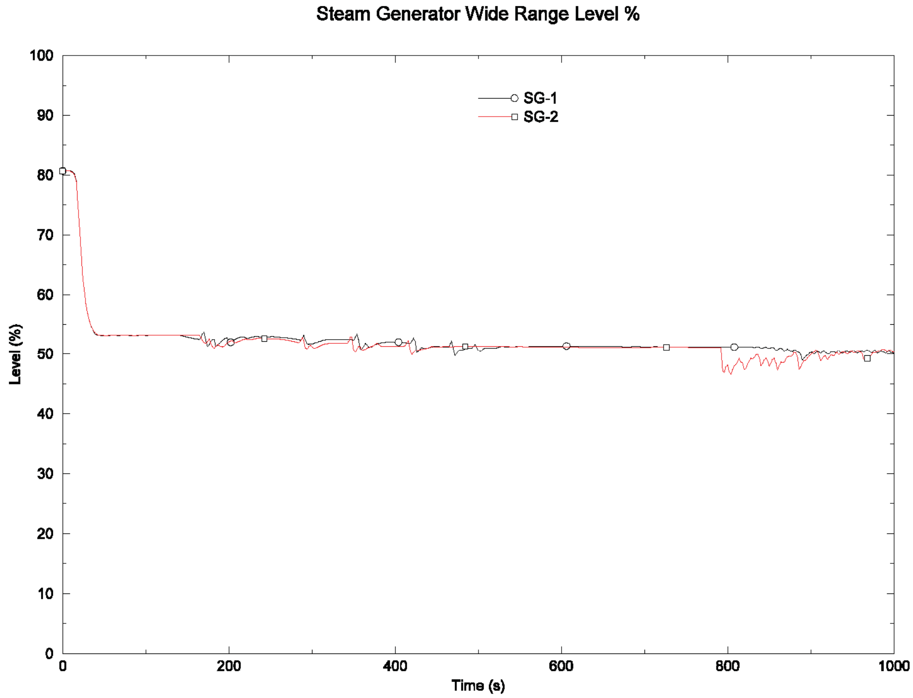


Figure 4-17
High Pressure Safety Injection Mass Flow Rates – 9.10 inch Break

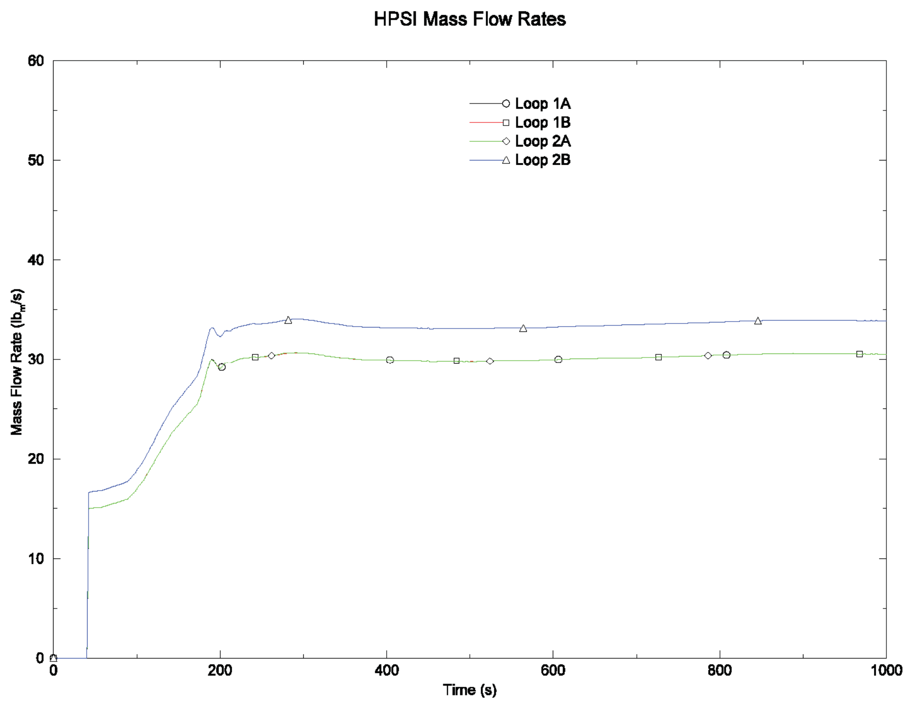


Figure 4-18
Low Pressure Safety Injection Mass Flow Rates – 9.10 inch Break

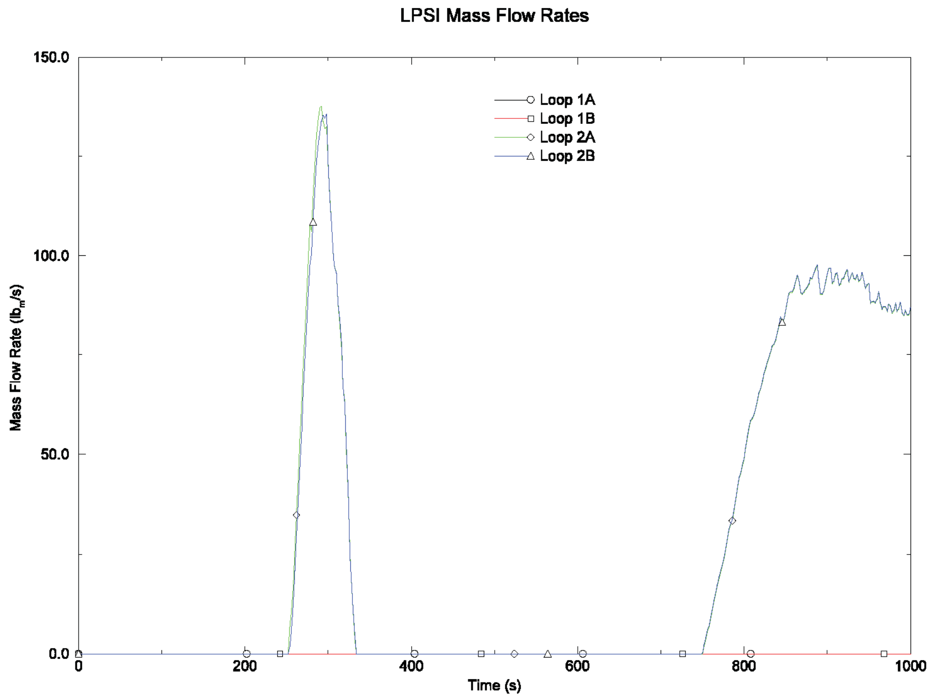


Figure 4-19
Safety Injection Tank Mass Flow Rates – 9.10 inch Break

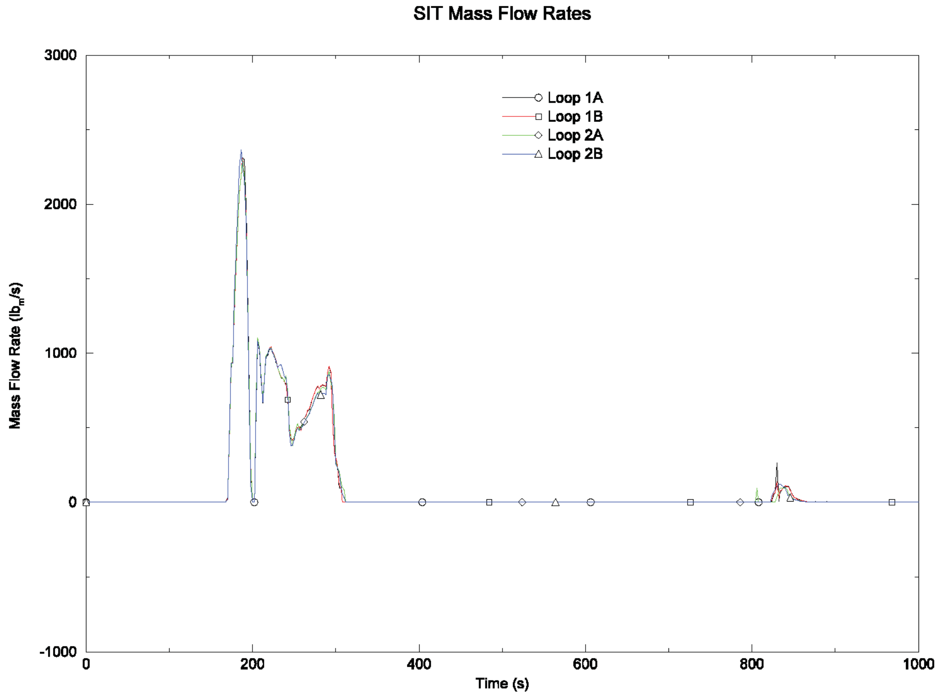


Figure 4-20
Break and ECCS Mass Flow Rates – 9.10 inch Break

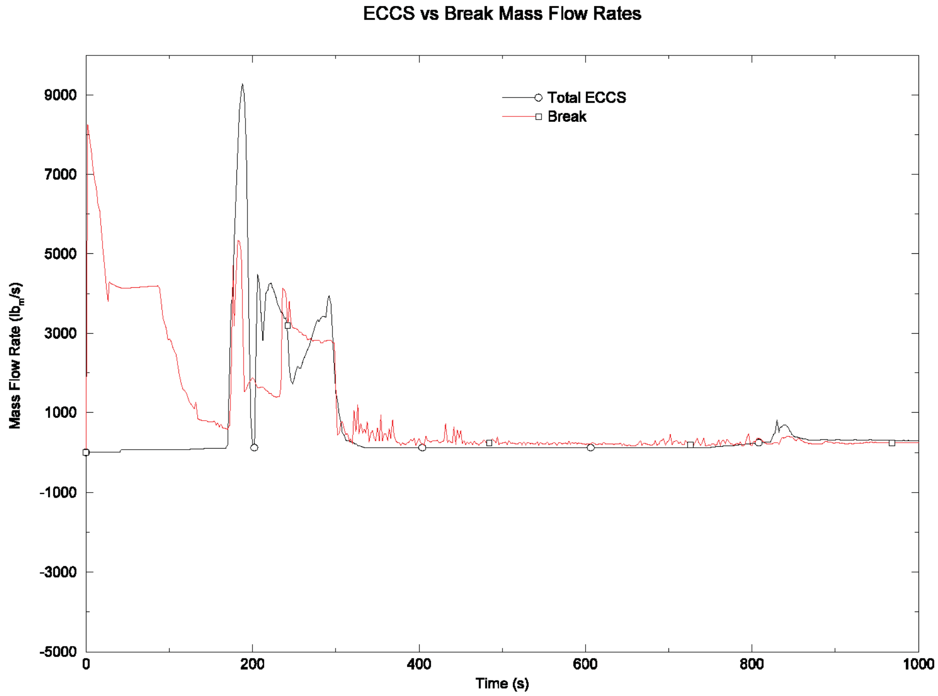


Figure 4-21
Hot Assembly Collapsed Liquid Level – 9.10 inch Break

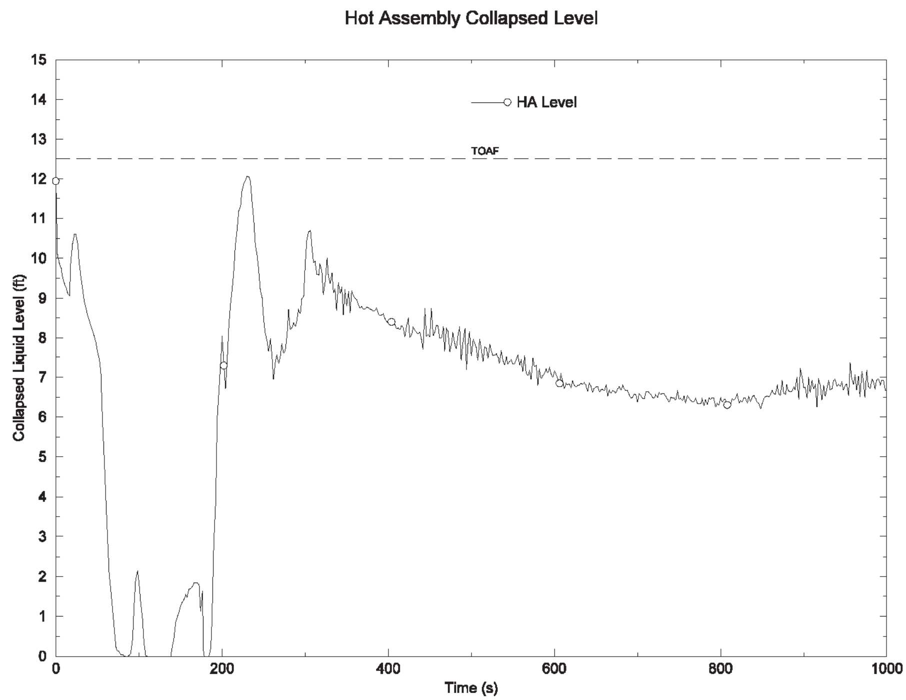


Figure 4-22
Hot Assembly Mixture Level – 9.10 inch Break

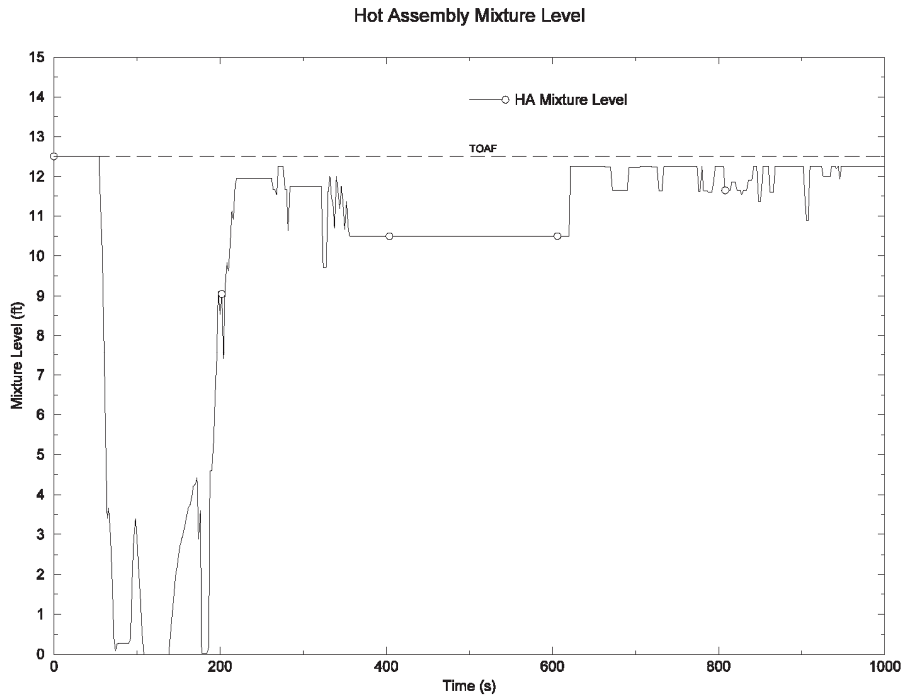


Figure 4-23
Peak Cladding Temperature at PCT Location (11.125 ft) – 9.10 inch Break

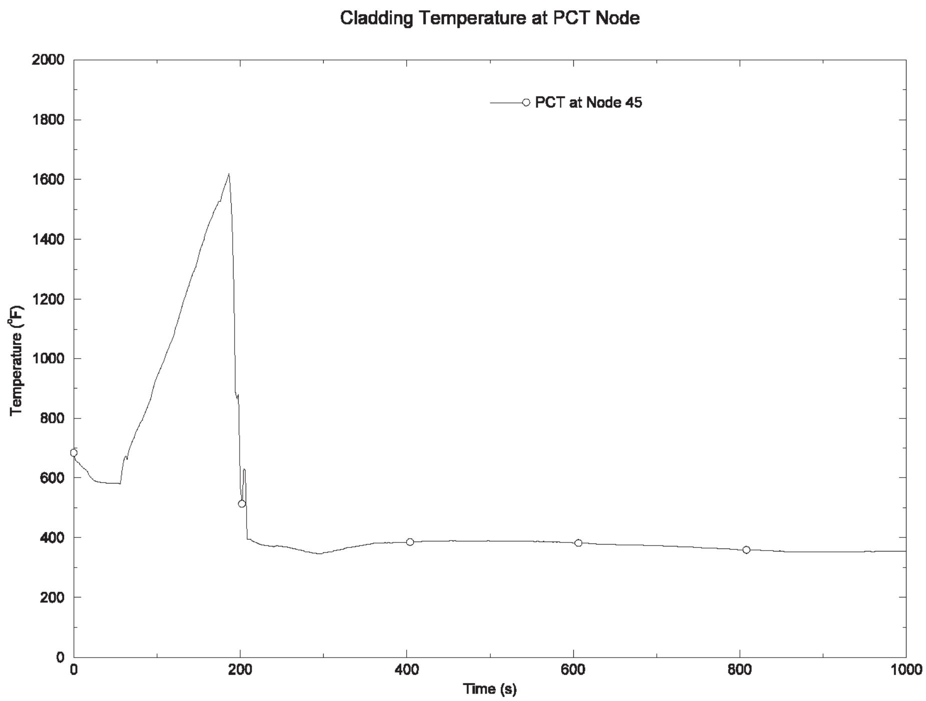


Table 4-3
Palo Verde VQP IOPSV Results

[ ]

Table 4-3 Notes:

(a) There is no significant transient heat up and therefore the PCT is equal to the initialized temperature

Table 4-4
IOPSV Sequence of Events

Table 4-4 Notes:

(a) There is no significant transient heat up and therefore the PCT is equal to the initialized temperature

Figure 4-24
IOPSV Pressurizer Level

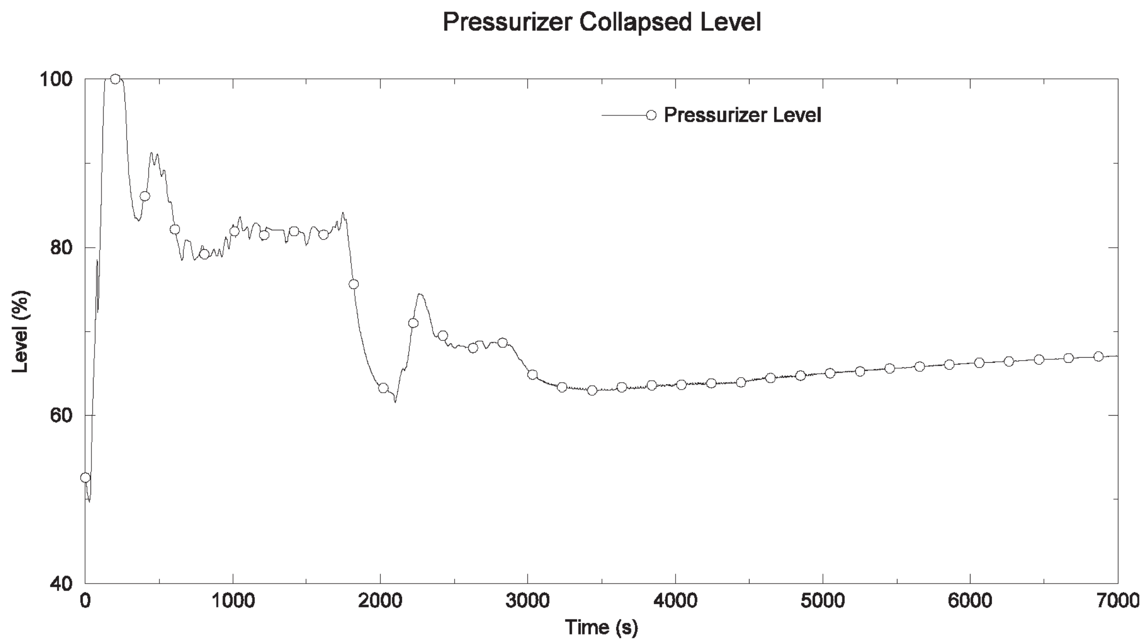


Figure 4-25
IOPSV ECCS and Break Flow Rate

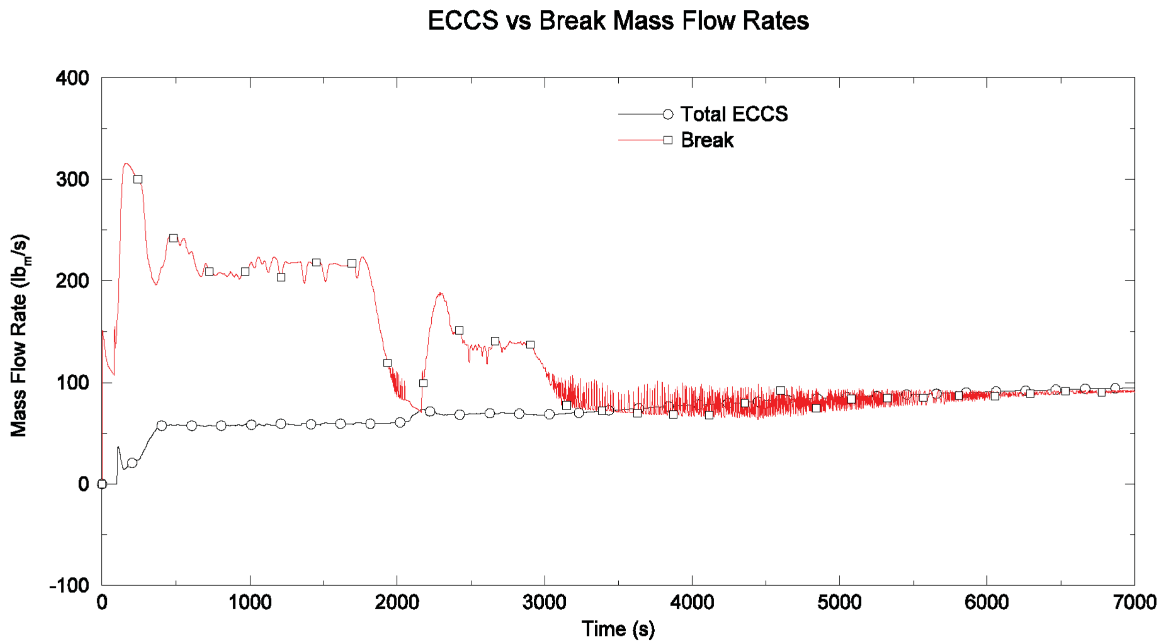


Figure 4-26
IOPSV System Pressure

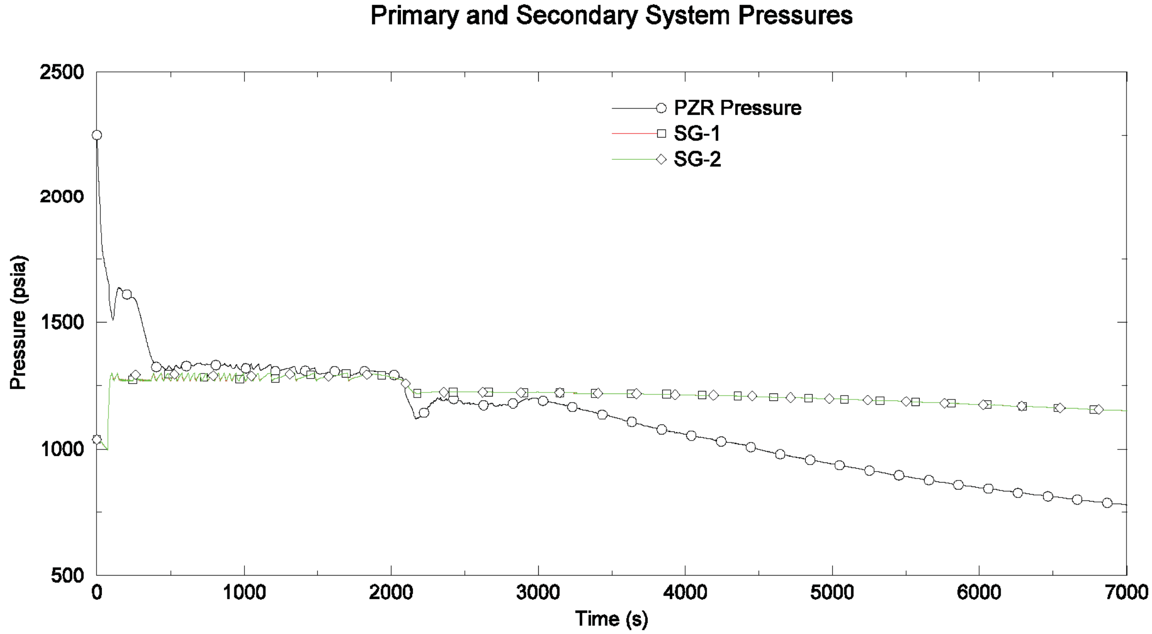


Figure 4-27
IOPSV Main Steam Safety Valve Mass Flow Rate

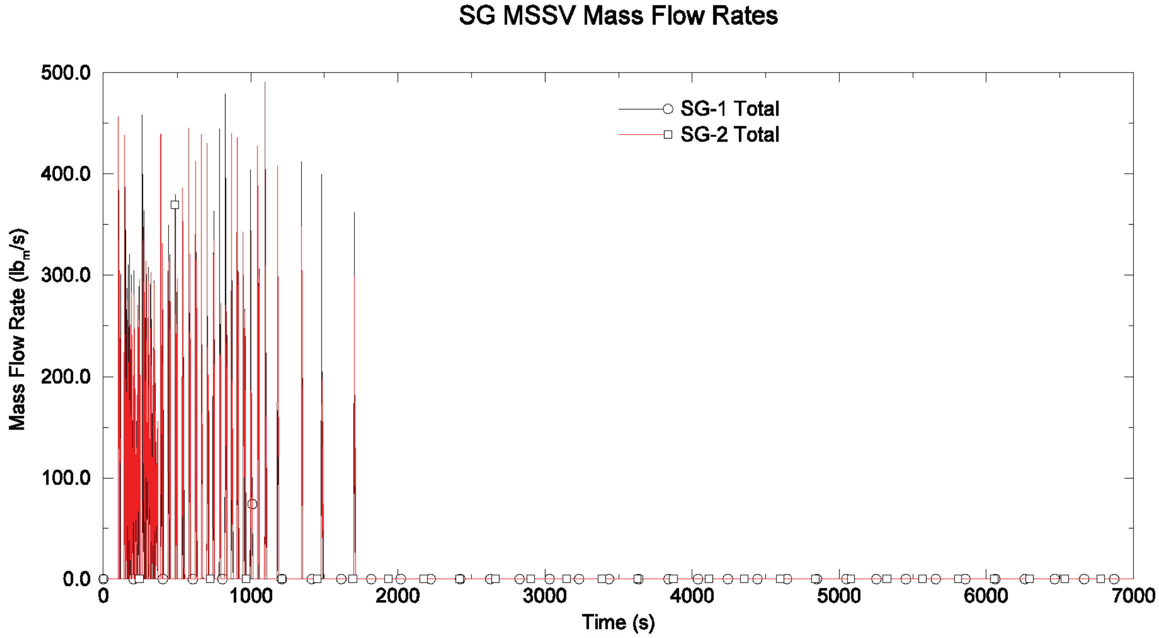


Figure 4-28
IOPSV Main Feedwater Mass Flow Rate

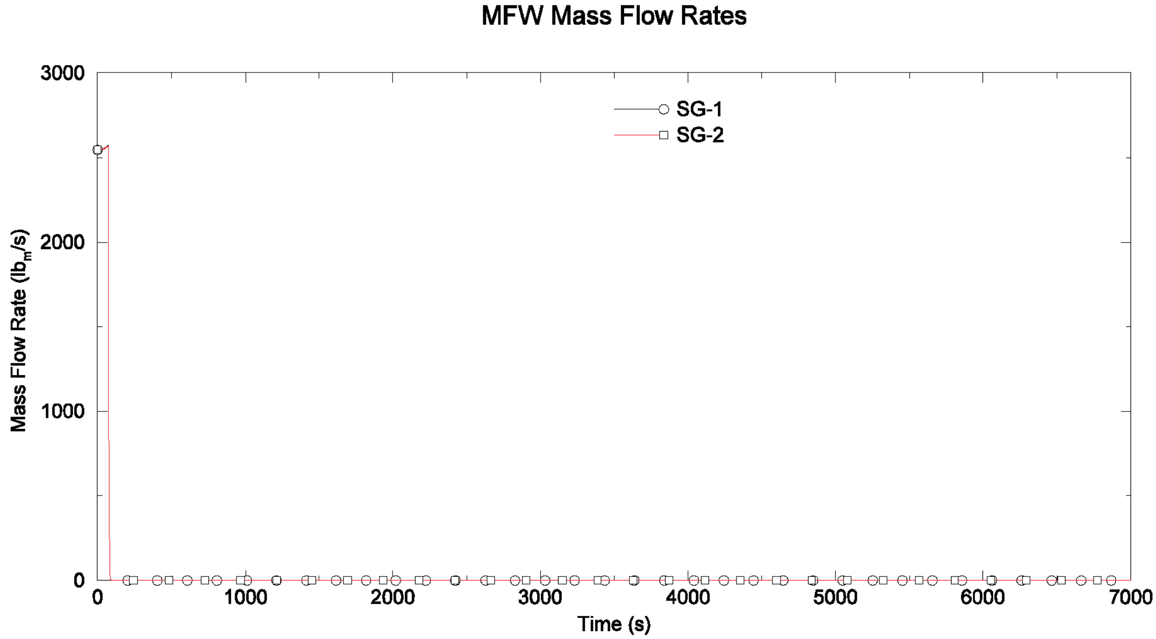


Figure 4-29
IOPSV Core Hot Spot Heat Transfer Coefficient

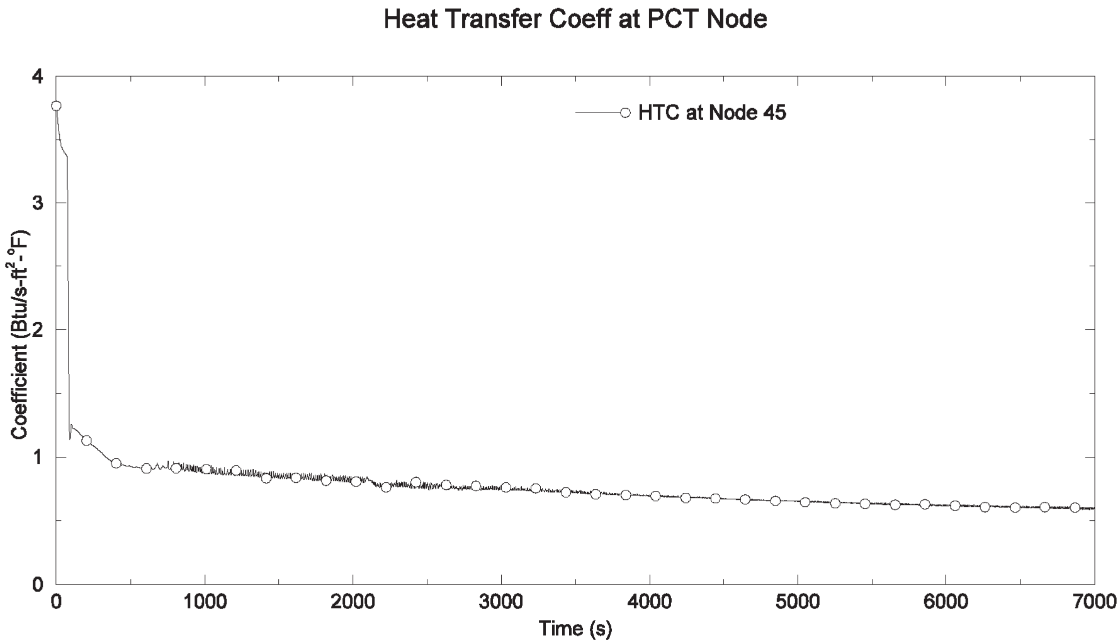


Figure 4-30
IOPSV Auxiliary Feedwater Mass Flow Rate

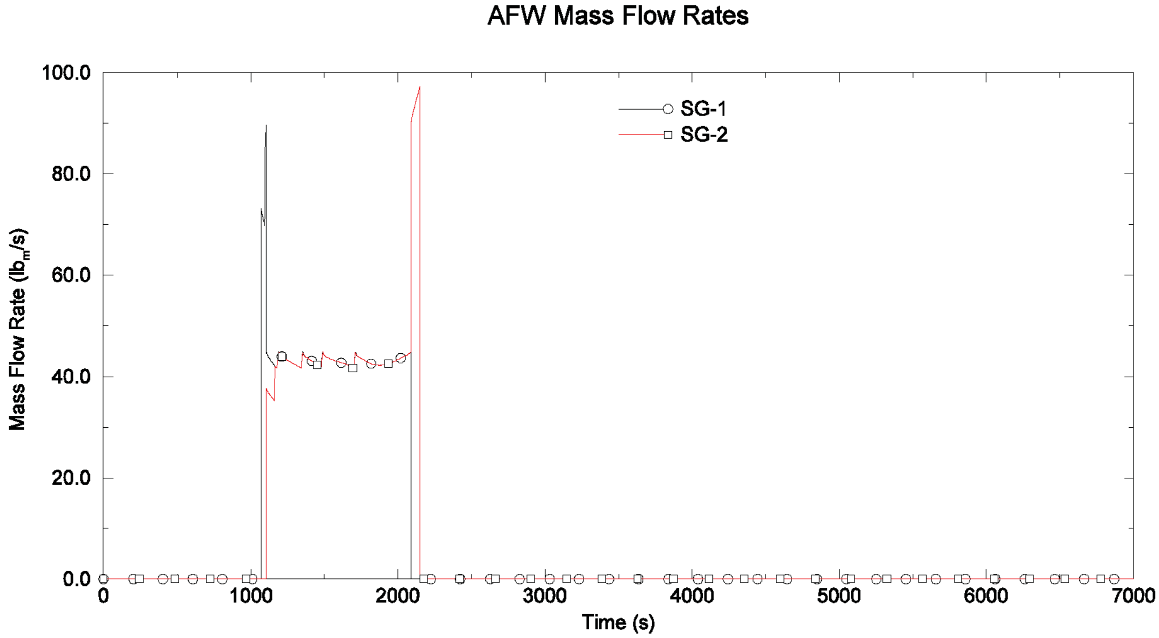


Figure 4-31
IOPSV Steam Generator Wide Range Level

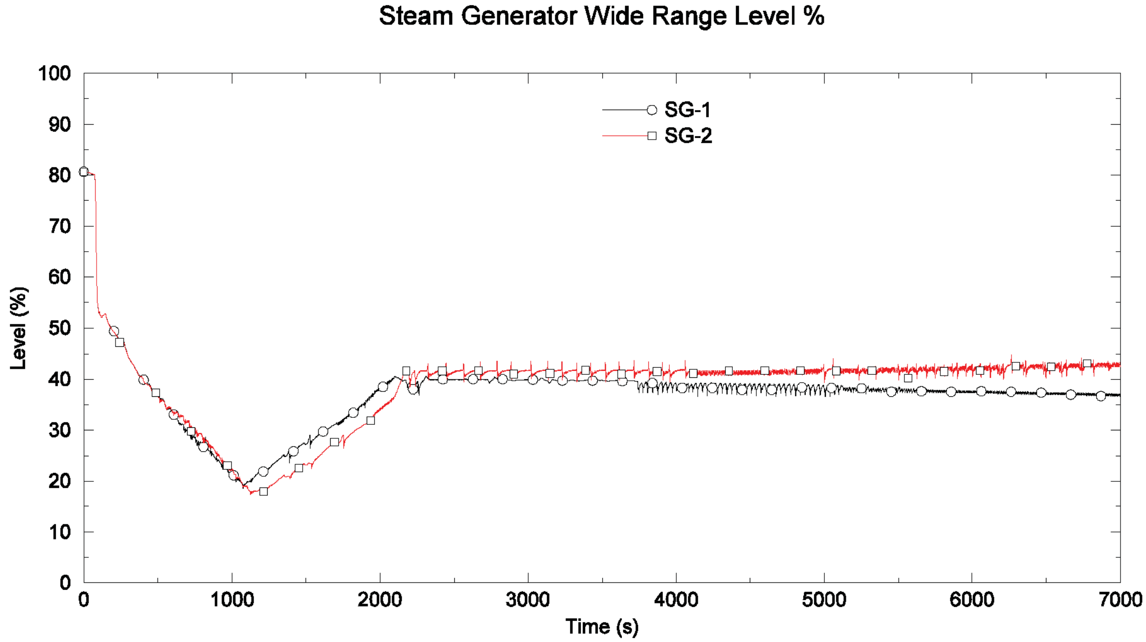


Figure 4-32
IOPSV Reactor Power

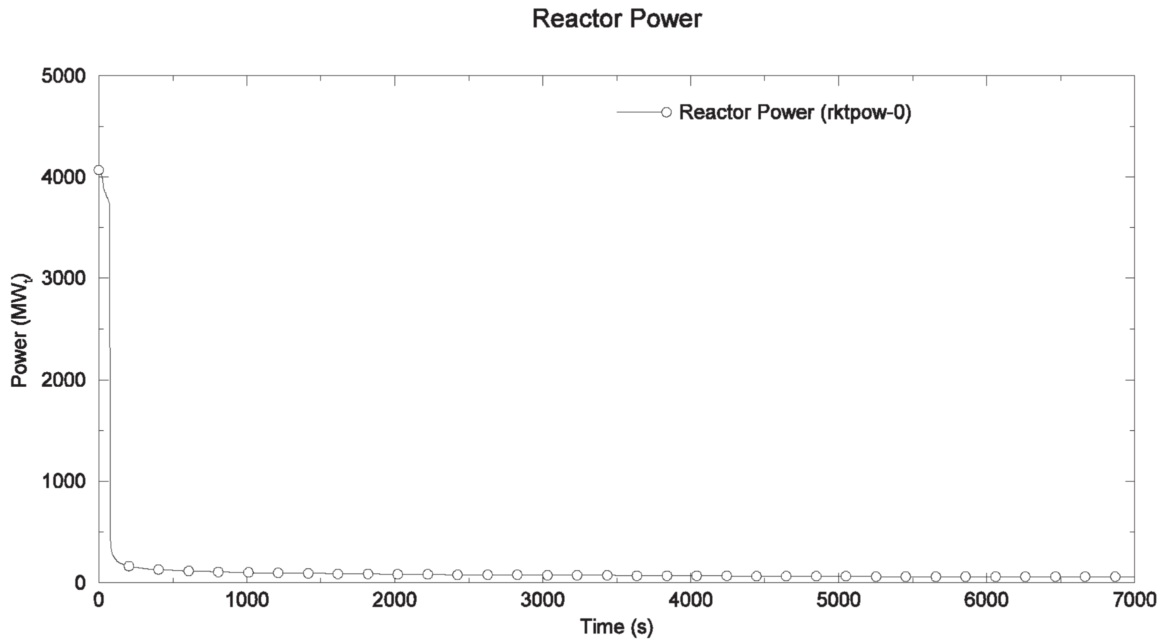


Figure 4-33
IOPSV Primary Coolant System and Reactor Vessel Mass

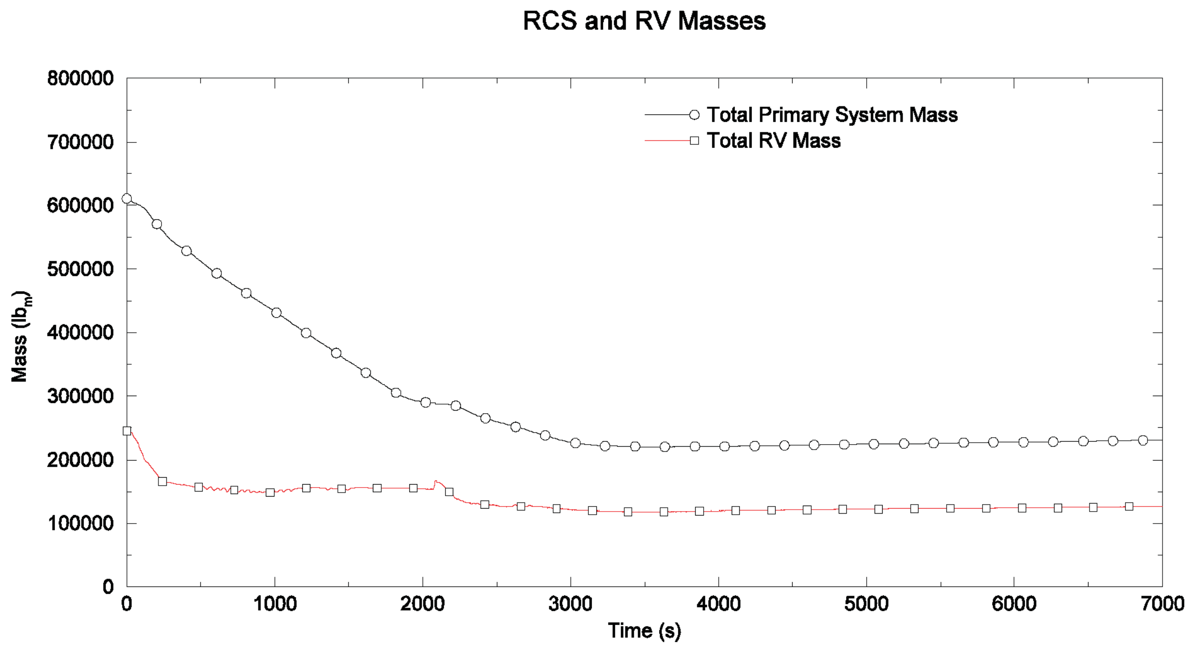


Figure 4-34
IOPSV Loop Seal Upside Level

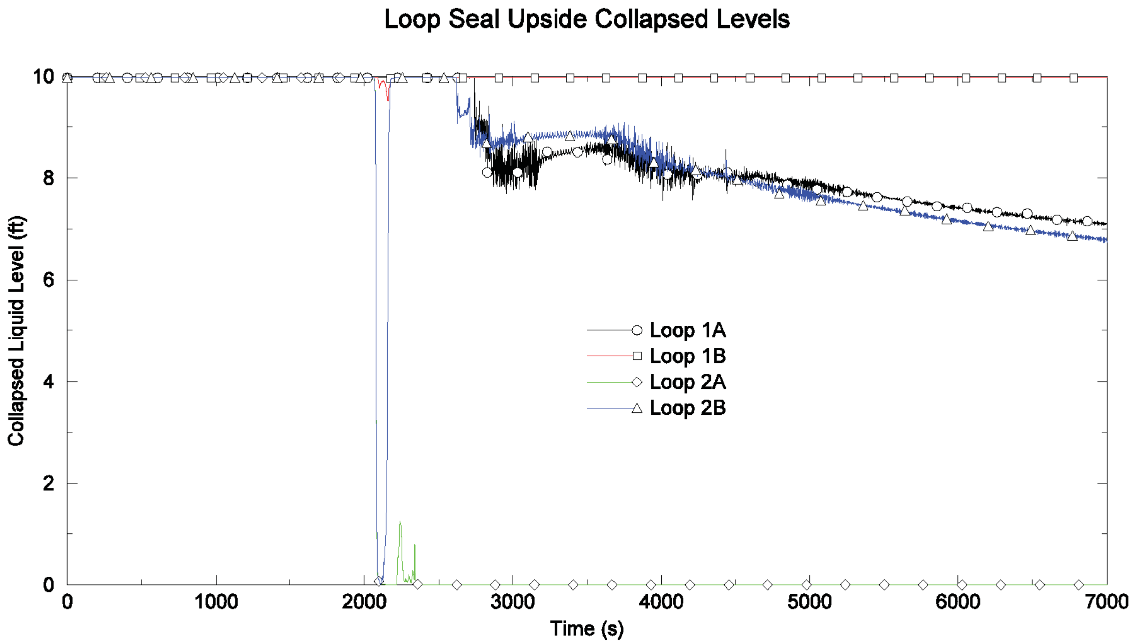


Figure 4-35
IOPSV Hot Assembly Collapsed Liquid Level

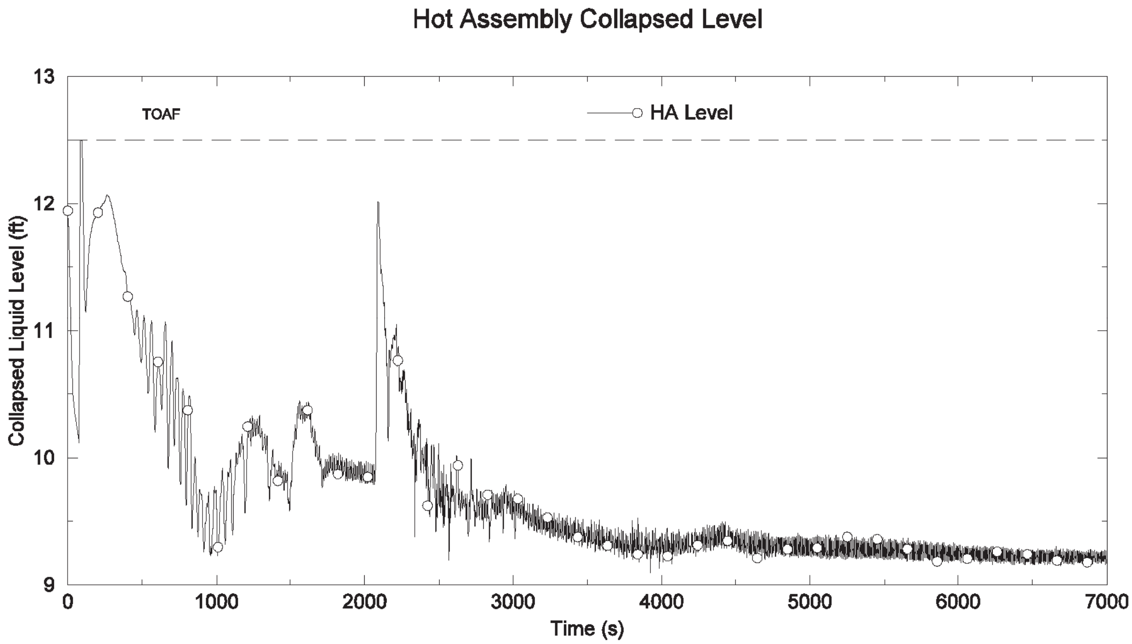


Figure 4-36
IOPSV Hot Assembly Two-Phase Mixture Level

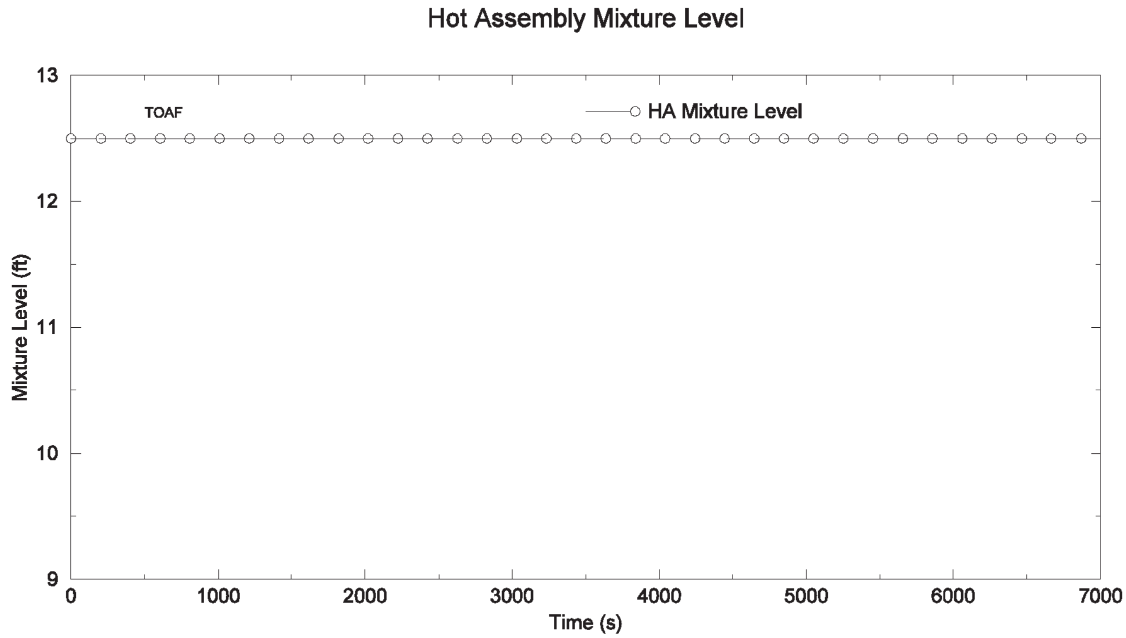


Figure 4-37
IOPSV Fluid Temperature at Core Hot Spot

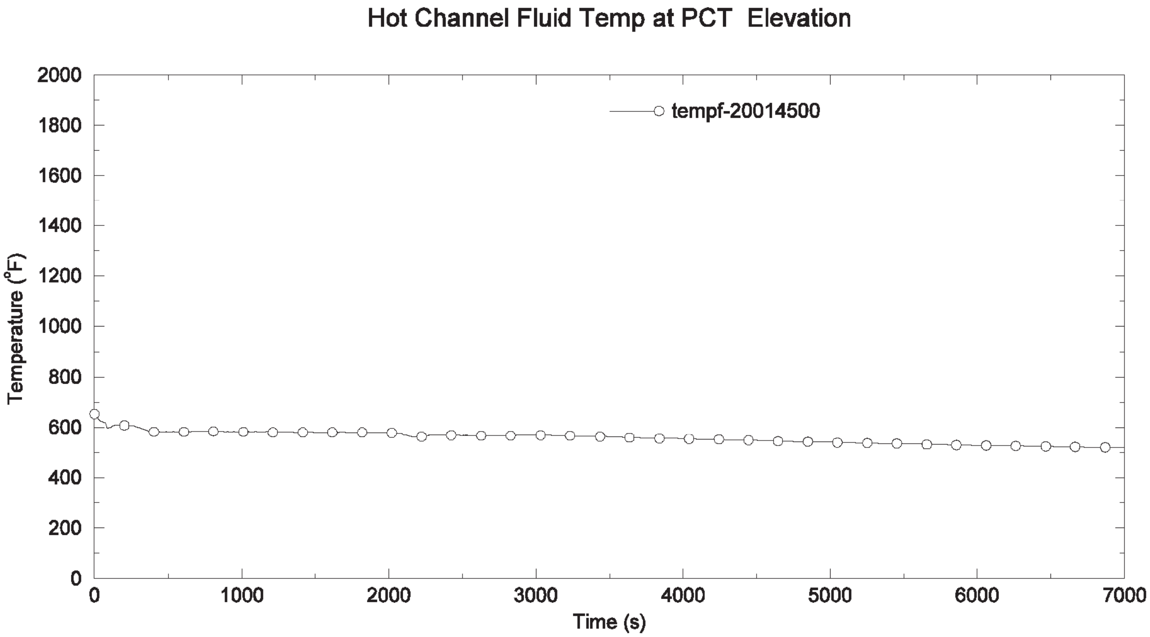
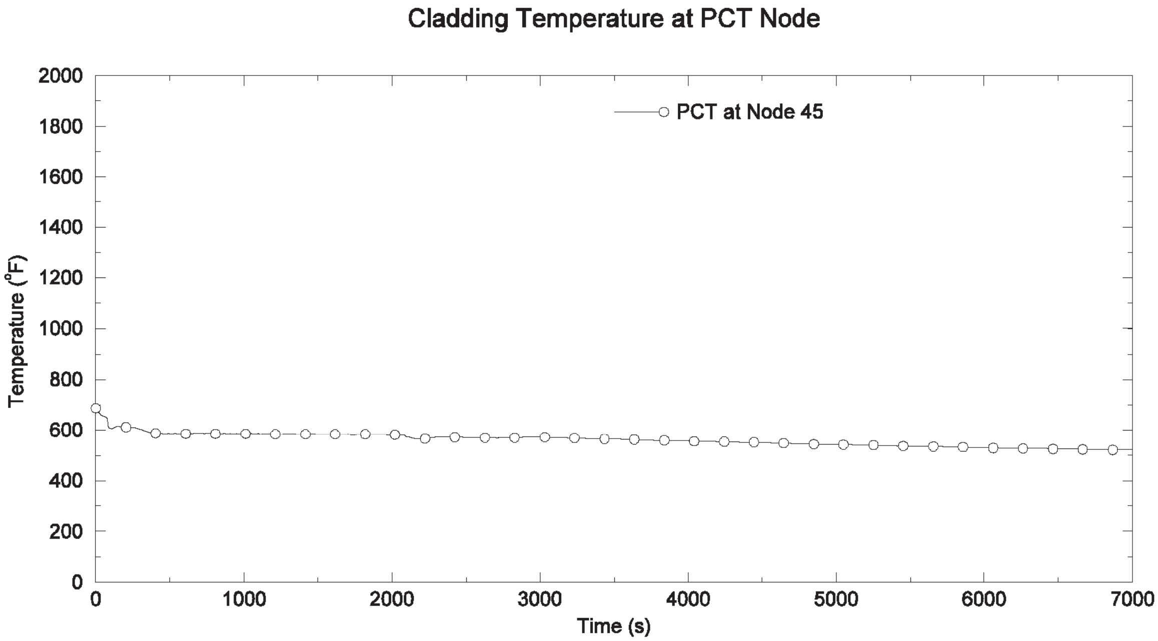


Figure 4-38
IOPSV Clad Temperature at Core Hot Spot



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3. Framatome Inc. Topical Report BAW-10240(P)(A) Revision 0, *Incorporation of M5™ Properties in Framatome ANP Approved Methods*, May 2004.

4. Code of Federal Regulations, Title 10, Part 50, Section 46, *Acceptance Criteria For Emergency Core Cooling Systems For Light-Water Nuclear Power Reactors*, August 2007.

5. Framatome Inc. Topical Report XN-NF-81-58(P)(A) Revision 2, Supplements 1 and 2, *RODEX2 FUEL Thermal-Mechanical Response Evaluation Model*, Exxon Nuclear Company, March 1984.

6. Framatome Inc. Topical Report ANF-81-58(P)(A) Revision 2, Supplements 3 and 4, *RODEX2 FUEL Thermal-Mechanical Response Evaluation Model*, Advanced Nuclear Fuels Corporation, June 1990.

7. Code of Federal Regulations, Title 10, Part 50, Appendix K, *ECCS Evaluation Models*, June 2000.