10 CFR 50.90

aps

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102-08841-TAH/KJG
August 28, 2024

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

Subject: **Palo Verde Nuclear Generating Station Units 1, 2, and 3**
Docket Nos. STN 50-528, 59-529, and 50-530
Renewed Operating License Number NPF-41, NPF-51, and NPF-74
License Amendment Request to Revise the Technical Specifications 3.5.1 and 3.5.2 Safety Injection Tank Pressure Bands, and to Use GOTHIC Code

Pursuant to 10 CFR 50.90, Arizona Public Service Company (APS) is submitting a request for an amendment to the Technical Specifications (TS) for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3.

The proposed changes revise TS Section 3.5.1, “Safety Injection Tanks (SITs) - Operating” and TS Section 3.5.2, “Safety Injection Tanks (SITs) - Shutdown” and their bases. Specifically, the proposed TS changes revise Surveillance Requirement (SR) 3.5.1.3 and SR 3.5.2.3 to increase the upper limit of their SIT pressure bands, and to list their pressure requirements in units of pounds per square inch absolute (psia) as reflected in the PVNGS safety analyses, with no instrument uncertainties included, instead of the SIT instrument units of pounds per square inch gauge (psig) with instrument uncertainties included.

The proposed changes also include use of the GOTHIC code as part of the methodology to perform calculations of the containment pressure and temperature response to various postulated pipe breaks.

The enclosure to this letter provides a description of the proposed changes, a technical evaluation, a regulatory evaluation including a no significant hazards consideration, and an environmental assessment. The enclosure is supported by eight attachments.

Attachment 1 of the enclosure provides marked-up existing TS pages. Attachment 2 of the enclosure provides revised (clean) TS pages. Attachment 3 of the enclosure provides marked-up TS Bases pages to show the conforming changes for information only.

Attachment 4 of the enclosure contains an affidavit signed by Framatome that sets forth the basis on which the proprietary information in Attachment 8 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 8 or the supporting Framatome affidavit should be addressed to Morris Byram of Framatome.

Attachment 5 of the enclosure provides a non-proprietary benchmark evaluation for use of the GOTHIC code as part of the methodology to perform calculations of the containment pressure and temperature response to various postulated pipe breaks.

Attachment 6 of the enclosure provides a non-proprietary technical analysis of the proposed SR 3.5.1.3 and SR 3.5.2.3 changes to increase the upper limit of their SIT pressure bands and to list their pressure requirements in units of psia as reflected in the PVNGS safety analyses, with no instrument uncertainties included, instead of the SIT instrument units of psig with instrument uncertainties included.

Attachment 7 of the enclosure provides a non-proprietary version of the Framatome licensing report for the large break loss of coolant accident (LOCA) analysis supporting this license amendment request. Attachment 8 of the enclosure provides a proprietary version of this report, which contains information proprietary to Framatome.

A pre-submittal meeting for this amendment request was held between APS and the NRC staff on June 28, 2024 (Agency Document Access and Management System [ADAMS] Accession Number ML24180A028). Approval of the proposed amendment is requested by June 1, 2025. Once approved, the amendment shall be implemented within 90 days.

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board has reviewed and approved this license amendment request. By copy of this letter, the amendment is being forwarded to the Arizona Department of Health Services - Bureau of Radiation Control for information.

No new commitments are being made to the NRC by this letter.

Should you need further information regarding this letter, please contact Michael D. Dilorenzo, Licensing Department Leader, at (623) 393-3495.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge.

Executed on August 28, 2024
(Date)

Sincerely,

Horton, Todd (Z10098)

Digitally signed by Horton, Todd (Z10098)
Date: 2024.08.28 14:22:48 -07'00'

TAH/KJG/cry

Enclosure: Description and Assessment of Proposed License Amendment

cc:	J. D. Monninger	NRC Region IV Regional Administrator
	W. T. Orders	NRC NRR Project Manager for PVNGS
	N. Cuevas	NRC Acting Senior Resident Inspector for PVNGS
	B. D. Goretzki	Arizona Department of Health Services – Bureau of Radiation Control

ENCLOSURE

Description and Assessment of Proposed License Amendment

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ATTACHMENT 8 – Framatome Licensing Report for Large Break LOCA Analysis [PROPRIETARY VERSION]

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Arizona Public Service Company (APS) is submitting a License Amendment Request (LAR) to revise the Technical Specifications (TS) for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3.

The proposed changes revise TS Section 3.5.1, “Safety Injection Tanks (SITs) - Operating” and TS Section 3.5.2, “Safety Injection Tanks (SITs) - Shutdown” and their bases. Specifically, the proposed TS changes revise Surveillance Requirement (SR) 3.5.1.3 and SR 3.5.2.3 to increase the upper limit of their SIT pressure bands, and to list their pressure requirements in units of pounds per square inch absolute (psia) as reflected in the PVNGS safety analyses, with no instrument uncertainties included, instead of the SIT instrument units of pounds per square inch gauge (psig) with instrument uncertainties included.

PVNGS currently uses the Bechtel Containment Pressure and Temperature Transient Analysis (COPATTA) code as part of the methodology for the calculation of the containment pressure and temperature response to various postulated pipe breaks. The proposed changes also include use of the industry standard Generation of Thermal Hydraulic Information for Containments (GOTHIC) code as a replacement to COPATTA as part of the methodology to perform calculations of the containment pressure and temperature response to various postulated pipe breaks, including determination of the response to the increase in the upper limit of the SIT pressure bands.

This TS change and methodology change are planned to be implemented within 90 days of license amendment approval.

2.0 BACKGROUND

2.1. PVNGS SIT Functions

The functions of the four SITs are to supply water to the reactor vessel during the blowdown phase of a Loss of Coolant Accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA. The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

The refill phase of a LOCA follows immediately where reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The balance of the SITs inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

The SITs are passive components partially filled with borated water and pressurized with nitrogen to facilitate injection into the reactor vessel. No operator or control action is required for the SITs to perform their function. Internal tank pressure is sufficient to discharge the contents to the RCS, if RCS pressure decreases below the SIT pressure.

Each SIT is piped into one RCS cold leg via the injection lines utilized by the High Pressure

Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) Systems. Each SIT is capable of being isolated from the RCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves are normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident.

The SIT gas and water volumes, gas pressure, and outlet pipe size are selected to allow one less than the required 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 limiting LOCA assumption during cold leg breaks, the entire contents of one SIT will be lost via the break during the blowdown phase of a LOCA.

TS Limiting Condition for Operation (LCO) 3.5.1 requires four SITs to be OPERABLE to ensure that the required contents of three of the SITs will reach the core during a LOCA. If the contents of fewer than three tanks are injected during the blowdown phase of a LOCA, then the Emergency Core Cooling System (ECCS) performance acceptance criteria of 10 CFR 50.46 could be violated. LCO 3.5.2 establishes the minimum conditions required to ensure that the required SITs are available to accomplish their core cooling safety function following a LOCA.

3.0 DETAILED DESCRIPTION

3.1. Description of Proposed Changes

3.1.1. TS 3.5.1 - Safety Injection Tanks (SITs) – Operating

Technical Specification 3.5.1 addresses an LCO and related SRs for the SITs during Operating Conditions. This TS section is applicable in MODES 1 and 2, and MODES 3 and 4 with pressurizer pressure equal to or greater than 1837 psia.

SR 3.5.1.3 requires verification that the nitrogen cover pressure in each required SIT is "≥ 600 psig and ≤ 625 psig." The proposed change revises the SR 3.5.1.3 SIT pressure band in each required SIT to "≥ 602 psia and ≤ 675 psia."

This change to SR 3.5.1.3 is an increase in the upper limit of its SIT pressure band. The new upper pressure limit of 675 psia remains within the SIT design pressure of 700 psig (714.2 psia). This change also lists the pressure requirements in units of psia as reflected in the PVNGS safety analyses, with no instrument uncertainties included, instead of the SIT instrument units of psig with instrument uncertainties included. The accounting for instrument uncertainties will continue as part of surveillance procedures.

A mark-up of the affected TS 3.5.1 page is provided in Attachment 1 to this enclosure. A revised (clean) TS 3.5.1 page is provided in Attachment 2 to this enclosure. A mark-up of the affected TS 3.5.1 Bases page in support of the TS change is provided for information only in Attachment 3 to this enclosure.

3.1.2. TS 3.5.2 - Safety Injection Tanks (SITs) – Shutdown

Technical Specification 3.5.2 addresses an LCO and related SRs for the SITs during Shutdown Conditions. This TS section is applicable in MODES 3 and 4 with pressurizer pressure less than 1837 psia.

SR 3.5.2.3 requires verification that the nitrogen cover pressure in each required SIT is "≥ 260 psig and ≤ 625 psig". The proposed change revises the SR 3.5.2.3 SIT pressure band in each required SIT to "≥ 250 psia and ≤ 675 psia".

This change to SR 3.5.2.3 is an increase in the upper limit of its SIT pressure band to provide consistency with the change to SR 3.5.1.3. The new upper pressure limit of 675 psia remains within the SIT design pressure of 700 psig (714.2 psia). This change also lists the pressure requirements in units of psia as reflected in the PVNGS safety analyses, with no instrument uncertainties included, instead of the SIT instrument units of psig with instrument uncertainties included. The accounting for instrument uncertainties will continue as part of surveillance procedures.

A mark-up of the affected TS 3.5.2 page is provided in Attachment 1 to this enclosure. A revised (clean) TS 3.5.2 page is provided in Attachment 2 to this enclosure. A mark-up of the affected TS 3.5.2 Bases page in support of the TS change is provided for information only in Attachment 3 to this enclosure.

3.1.3. Use of GOTHIC Code

As described in the PVNGS Updated Final Safety Analysis Report (UFSAR), PVNGS currently uses the Bechtel COPATTA code as part of the methodology for the calculation of the containment pressure and temperature response to various postulated pipe breaks. This methodology is not addressed in the PVNGS Technical Specifications.

The proposed changes to the PVNGS licensing basis include use of the GOTHIC code as part of the methodology to perform calculations of the pressure and temperature response to various postulated pipe breaks, including determination of the response to the increase in the upper limit of the SR 3.5.1.3 SIT pressure band.

Attachment 5 to this enclosure presents a benchmark evaluation for use of the GOTHIC code. The benchmark evaluation uses GOTHIC Version 8.4 (Reference 7.1). The GOTHIC code is being continuously maintained and updated to include new features and/or correct problems. Therefore, although the analysis models and methods described in the benchmark evaluation were developed using GOTHIC Version 8.4, APS intends to use future versions of GOTHIC as they become available.

3.2. Reason for Proposed Changes

At the start of an operating cycle, fluctuations in the containment pressure and temperature may increase SIT pressure leading to cycling of the SIT vent valves as they open and close to maintain the SIT pressure within its operating band. If a SIT vent valve does not fully close during this cycling, then the plant is challenged with restoring operability of the SIT vent valve within the completion times required by TS 3.5.1 and TS 3.5.2.

In order to reduce the number of challenges to the SIT vent valves, this proposed TS change increases the upper limit of the SIT pressure band specified in SR 3.5.1.3 of TS 3.5.1, and increases the upper limit of the SIT pressure band specified in SR 3.5.2.3 of TS 3.5.2.

The current SR 3.5.1.3 and SR 3.5.2.3 list the nitrogen cover pressure for each SIT as values in units of psig with instrument uncertainties included. The proposed change will list the SIT nitrogen cover pressure as design values in units of psia with no instrument uncertainties included. Listing the SIT SRs for nitrogen cover pressure in units of psia will match the PVNGS safety analyses that model SIT nitrogen cover pressure in units of psia. The accounting for instrument uncertainties will continue as part of surveillance procedures. All requirements and applicability of the SRs will remain the same.

PVNGS currently uses the Bechtel COPATTA code as part of the methodology for the calculation of the containment pressure and temperature response to various postulated pipe breaks. In 2021, Bechtel elected to discontinue development, maintenance, and licensing of its software products, including the COPATTA code. The Bechtel actions necessitate APS use of a different code to perform containment pressure and temperature response analyses. APS has chosen to use the industry standard code GOTHIC to perform containment pressure and temperature response analyses, including determination of the response to the increase in the upper limit of the SIT pressure bands.

4.0 TECHNICAL EVALUATION

4.1. Benchmark Evaluation for Use of the GOTHIC Code

The proposed changes include use of the industry standard GOTHIC code as a replacement to the COPATTA code as part of the methodology to perform calculations of the containment pressure and temperature response to various postulated pipe breaks. Attachment 5 to this enclosure presents a benchmark evaluation for use of the GOTHIC code.

4.2. Technical Analysis of Changes to the SIT Pressure Bands

The proposed TS changes revise SR 3.5.1.3 and SR 3.5.2.3 to increase the upper limit of their SIT pressure bands. The changes also list pressure requirements in units of psia as reflected in the PVNGS safety analyses, with no instrument uncertainties included, instead of the SIT instrument units of psig with instrument uncertainties included. Attachment 6 to this enclosure presents a technical analysis of the impact of the proposed changes on the large break LOCA transient analyses, the small break LOCA transient analyses, and the post-LOCA long-term cooling analyses. Attachment 6 to this enclosure also presents a technical analysis of the impact of the proposed changes on the post-LOCA containment pressure and temperature response analyses as evaluated with the GOTHIC code. The proposed SR changes have been evaluated to determine whether the applicable regulations and requirements continue to be met.

Title 10 of the Code of Federal Regulations (10 CFR) Paragraph 50.36(c)(2)(ii)(B) requires that TS limiting conditions for operation be established for a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

TS 3.5.1 and TS 3.5.2 help to ensure that the following ECCS performance acceptance criteria established by 10 CFR 50.46 (Reference 7.2) will be met following a Loss of Coolant Accident (LOCA):

Criterion 1: Peak Cladding Temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

Criterion 2: Maximum Cladding Oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

Criterion 3: Maximum Hydrogen Generation. 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.

Criterion 4: Coolable Geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

Criterion 5: Long-Term Cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

TS 3.5.1 and TS 3.5.2 also help to ensure that the following General Design Criteria (GDC) established by 10 CFR 50 Appendix A (Reference 7.3) for containment structure design will be met following a postulated pipe break inside the containment structure:

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

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

Per Attachment 6 to this enclosure, the ECCS performance criteria and containment structure design criteria continue to be met with the increase in the upper limit of the SIT pressure bands.

5.0 REGULATORY EVALUATION

5.1. Precedent

Multiple precedents have been established for including use of the GOTHIC code as part of the methodology to perform calculations of the containment pressure and temperature response to various postulated pipe breaks. These precedents include submittals by Entergy Operations, Inc for Waterford Steam Electric Station, Unit 3, Southern Nuclear Operating Company, Inc for Joseph M. Farley Nuclear Plants, Units 1 and 2, and Luminant Generation Company LLC for Comanche Peak Steam Electric Station, Units 1 and 2.

Waterford Steam Electric Station, Unit 3 has used the GOTHIC code to analyze the limiting large break LOCA and limiting Main Steam Line Break (MSLB) events. Their License Amendment No. 165, dated July 6, 2000 (Reference 7.4), provided approval for use of the GOTHIC code. Benchmarking for this plant showed the capability of GOTHIC in the containment pressure and temperature response calculations.

Joseph M. Farley Nuclear Plants, Units 1 and 2 have used the GOTHIC code to perform containment response analysis in support of their power uprate. Their License Amendment Nos. 137 and 129 for Unit Nos. 1 and 2, respectively, dated April 29, 1998 (Reference 7.5), provided approval for use of the GOTHIC code. Benchmarking for these plants showed the capability of GOTHIC in the containment pressure and temperature response calculations.

Comanche Peak Steam Electric Station, Units 1 and 2 have used the GOTHIC code to perform LOCA and MSLB containment response analyses in support of their power uprate. Their License Amendment Nos. 146 and 146 for Unit Nos. 1 and 2, respectively, dated June 27, 2008 (Reference 7.6), provided approval for use of the GOTHIC code. Benchmarking for these plants showed the capability of GOTHIC in the containment pressure and temperature response calculations.

The NRC Safety Evaluations addressing the Kewaunee and Prairie Island use of GOTHIC state that the NRC staff concludes that the mist formation model shall not be used for licensing calculations (Reference 8.14 and Reference 8.15). Consistent with the GOTHIC benchmark run documented in Enclosure Attachment 5 and Attachment6, section 5, the PVNGS post-LOCA containment pressure and temperature analysis does not use the mist formation model. The PVNGS analysis uses the Tagami and Uchida condensing heat transfer correlations as described in the in PVNGS UFSAR Section 6.2.1.1.3.1.

5.2. No Significant Hazards Consideration

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.

SR 3.5.1.3 requires verification that the nitrogen cover pressure in each required SIT is "≥ 600 psig and ≤ 625 psig". The proposed change revises the SR 3.5.1.3 SIT pressure band in each required SIT to "≥ 602 psia and ≤ 675 psia".

SR 3.5.2.3 requires verification that the nitrogen cover pressure in each required SIT is "≥ 260 psig and ≤ 625 psig". The proposed change revises the SR 3.5.2.3 SIT pressure band in each required SIT to "≥ 250 psia and ≤ 675 psia".

To increase operational flexibility, this change will revise SR 3.5.1.3 and SR 3.5.2.3 to increase the upper limit of their SIT pressure bands. This change will also list the SIT nitrogen cover pressure in units of pounds per square inch absolute (psia) [from the safety analyses and matching the analytical value used in the Emergency Core Cooling System (ECCS) performance and containment pressure and temperature analyses], instead of values in units of pounds per square inch gauge (psig) with instrument uncertainty included. The accounting for instrument uncertainties will continue as part of surveillance procedures. All requirements and applicability of the SRs will remain the same. Listing the SIT SR nitrogen cover pressure requirements in psia values that match the accident analysis initial conditions will maintain analysis validity and not increase the severity of any accident.

The requested TS changes do not involve any plant modifications that could affect system reliability, component performance, or the possibility of operator error. 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 to postulated accidents have been analyzed using the increased upper limit of the SIT pressure bands. These evaluations results show that the consequences of the postulated accidents remain within applicable acceptance criteria.

Through benchmarking and sensitivity studies, it is demonstrated that the use of the GOTHIC code for the post-LOCA containment pressure and temperature response analysis application produces results that satisfy all applicable design and safety analysis acceptance criteria. Since use of the GOTHIC code conforms to design bases and its results are bounded by the design limit, the GOTHIC results remain within the plant design basis and will not cause an increase in the probability or consequences of an accident previously evaluated. Adherence to safety analysis acceptance criteria prevents use of the GOTHIC code from creating new challenges to components and systems that could adversely affect their ability to mitigate accident consequence or diminish integrity of any fission product barrier. Thus, the requested use of the GOTHIC code as part of the methodology to perform calculations of the containment pressure and temperature response to various postulated pipe breaks does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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.

SR 3.5.1.3 requires verification that the nitrogen cover pressure in each required SIT is "≥ 600 psig and ≤ 625 psig". The proposed change revises the SR 3.5.1.3 SIT pressure band in each required SIT to "≥ 602 psia and ≤ 675 psia".

SR 3.5.2.3 requires verification that the nitrogen cover pressure in each required SIT is "≥ 260 psig and ≤ 625 psig". The proposed change revises the SR 3.5.2.3 SIT pressure band in each required SIT to "≥ 250 psia and ≤ 675 psia".

To increase operational flexibility, this change will revise SR 3.5.1.3 and SR 3.5.2.3 to increase the upper limit of their SIT pressure bands. This change will also list the SIT nitrogen cover pressure in units of psia [matching the analytical value used in the ECCS performance and containment pressure and temperature analyses], instead of values in units of psig with instrument uncertainty included. The accounting for instrument uncertainties will continue as part of surveillance procedures. All requirements and applicability of the SRs will remain the same. The response to postulated accidents have been analyzed using the increased upper limit of the SIT pressure bands.

The proposed changes 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. The proposed changes do not introduce any new safety functions for plant structures, systems, or components.

The use of the GOTHIC code is a change in analysis methods applied to containment pressure and temperature response to various postulated pipe breaks. Analysis methods are not accident initiators. The GOTHIC code will be applied in a manner consistent with the current licensing basis COPATTA code and with current plant design bases and licensed accident analysis methodologies. The use of the GOTHIC code does not adversely affect any fission product barrier, nor does it alter the safety function of safety related systems, structures, and components depended upon for accident prevention or mitigation. Equipment important to safety will continue to function within design. Through benchmarking and sensitivity studies, it is demonstrated that the use of the GOTHIC code for the post-LOCA containment pressure and temperature response analysis application produces results that satisfy all applicable design and safety analysis acceptance criteria. Thus, the requested use of the GOTHIC code as part of the methodology to perform calculations of the containment pressure and temperature response to various postulated pipe breaks does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes do not create the possibility of a new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing basis and thus this proposed change does not create the possibility of a new or different kind of accident than previously evaluated.

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

Response: No.

SR 3.5.1.3 requires verification that the nitrogen cover pressure in each required SIT is "≥ 600 psig and ≤ 625 psig". The proposed change revises the SR 3.5.1.3 SIT pressure band in each required SIT to "≥ 602 psia and ≤ 675 psia".

SR 3.5.2.3 requires verification that the nitrogen cover pressure in each required SIT is "≥ 260 psig and ≤ 625 psig". The proposed change revises the SR 3.5.2.3 SIT pressure band in each required SIT to "≥ 250 psia and ≤ 675 psia".

To increase operational flexibility, this change will revise SR 3.5.1.3 and SR 3.5.2.3 to increase the upper limit of their SIT pressure bands. This change will also list the SIT nitrogen cover pressure in units of psia [matching the analytical value used in the ECCS performance and containment pressure and temperature analyses], instead of values in units of psig with instrument uncertainty included. The accounting for instrument uncertainties will continue as part of surveillance procedures. All requirements and applicability of the SRs will remain the same. The response to postulated accidents have been analyzed using the increased upper limit of the SIT pressure bands and continue to meet their applicable general design criteria (GDC) requirements.

The existing TS safety limits are not being changed. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The use of the GOTHIC code affects the containment pressure and temperature response to various postulated pipe breaks. Through benchmarking and sensitivity studies, it is demonstrated that the use of the GOTHIC code for the post-LOCA containment pressure and temperature response analysis application produces results that satisfy all applicable design and safety analysis acceptance criteria. The results predicted by use of the GOTHIC code for these analyses remain within the limiting design basis accidents of record. Safety analysis acceptance criteria are satisfied and adherence to safety analysis acceptance criteria using the GOTHIC code assures that Technical Specification limits will not be exceeded during normal operation. Thus, the requested use of the GOTHIC code as part of the methodology to perform calculations of the containment pressure and temperature response to various postulated pipe breaks does not involve a significant reduction in the margin of safety.

Modifying the SRs to list SIT nitrogen cover pressure in units of psia does not change the physical or operational conditions. The increase in the upper limit of the SIT pressure bands has been analyzed and the existing TS safety limits continue to be maintained. Therefore, as there is no change to the design basis or safety limit, the proposed changes do not involve a significant reduction in the margin of safety.

5.3. Conclusion

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.

6.0 ENVIRONMENTAL ASSESSMENT

The proposed amendment involves components located within the restricted area, as defined in 10 CFR 20, Standards for Protection Against Radiation, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or 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 need be prepared in connection with the proposed amendment would change a requirement with respect to installation or use of a facility.

7.0 REFERENCES

7.1 GOTHIC, "Generation of Thermal Hydraulic Information for Containments," Version 8.4(QA), Electric Power Research Institute, Palo Alto, CA, 2022.

7.2 Code of Federal Regulations, Title 10, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

7.3 Code of Federal Regulations, Title 10, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

7.4 Waterford 3 Steam Electric Station License Amendment No. 165 for Reducing Operable Containment Fan Coolers in the Containment Cooling System, July 6, 2000, Agency Document Access and Management System (ADAMS) Accession No. ML003731172.

7.5 Joseph M. Farley Nuclear Plants, Units 1 and 2 License Amendment Nos. 137 and 129 for Power Uprate, April 29, 1998, ADAMS Accession Nos. ML013130055 and ML012140259.

7.6 Comanche Peak Steam Electric Station, Units 1 and 2 - Issuance of Amendments [Nos. 146 and 146, respectively] Re: License Amendment Request 07-004, Revision to Operating License and Technical Specification 1.0, "Use And Application," to Revise Rated Thermal Power from 3458 MWt to 3612 MWt (TAC Nos. MD6615 and MD6616), June 27, 2008, ADAMS Accession No. ML081510173.

ATTACHMENT 1

Technical Specifications Page Mark-up

Affected Pages: 3.5.1-2, 3.5.2-2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. AND	6 hours
	D.2 Reduce pressurizer pressure to < 1837 psia.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each SIT isolation valve is fully open.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	Verify borated water volume in each SIT is ≥ 1750 cubic feet and ≤ 1950 cubic feet.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify nitrogen cover pressure in each SIT is ≥ ~~600 psig~~ 602 psia and ≤ ~~625 psig~~ 675 psia.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify each required SIT isolation valve is fully open when pressurizer pressure is ≥ 430 psia.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.2	Verify borated water volume in each required SIT is: a. For four OPERABLE SITs, > 908 cubic feet and < 2000 cubic feet. OR b. For three OPERABLE SITs, > 1361 cubic feet and < 2000 cubic feet.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.3	Verify nitrogen cover pressure in each required SIT is ≥ ~~260 psig~~ 250 psia and ≤ ~~625 psig~~ 675 psia.	In accordance with the Surveillance Frequency Control Program

(continued)

ATTACHMENT 2

Clean Technical Specifications Pages

Affected Pages: 3.5.1-2, 3.5.2-2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	D.2 Reduce pressurizer pressure to < 1837 psia.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each SIT isolation valve is fully open.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	Verify borated water volume in each SIT is ≥ 1750 cubic feet and ≤ 1950 cubic feet.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify nitrogen cover pressure in each SIT is ≥ 602 psia and ≤ 675 psia.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify each required SIT isolation valve is fully open when pressurizer pressure is ≥ 430 psia.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.2	Verify borated water volume in each required SIT is: a. For four OPERABLE SITs, > 908 cubic feet and < 2000 cubic feet. <u>OR</u> b. For three OPERABLE SITs, > 1361 cubic feet and < 2000 cubic feet.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.3	Verify nitrogen cover pressure in each required SIT is ≥ 250 psia and ≤ 675 psia.	In accordance with the Surveillance Frequency Control Program

(continued)

ATTACHMENT 3

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

Affected Pages: B 3.5.1-5 and B 3.5.2-4

APPLICABLE SAFETY ANALYSES ~~(continued)~~

A minimum pressure of ~~588 psig~~ 602 psiaand a maximum pressure of ~~637 psig~~675 psia are used in the analyses. ~~To allow for instrument accuracy, a 600 psig minimum and 625 psig maximum are specified~~. The required SIT nitrogen cover pressures from the safety analyses are converted to pounds per square inch gauge (psig) with instrument uncertainties applied to enable performance of surveillance procedures. The maximum allowable boron concentration of 4400 ppm is based upon boron precipitation limits in the core following a LOCA. Establishing a maximum limit for boron is necessary since the time at which boron precipitation would occur in the core following a LOCA is a function of break location, break size, the amount of boron injected into the core, and the point of ECCS injection. Post LOCA emergency procedures directing the operator to establish simultaneous hot and cold leg injection are based on the worst case minimum boron precipitation time. Maintaining the maximum SIT boron concentration within the upper limit ensures that the SITs do not invalidate this calculation. An excessive boron concentration in any of the borated water sources used for injection during a LOCA could result in boron precipitation earlier than predicted.

The 2300 ppm minimum boron concentration in the SITs assures that the back leakage from the RCS will not dilute the SITs below the minimum boron concentration in the safety analysis. The minimum safety analysis boron requirements of 2000 ppm are based on beginning of life reactivity values and are selected to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. During a large break LOCA, all Control Element Assemblies (CEAs) are assumed not to insert into the core, and the initial reactor shutdown is accomplished by void formation during blowdown. Sufficient boron concentration must be maintained in the SITs to prevent a return to criticality during reflood. Although this requirement is similar to the basis for the minimum boron concentration of the Refueling Water Tank (RWT), the minimum SIT concentration is lower than that of the RWT since the SITs need not account for dilution by the RCS during a large break LOCA.

The SITs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the SITs are available to accomplish their core cooling safety function following a LOCA. Four SITs are required to be OPERABLE to ensure that 100% of the contents of three of the SITs will reach the core during a LOCA.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

The minimum nitrogen cover pressure requirement ensures that the contained gas volume will generate discharge flow rates during injection that are consistent with those assumed in the safety analyses.

The maximum nitrogen cover pressure limit ensures that excessive amounts of gas will not be injected into the RCS after the SITs have emptied.

A minimum pressure of ~~235 psig~~ 250 psia (analysis value of 249.2 psia rounded conservatively) and a maximum pressure of ~~637 psig~~ 675 psia are used in the analyses. ~~To allow for instrument accuracy, a 260 psig minimum and 625 psig maximum are specified~~The required SIT nitrogen cover pressures from the safety analyses are converted to pounds per square inch gauge (psig) with instrument uncertainties applied to enable performance of surveillance procedures. The maximum allowable boron concentration of 4400 ppm is based upon boron precipitation limits in the core following a LOCA. Establishing a maximum limit for boron is necessary since the time at which boron precipitation would occur in the core following a LOCA is a function of break location, break size, the amount of boron injected into the core, and the point of ECCS injection. Post LOCA emergency procedures directing the operator to establish simultaneous hot and cold leg injection are based on the worst case minimum boron precipitation time. Maintaining the maximum SIT boron concentration within the upper limit ensures that the SITs do not invalidate this calculation. An excessive boron concentration in any of the borated water sources used for injection during a LOCA could result in boron precipitation earlier than predicted.

The 2300 ppm minimum boron concentration in the SITs assures that the back leakage from the RCS will not dilute the SITs below the minimum boron concentration in the safety analysis. The minimum safety analysis boron requirements of 2000 ppm are based on beginning of life reactivity values and are selected to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. Sufficient boron concentration must be maintained in the SITs to prevent a return to criticality during reflood. Although this requirement is similar to the basis for the minimum boron concentration of the Refueling Water Tank (RWT), the minimum SIT concentration is lower than that of the RWT since the SITs need not account for dilution by the RCS.

SIT-Shutdown satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

In MODES 3 and 4 with pressurizer pressure less than 1837 psia, the LCO establishes the minimum conditions required to ensure that the required SITs are available to accomplish their core cooling safety function following a LOCA. The number of SITs required to be OPERABLE is based on the minimum required

(continued)

ATTACHMENT 4

Affidavit from Framatome
Submitted in Accordance with 10 CFR 2.390
to Consider Attachment 8 as a Proprietary Document

AFFIDAVIT

1. My name is Morris Byram. I am Product Manager, Licensing & Regulatory Affairs for Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the Framatome information contained in the enclosure entitled “Palo Verde Units 1,2, and 3 Large Break LOCA Analysis,” in the Arizona Public Service (APS) Company letter number 102-08841, and referred to herein as “Document.” Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

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

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) “Trade secrets and commercial or financial information.”

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

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

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

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

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

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

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

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

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

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

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

Executed on: (7/18/2024)

BYRAM Morris

Digitally signed by BYRAM
Morris
Date: 2024.07.18 13:46:49 -07'00'

(NAME)
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morris.byram@framatome.com

ATTACHMENT 5

Benchmark Evaluation for Use of the GOTHIC Code

ATTACHMENT 5

TABLE OF CONTENTS

1.0 INTRODUCTION

As described in the PVNGS Updated Final Safety Analysis Report (UFSAR), PVNGS currently uses the Bechtel Containment Pressure and Temperature Transient Analysis (COPATTA) code as part of the methodology for the calculation of the containment pressure and temperature response to various postulated pipe breaks. This methodology is not addressed in the PVNGS Technical Specifications.

In 2021, Bechtel elected to discontinue development, maintenance, and licensing of its software products, including the COPATTA code. The Bechtel actions necessitate APS use of a different code to perform containment pressure and temperature response analyses.

This benchmark evaluation addresses use of the industry standard Generation of Thermal Hydraulic Information for Containments (GOTHIC) code, in place of the current COPATTA code, as part of the methodology to perform calculations of the containment pressure and temperature response to various postulated pipe breaks.

For the purpose of the containment peak pressure and temperature analysis, the safety injection (SI) system and the containment heat removal systems (i.e., containment spray system) were assumed to operate in the mode that maximizes the containment peak pressure. For the SI system, both maximum and minimum Emergency Core Cooling System (ECCS) performances were evaluated. The double-ended discharge leg slot (DEDLS) break LOCA with maximum ECCS (i.e., SI) flow was identified as the pipe break with the highest peak pressure. Therefore, this benchmark evaluation documents a benchmark of COPATTA and GOTHIC codes for the DEDLS break LOCA with maximum SI flow.

2.0 GOTHIC DESCRIPTION

GOTHIC is an integrated, general purpose thermal-hydraulics software package for design, licensing, safety and operating analysis of nuclear power plant containments, confinement buildings, and system components. Applications of GOTHIC include evaluation of containment and containment sub-compartment response to the full spectrum of high energy line breaks within the design basis envelope as described in UFSAR Chapter 6, and a wide variety of systems evaluations involving multiphase flow and heat transfer, gas mixing and other thermal hydraulic behavior. Applications may include, but are by no means limited to, pressure and temperature determination, equipment qualification profiles and inadvertent system initiation, and degradation or failure of engineered safety features. As a general purpose tool, GOTHIC can be used for a wide variety of plant operations support issues involving single and multiphase heat transfer and fluid flow provided that the application is consistent with the underlying physical basis and assumptions and the code validation basis.

GOTHIC includes an extensive set of models for operating equipment. These items, referred to collectively as components, include the following:

- Pumps and fans
- Valves and doors
- Heat exchangers and fan coolers

- Vacuum breakers
- Spray nozzles
- Coolers and heaters
- Volumetric fans (annular fans, deck fans, etc.)
- Hydrogen recombiners (forced and natural convection)
- Ignitors (spark device used to ignite hydrogen burns)
- Pressure relief valves (PRVs)
- Tracer filters
- Dryer/Demisters
- Charcoal filters

Initial conditions allow the user to specify the state of the fluid and solid structures within the modeled region at the start of the transient. These include the initial temperature and composition of the vapor phase, the location and temperature of liquid pools, the location and amount of a liquid component, and the temperatures of solid structures within the building.

Neutron kinetics models are available in GOTHIC to predict the time dependent processes associated with the nuclear fission chain reaction, predominantly the time dependent behavior of the neutron population. Prompt and delayed fission modeling is consistent with the traditional "space in-dependent" or "point-kinetics" power formulation. However, the formulation is extended to also address heat generation from decay of actinides and from decay of fission products in the form of "decay heat".

A tracer transport model with optional radioactive decay is available in GOTHIC. Tracers are included in the initial system inventory or can be added to the system via Tracer Sources located in a cell or a boundary condition. Radioactive decay can be modeled with up to three daughter products for each tracer. The inventory for each tracer can be tracked in the vapor and liquid phases, each drop field, conductor surfaces and tracer and charcoal filters.

The user can define a liquid component that represents particles or liquid globules that move with the liquid and can settle from a pool. A liquid component is tracked as a volume fraction of the continuous liquid phase.

Table Functions and Control Variables can be used to modify nominal parameter values and provide a great deal of flexibility for modeling. Control Variables are defined by a functional form and a set of input Variables. Trips are used to control the operational status of boundary conditions and components. Trips can be defined to activate based on the value of several variables, including control variables. The ability to invoke a trip based on the value of a control variable extensively expands the possibilities of trip conditions.

The GOTHIC Technical Manual (Reference 5.1) describes the equations and models and the numerical methods used to solve them. The GOTHIC Technical Manual provides the necessary information for understanding GOTHIC, including assumptions and limitations.

3.0 BENCHMARKING OF GOTHIC RESULTS

The general approach for the conversion of the COPATTA model of the DEDLS break LOCA with maximum SI flow to a corresponding GOTHIC model is to ensure as much as practical that identical inputs are used between COPATTA and GOTHIC modeling. The inputs to the GOTHIC benchmark model were not intentionally adjusted to match COPATTA results. When the results between GOTHIC and COPATTA differed, GOTHIC parameters were subsequently adjusted and additional runs were performed, to ensure the reason(s) for the differences were understood. To ensure the validity of the converted GOTHIC model, the containment pressure and temperature results of the COPATTA evaluation of the DEDLS with maximum SI flow case are compared with the corresponding GOTHIC evaluation of the DEDLS with maximum SI flow case. Any significant variations in containment pressure and temperature profiles are dispositioned accordingly.

The conversion of the COPATTA model of the DEDLS break LOCA with maximum SI flow to a corresponding GOTHIC model is performed using GOTHIC Version 8.4 (Reference 5.2). However, the analysis models and methods described in this benchmark evaluation are not intended to be restricted to a specific GOTHIC code version.

3.1 COPATTA to GOTHIC Benchmark Model Conversion

The COPATTA to GOTHIC model conversion for the benchmark uses GOTHIC inputs and modeling techniques that are intended to be identical to the COPATTA inputs and modeling techniques. It is noted that the COPATTA and GOTHIC benchmark runs both model nitrogen gas injection from one SIT. For the PVNGS UFSAR COPATTA results, the impact of nitrogen gas injection from the other three SITs was determined outside of the COPATTA run, and therefore only one SIT is modeled in the GOTHIC benchmark run.

The following GOTHIC modeling elements are utilized for the COPATTA benchmark. Detailed descriptions of the input are presented in the GOTHIC User Manual (Reference 5.3).

Control Volumes – Lumped parameter Control Volumes are used to define the containment free volume and the available water inventory in the Reactor Coolant System (RCS) at the End of Post-Reflood. The containment control volume models both the containment vapor space and any liquid in the sump. The RCS control volume models the RCS water below the break available for mass and energy balance at the End of Post-Reflood.

Thermal Conductors, Conductor Types, and Materials – Thermal Conductors are used to define the heat sinks inside containment. Conductor Types and Material Types are used to model conduction through heat sinks. Surface options are used to model the heat exchange between the heat sink surfaces and the containment atmosphere/sump through condensation and convection. Additionally, surface options are used to define heat exchange with external boundaries such as the outside atmosphere.

Boundary Conditions – Boundary Conditions provide system mass and energy sources and sinks within containment. These include mass and energy release from the break, mass of gas injected to containment via the SITs, and Safety Injection (SI) and Containment Spray (CS) flows during both direct Injection and Recirculation modes of operation.

Control Variables – Control Variables are used to define SI and CS flow profiles and to track the temperature profiles of the sump and containment vapor space, pressure of containment, Shutdown Cooling (SDC) heat exchanger heat transfer coefficients, containment steam and gas mass ratios (used for spray efficiency) and other key parameters.

Flow Paths – Flow Paths connect Boundary Conditions to the containment atmosphere and sump. Additionally, a flow path connects the available RCS inventory for boiling or flooding to containment.

Heaters – Heaters provide sources of heat due to reactor core decay heat and miscellaneous sensible energy added to the containment atmosphere.

Heat Exchangers – A Heat Exchanger component is used to model the SDC Heat Exchanger performance. Note that no Containment Air Coolers are modeled in the COPATTA model.

Valves – A Valve component is used to maintain isolation of the RCS inventory prior to End of Post-Reflood, consistent with the COPATTA modeling.

Table Functions – Table Functions are used to define mass and enthalpy profiles, decay and miscellaneous sensible energy added to the containment atmosphere profiles, the CS nozzle efficiency profile, fraction of SI flow spillage to the sump, and SDC heat exchanger heat transfer coefficients. They are also used to import resultant transient COPATTA pressure, temperature, condensing heat transfer coefficient, and energy profiles for comparison to GOTHIC results within the GOTHIC postprocessor.

Initial Conditions – Initial Conditions are used to initialize the containment heat sinks and control volume fluid conditions.

Trips – Trips are used to turn off the injection mode of operation, turn on the recirculation mode of operation, turn on/off non-condensable gas injection from the SIT, and turn on/off heaters as appropriate consistent with COPATTA modeling.

3.2 COPATTA to GOTHIC Results Comparison

The COPATTA to GOTHIC benchmark is performed by comparing the results of the COPATTA evaluation of the DEDLS with maximum SI flow case to the results of the corresponding GOTHIC evaluation of the DEDLS with maximum SI flow case. To demonstrate validity, the following results have been compared:

- Containment Pressure
- Containment Vapor Temperature
- Containment Sump Temperature
- Containment Liner Temperature
- Transient Energy Profiles
- Transient Mass Profiles

The following terms define the results of this COPATTA to GOTHIC results comparison:

Graphically Identical – Results from GOTHIC are visually indistinguishable from the

COPATTA results when overlaid on the same plot.

Nearly Identical – Results from GOTHIC have slight variations for brief periods of time when compared to the equivalent COPATTA results. These variations are considered inconsequential based on engineering judgment.

Valid – Acknowledgment that the conversion from COPATTA to GOTHIC was deemed appropriate for the specific set of results evaluated.

3.2.1 Containment Pressure Benchmarking

In Figure 3.0-1, the COPATTA containment pressure profile is compared to the resultant GOTHIC containment pressure profile. The plot shows GOTHIC predicts a lower peak containment pressure than COPATTA by approximately 0.83 psi. The COPATTA results also show a pressure increase up to 95 seconds (the time of Containment Spray initiation) and then a momentary sharp reduction in the pressure before reaching the peak pressure. This is similar in trend to the temperature profile (Figure 3.0-3) where there is a significant increase in containment vapor space temperature compared to the GOTHIC results.

GOTHIC employs a sophisticated method for modeling liquid droplets, using industry-standard mean drop diameters for blowdown and spray phases, which allows droplets to remain suspended in the containment atmosphere until reaching saturation conditions. Westinghouse also noted this when comparing GOTHIC results to results from other legacy containment codes (Reference 5.4). This approach results in GOTHIC predicting a lower peak containment pressure compared to COPATTA due to more realistic heat and mass transfer dynamics. In contrast, COPATTA immediately sends introduced liquid to the sump unless the containment atmosphere becomes superheated, in which case small droplets are reintroduced for instant vaporization. This conservative approach leads to spikes in temperature above saturation in COPATTA's results, contrasting with GOTHIC's maintenance of saturation conditions. The GOTHIC modeling is more realistic and removes some of the excessive conservatism in the COPATTA modeling.

A sensitivity study in GOTHIC illustrates that modifying the droplet removal process and enhancing thermal equilibrium through a recirculation flow path can improve agreement with COPATTA's outcomes, highlighting the impact of droplet handling on containment simulation results between the two codes. The sensitivity case adds a recirculation flow path with a volumetric fan operating at a high flow rate and a reduced droplet diameter to promote thermal equilibrium as the drops enter the containment vapor space. The modeling is intended to replicate the COPATTA results by quickly removing suspended droplets. Figure 3.0-2 shows much closer agreement between COPATTA and GOTHIC containment atmosphere pressure profiles when using the recirculation fan modeling and reduced droplet size in GOTHIC.

Figure 3.0-1: Containment Pressure Profile GOTHIC vs. COPATTA

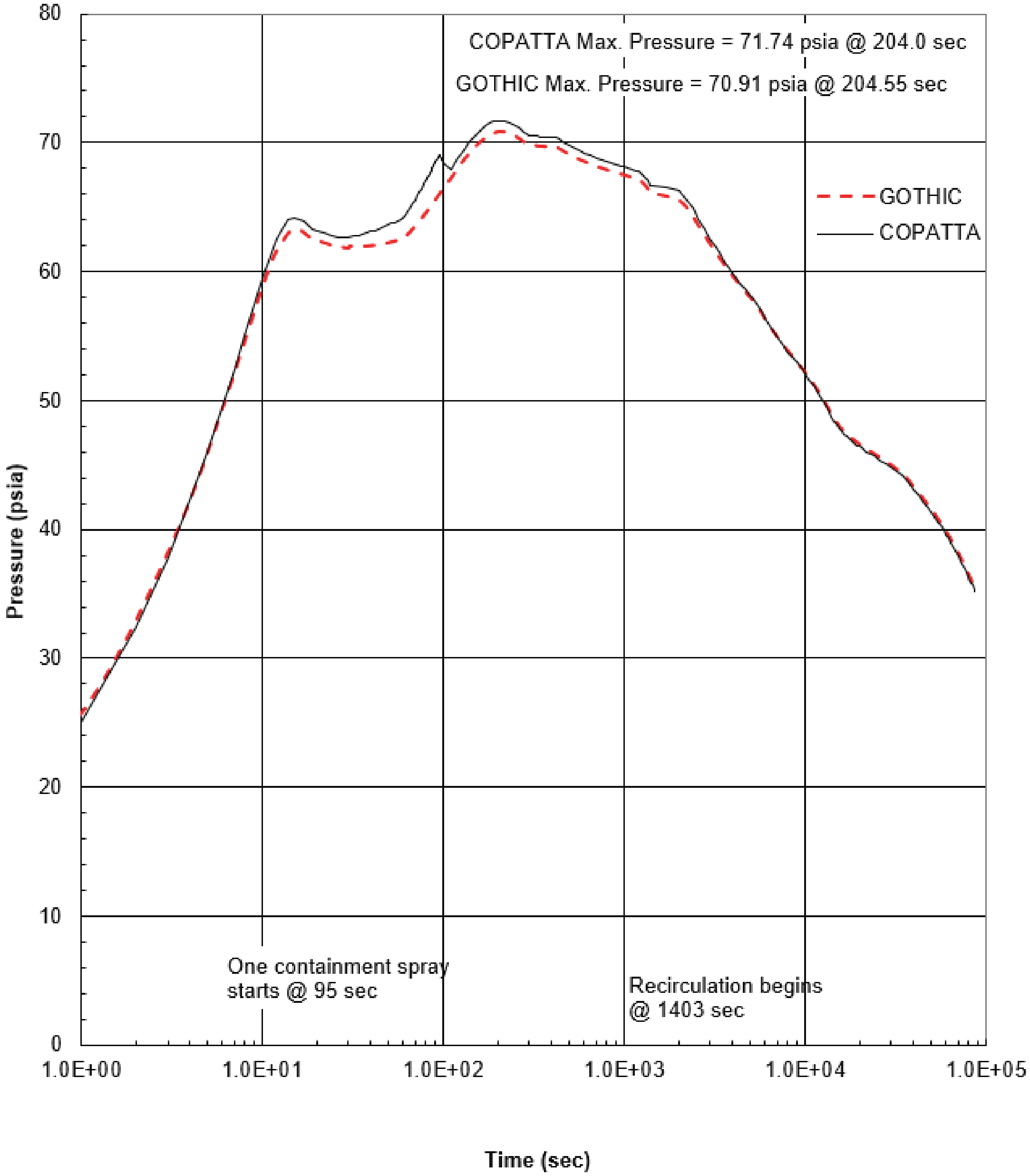
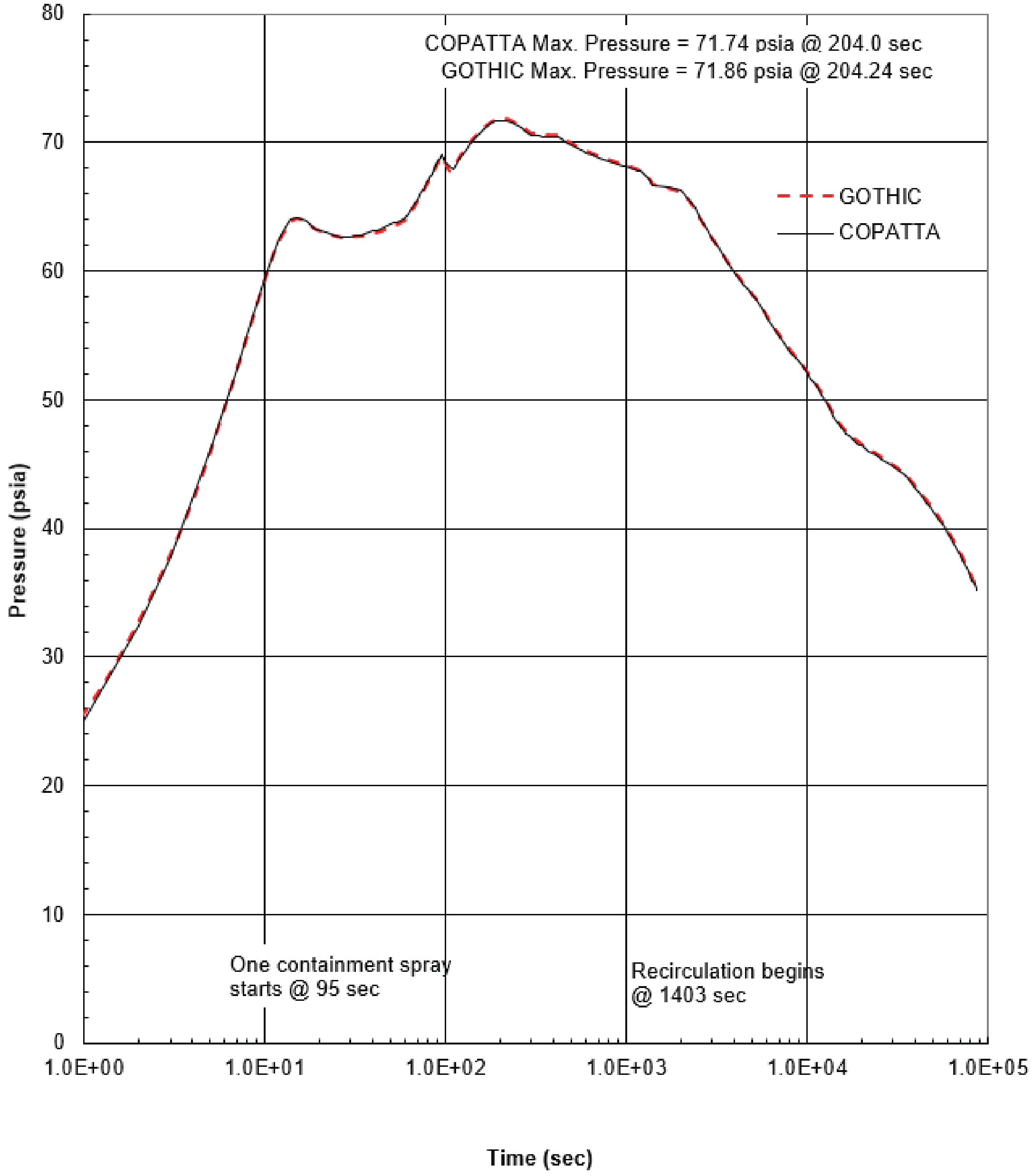


Figure 3.0-2: Containment Pressure Profile GOTHIC vs. COPATTA – Recirculation Fan Modeled



3.2.2 Containment Vapor and Sump Temperature Benchmarking

Figure 3.0-3 shows the COPATTA containment temperature profile compared to the GOTHIC containment temperature profile for both the vapor space and liquid sump regions of containment.

<u>Vapor Temperature</u> - The plotted COPATTA results show a sharp vapor temperature increase up to 95 seconds (the time of Containment Spray initiation) and then a sharp reduction in

temperature. This trend is similar to the pressure profile (Figure 3.0-1) where there is a significant increase in containment pressure compared to the GOTHIC results. Figure 3.0-3 shows the resulting GOTHIC containment vapor temperature, which is maintained at saturation compared to the COPATTA results, including a spike in temperature above the saturation temperature at the steam pressure. This spike in temperature (or superheating) is due to how COPATTA treats suspended liquid in the vapor. At the end of each time step, COPATTA removes the liquid phase from the containment atmosphere and adds it to the sump (Reference 5.5). In GOTHIC, the liquid enters as droplets that remain suspended in the atmosphere and maintain it at saturation, with no superheat (Reference 5.1). The GOTHIC modeling is more realistic and removes some of the excessive conservatism in the COPATTA modeling.

Figure 3.0-4 shows the temperature results from the fan recirculation sensitivity case described in Section 3.2.1. These results show much closer agreement between COPATTA and GOTHIC containment atmosphere temperature profiles, further confirming that the discrepancy in containment atmosphere temperature in Figure 3.0-3 is caused by COPATTA removing the suspended liquid from the containment atmosphere and directly adding it to the sump. The COPATTA results show a drop in vapor temperature from 120°F to approximately 93°F within the first 0.1 seconds. The temperature then rapidly climbs thereafter. GOTHIC does not exhibit this trend for the nominal case; however; GOTHIC exhibits the same vapor temperature drop when the recirculation fan is modeled. Thus, this difference in vapor temperature before 1 second can be attributed to the different approaches used by the two codes in handling the droplet phase.

<u>Sump Temperature</u> – Temperature differences in the short term are due to the way the codes treat the liquid phase introduced to the containment atmosphere from the break. It is also important to note for the short-term comparison that there is no initial inventory in the sump. In GOTHIC, the liquid phase emerging from the break is modeled as droplets that remain suspended in the containment atmosphere before either vaporizing or eventually settling to the sump (Reference 5.1). In COPATTA, any portion of the blowdown that emerges in the liquid phase is added to the sump at the saturation temperature at the end of each time step (Reference 5.5).

Since both the COPATTA and GOTHIC models do not credit heat transfer between the sump and the containment atmosphere, the sump temperature is unimportant in the model until recirculation initiates at 1403 seconds (note that COPATTA bounds GOTHIC in the short term). By this time, the calculated sump temperatures from GOTHIC and COPATTA show good agreement as illustrated in Figure 3.0-3.

Figure 3.0-3: Containment Vapor and Sump Temperature Profiles GOTHIC vs. COPATTA

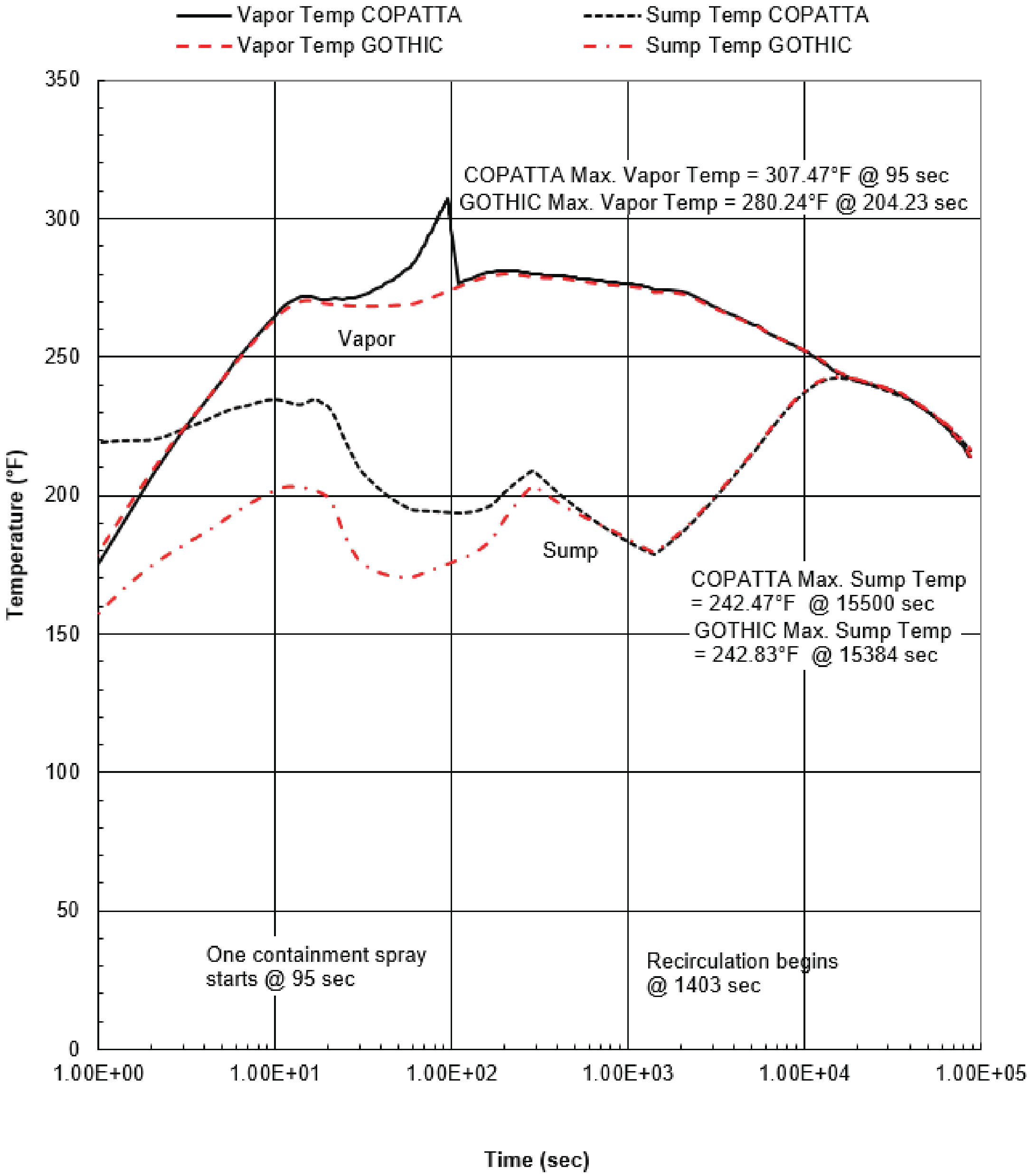
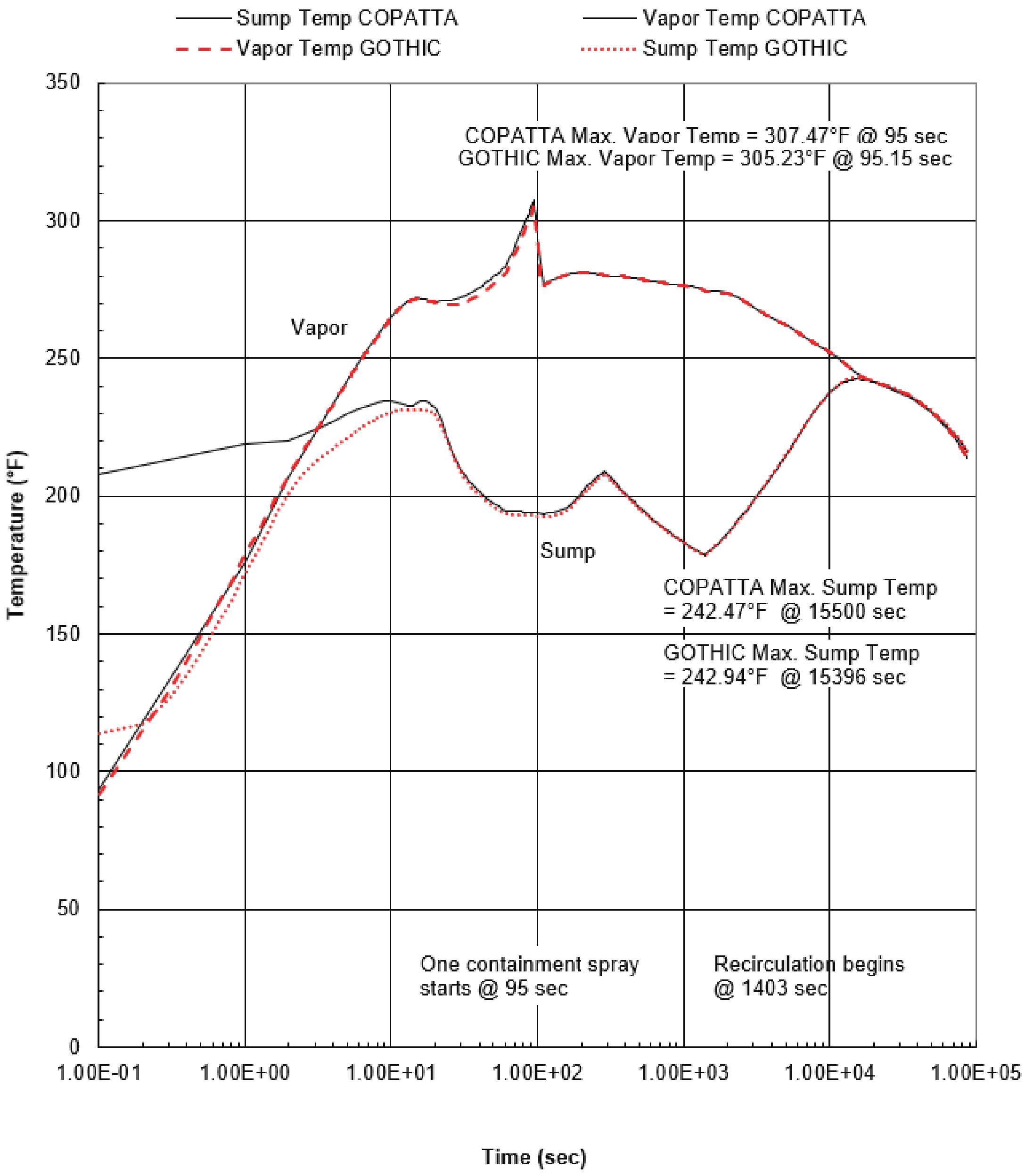


Figure 3.0-4: Containment Vapor and Sump Temperature Profiles GOTHIC vs. COPATTA – Recirculation Fan Modeled

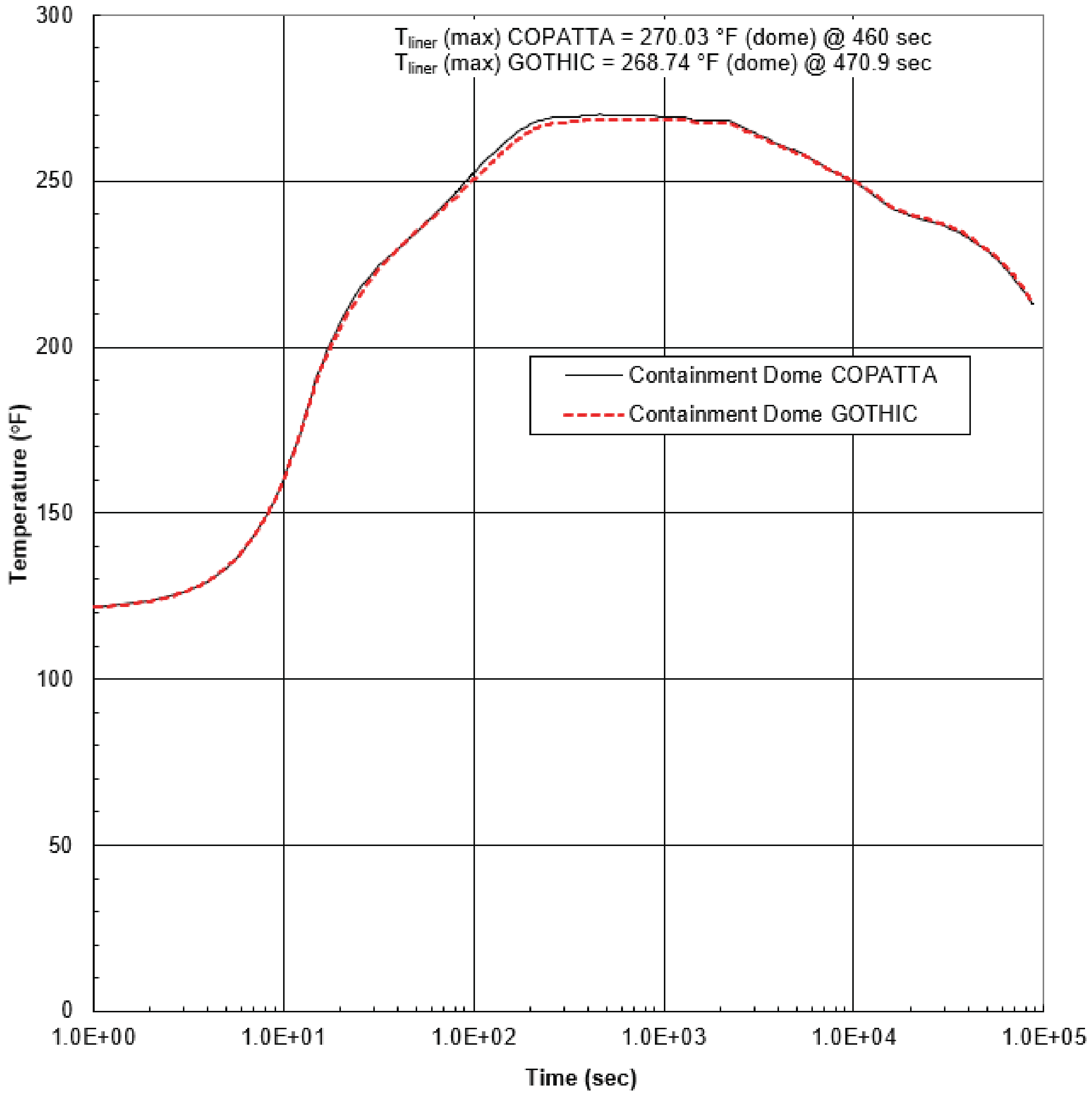


3.2.3 Containment Liner Temperature Benchmarking

Figure 3.0-5 shows the COPATTA containment liner temperature profile compared to the GOTHIC containment liner temperature profile for the containment dome. Note that results for the containment liner for the containment cylinder walls and buttress sections are nearly identical to the results for the containment dome and therefore are not shown in Figure 3.0-5.

Figure 3.0-5 shows good agreement between the COPATTA and GOTHIC results. The slight variation in temperature is attributed to the variation in the way the Uchida condensation heat transfer coefficient is modeled, in that GOTHIC provides a more precise Uchida correlation curve and yields a slightly lower condensing heat transfer coefficient for the steam-to-gas mass ratios utilized in the analysis.

Figure 3.0-5: Containment Liner Temperature Profiles GOTHIC vs. COPATTA



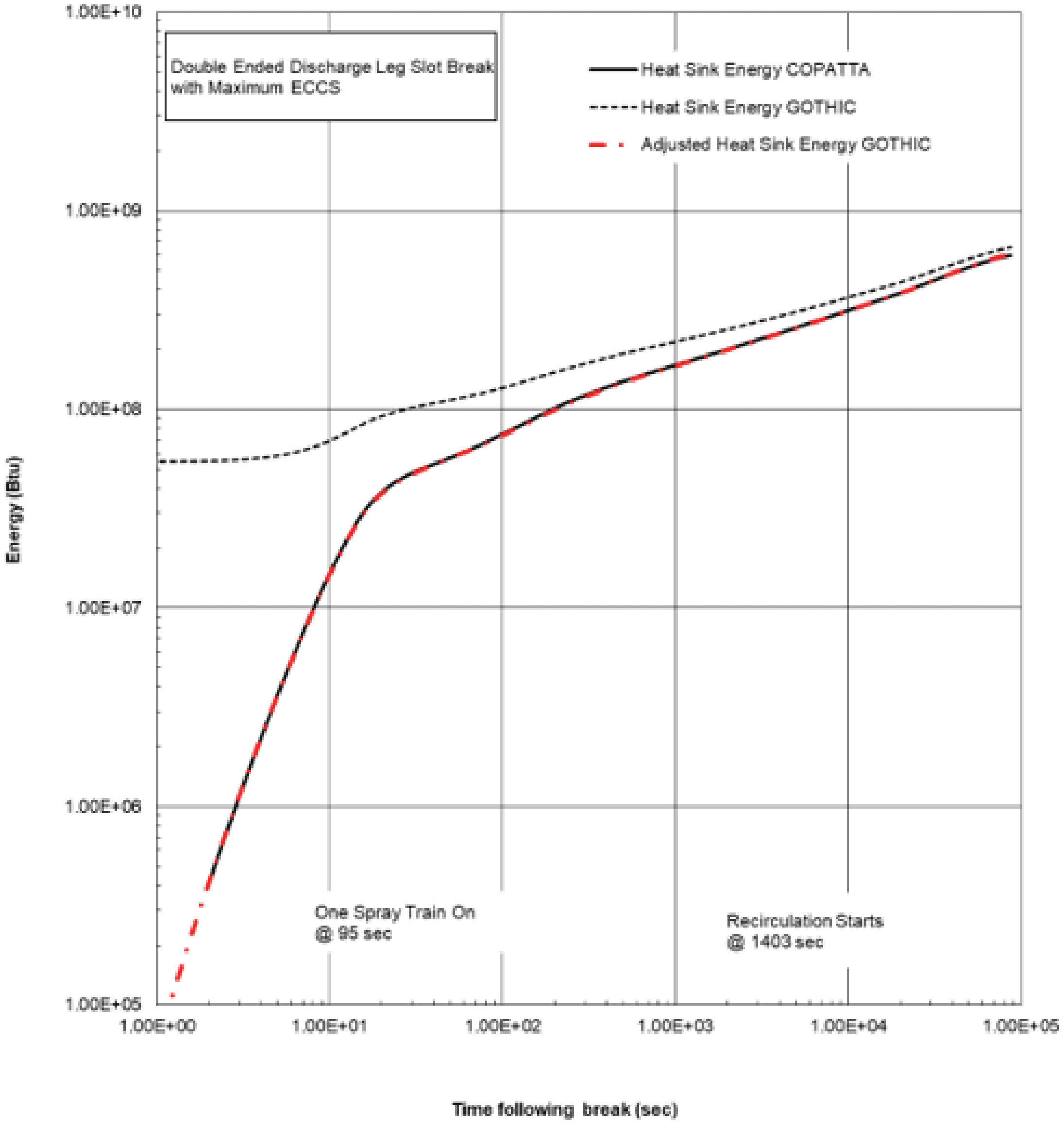
3.2.4 Transient Energy Profiles Benchmarking

3.2.4.1 <u>Heat Sink Energy Benchmarking</u>

The Heat Sink Energy in COPATTA is the integral of the heat transfer rate over time (Btu). Comparison of the COPATTA and GOTHIC results for the integrated heat sink (or conductor) energy is shown in Figure 3.0-6.

The COPATTA heat sink energy profile starts near zero at the beginning of the transient run. The GOTHIC heat sink energy profile has a significant amount of reported energy after the first printed time interval. This indicates there is a discrepancy between the reference point used in determining the integrated heat sink energy that COPATTA uses compared to GOTHIC. Making an adjustment to the initial reported energy term in GOTHIC results by setting the energy at the initial printed time interval to zero results in graphically identical heat sink energy profiles for COPATTA and GOTHIC.

Figure 3.0-6: Integrated Heat Sink Energy GOTHIC vs. COPATTA

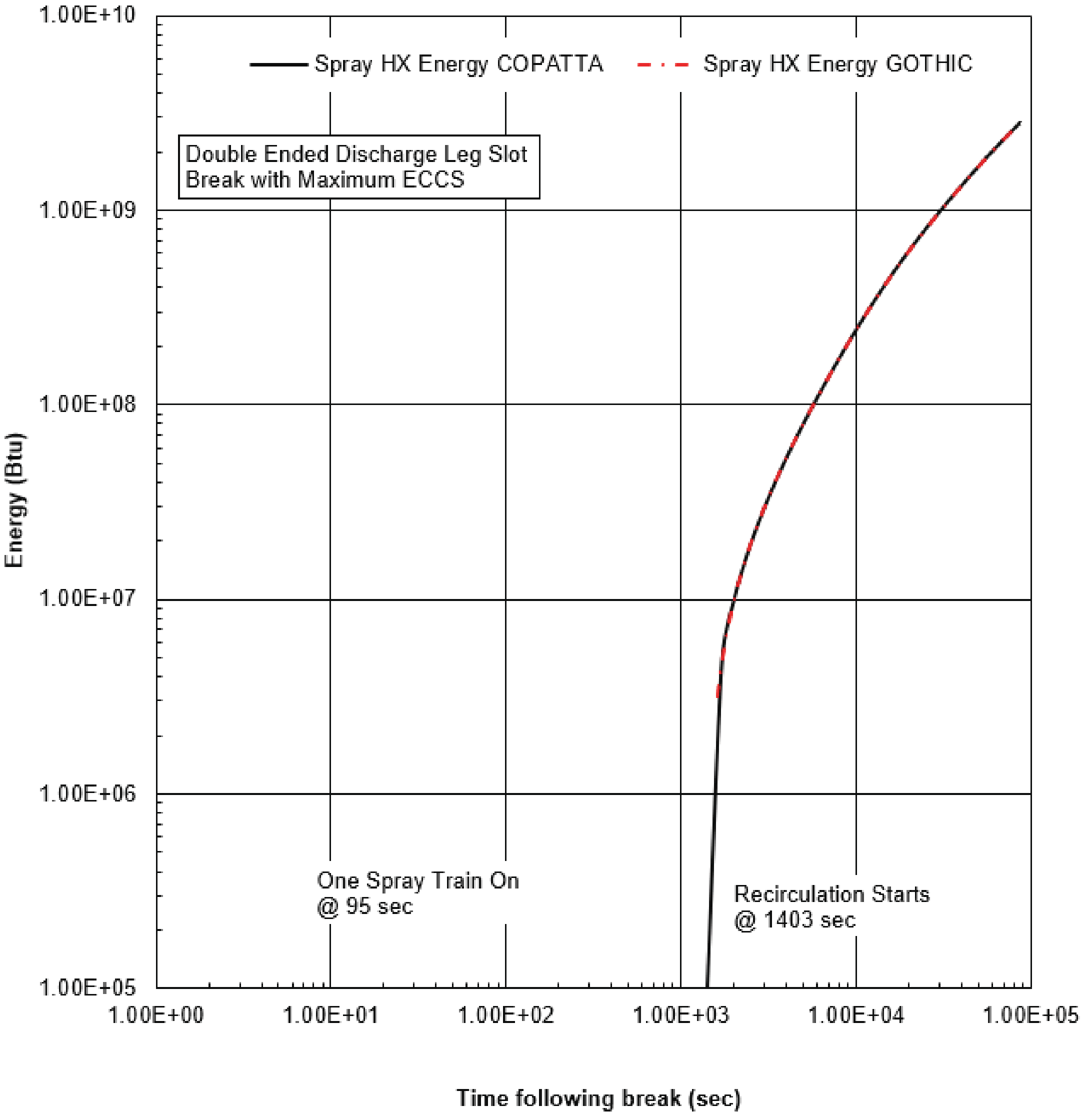


3.2.4.2 <u>Spray Heat Exchanger (SDC Heat Exchanger) Energy Benchmarking</u>

Comparison of the COPATTA and GOTHIC results for the total integrated SDC heat exchanger energy is shown in Figure 3.0-7.

The GOTHIC results are nearly identical to the COPATTA results for the SDC heat exchanger energy.

Figure 3.0-7: SDC Heat Exchanger Energy GOTHIC vs. COPATTA

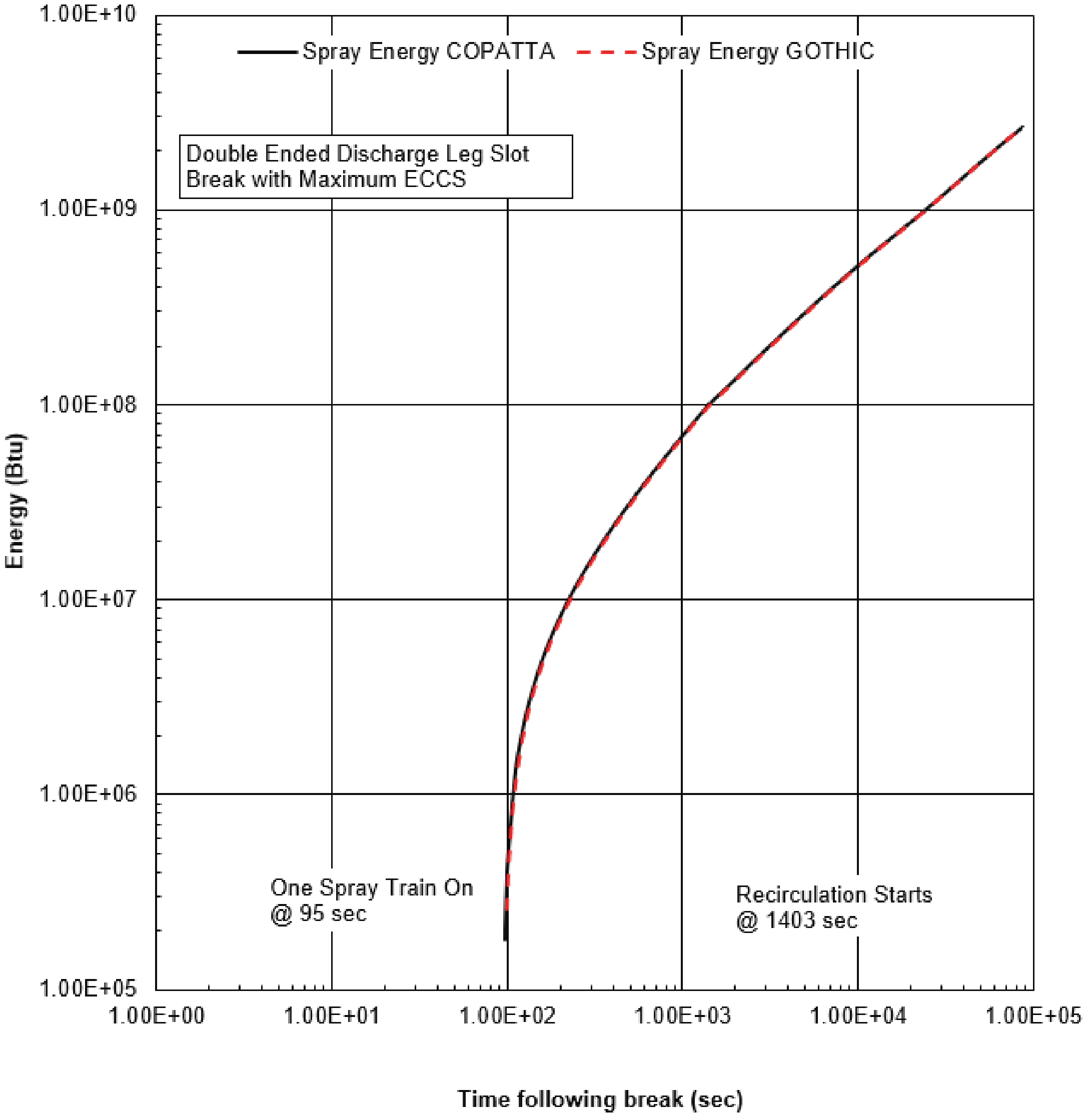


3.2.4.3 <u>Spray Energy Benchmarking</u>

Spray Energy (Btu) is the integrated value of energy transferred from the containment atmosphere to the sump via the containment spray. Comparison of the COPATTA and GOTHIC results for the total integrated spray energy is shown in Figure 3.0-8.

The GOTHIC results are graphically identical to the COPATTA results for the spray energy and are considered valid.

Figure 3.0-8: Spray Energy GOTHIC vs. COPATTA

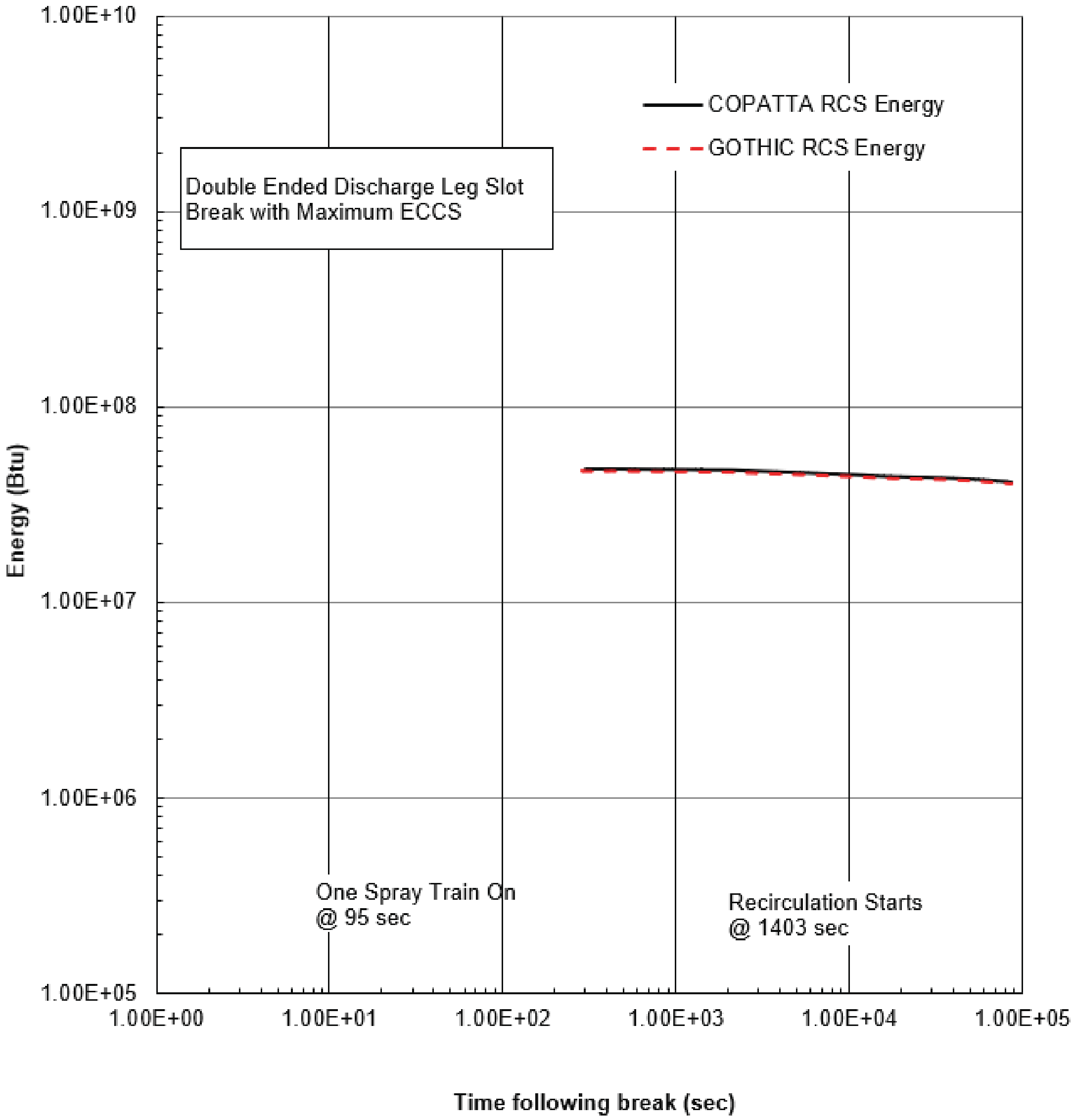


3.2.4.4 Reactor Vessel Energy Benchmarking

Reactor Vessel Energy (Btu) is the amount of energy (relative to 32°F) contained in the RCS water inventory that remains in the RCS Comparison of the COPATTA and GOTHIC results for the reactor vessel energy is shown in Figure 3.0-9.

The GOTHIC results are graphically identical to the COPATTA results for the vessel energy and are considered valid.

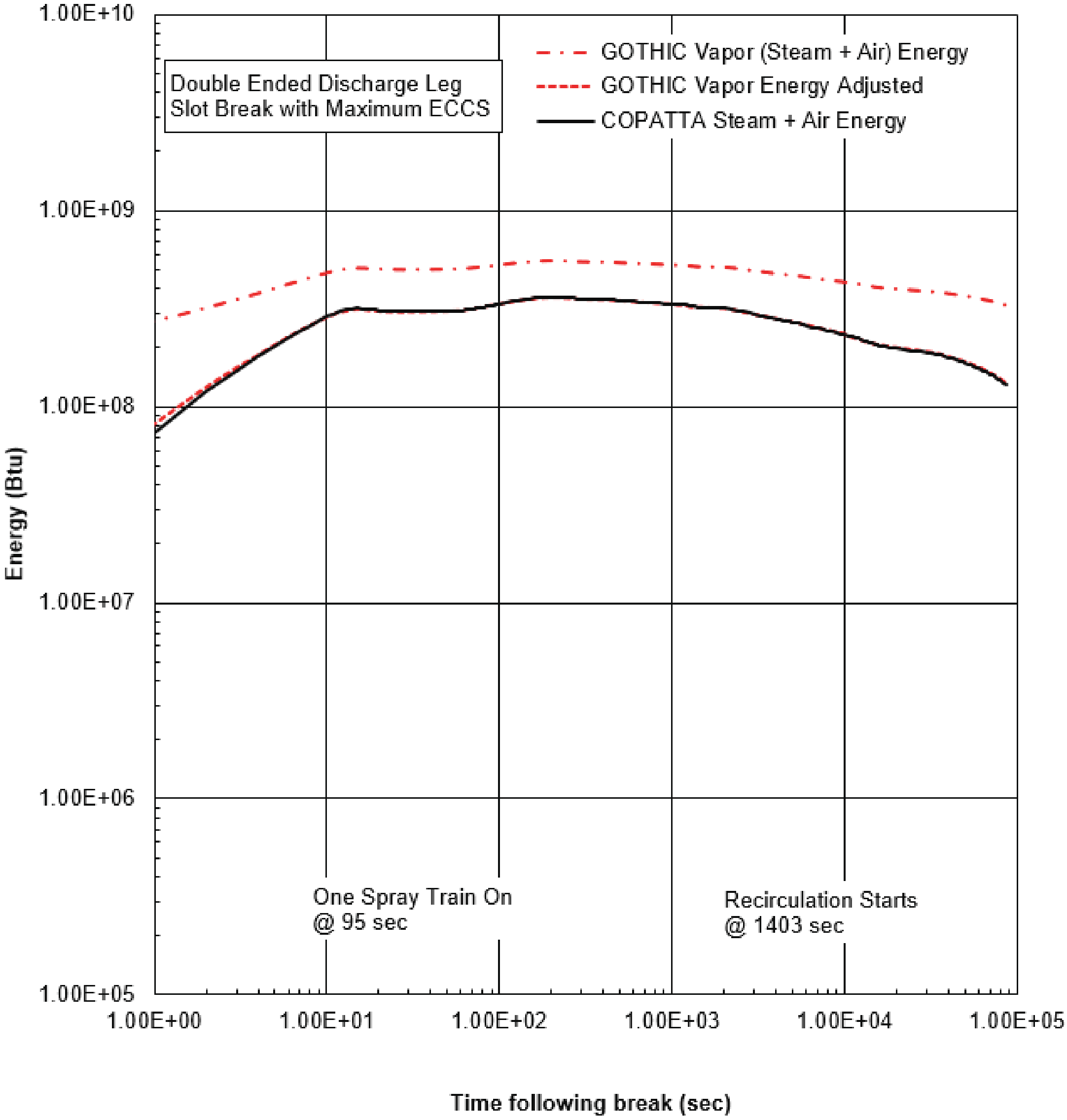
Figure 3.0-9: Reactor Vessel (RCS) Energy GOTHIC vs. COPATTA



3.2.4.5 Steam Energy Benchmarking

Steam Energy (Btu) is the internal energy of the steam relative to 32°F contained in the containment atmosphere. An adjustment is required to compare the COPATTA output to the GOTHIC output. The adjustment is the difference between the energy calculated in GOTHIC at 0.1 seconds and the energy calculated in COPATTA at 0.1 seconds. This adjustment results in a vapor energy profile that is corrected to the initial energy. Figure 3.0-10 shows the steam energy from COPATTA compared to the GOTHIC steam energy. After making the adjustment, the resulting profiles are nearly identical.

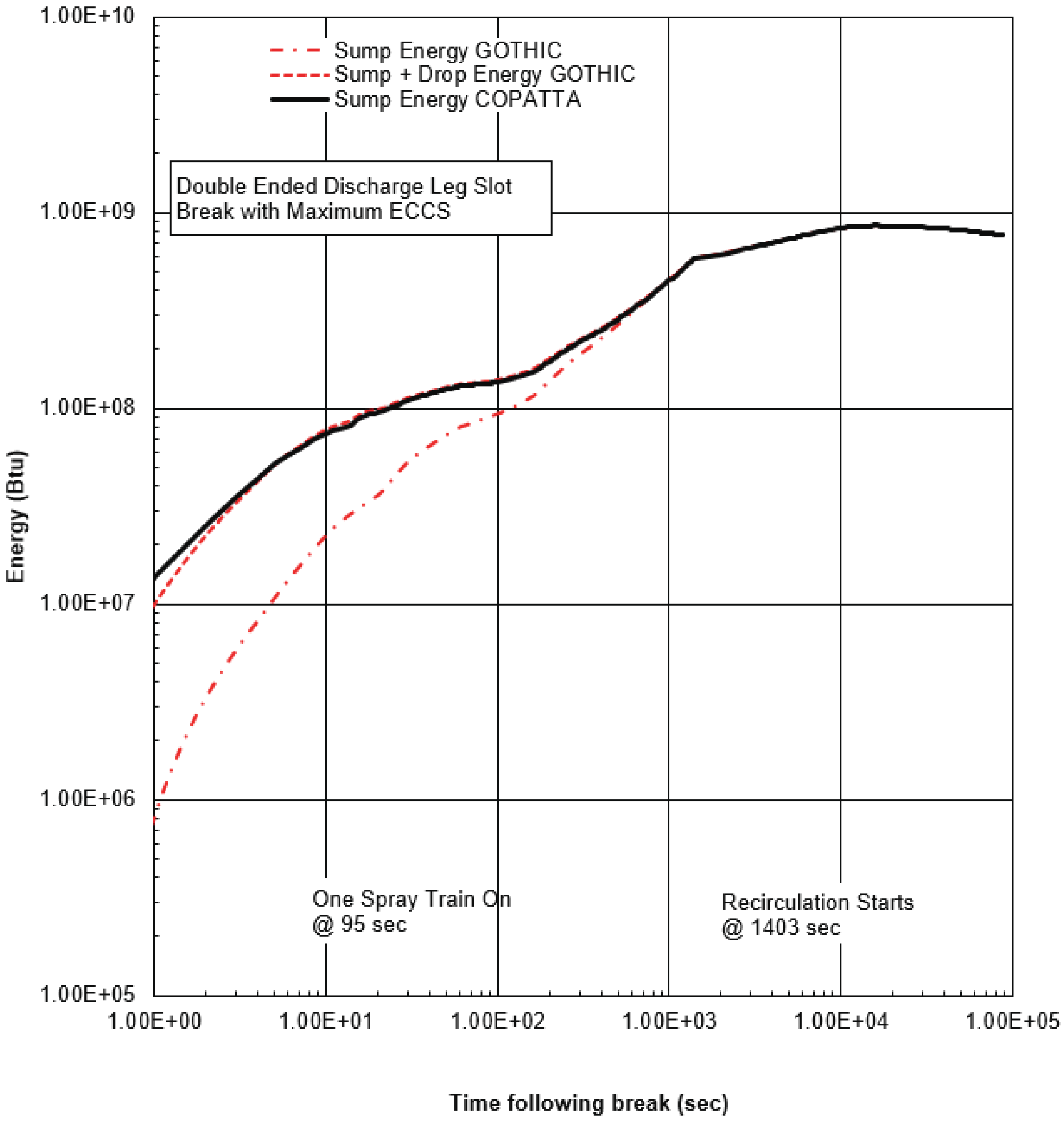
Figure 3.0-10: Steam Energy GOTHIC vs. COPATTA



3.2.4.6 Sump Energy Benchmarking

Sump Energy (Btu) is the internal energy of the water in the sump relative to 32°F Figure 3.0-11 shows the sump energies from COPATTA and GOTHIC. The discrepancy early in time is attributed to the difference in the way the codes treat the liquid phase introduced to the containment atmosphere from the break as discussed in Section 3.2.2. In GOTHIC, the liquid phase emerging from the break is assumed to enter in droplet form and remain suspended in the containment atmosphere for some time before reaching the sump. Note that when combining the GOTHIC sump and droplet energy, the resulting profile is nearly identical to the COPATTA sump energy profile, confirming that the variation is due to the difference in treatment of the liquid phase emerging from the break.

Figure 3.0-11: Sump (Liquid) Energy GOTHIC vs. COPATTA

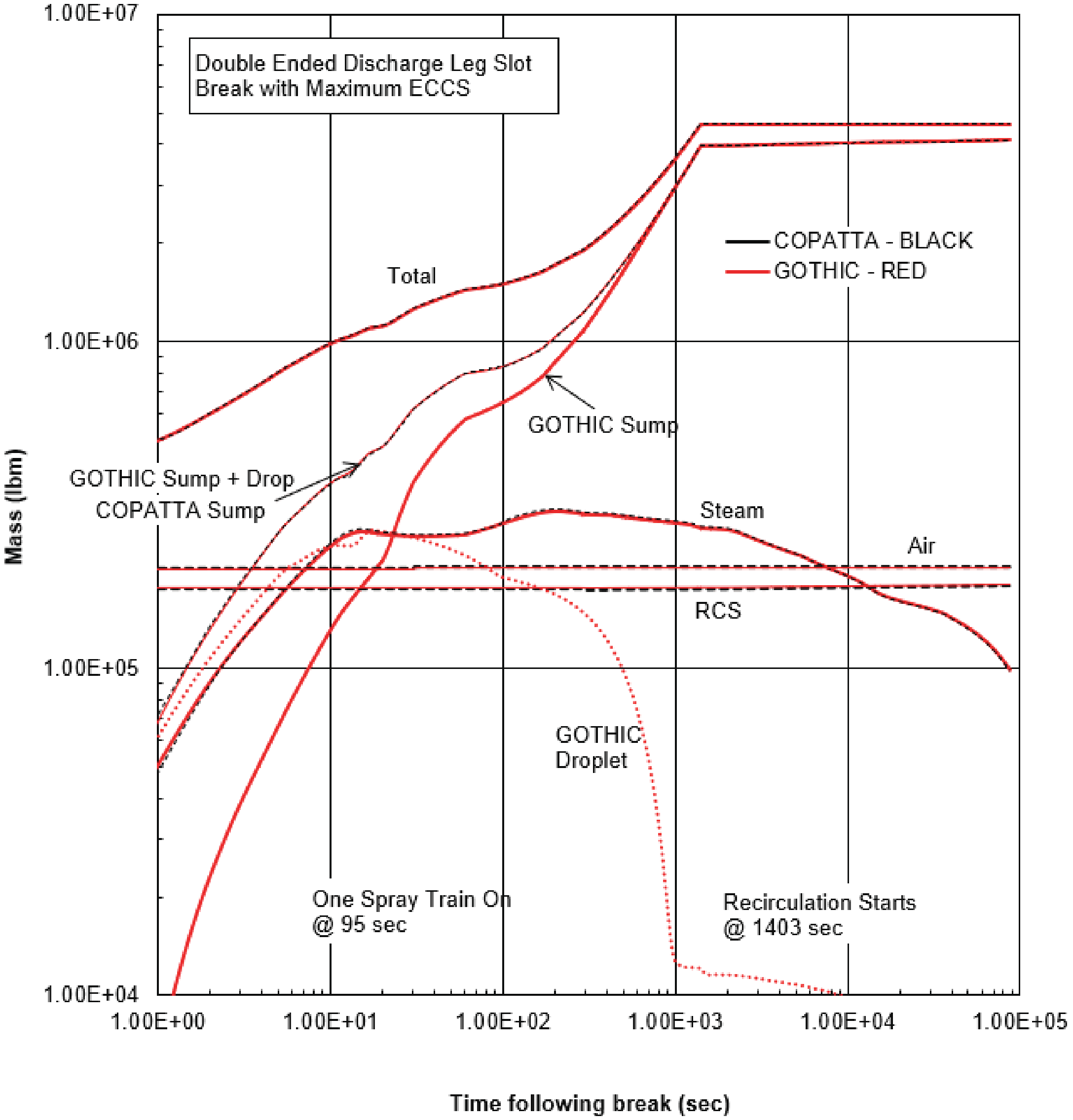


3.2.5 Transient Mass Profiles Benchmarking

Transient mass profiles are developed to trace the mass of steam, water, air, and droplets throughout the model. Figure 3.0-12 shows the water mass inside the RCS and the containment sump, and the steam, air, and water droplet mass in the containment atmosphere throughout the duration of the transient. A check is performed between the total system mass and the summation of the RCS water mass, containment sump water mass, and the containment atmosphere steam, air, and droplet masses. No additional mass enters containment after recirculation begins. Additionally, the sump water mass levels off at approximately 1403 seconds as recirculation begins. The air mass is nearly constant throughout the duration of the transient except for an increase between 30 to 30.5 seconds, which is due to SIT nitrogen venting to containment. The RCS water mass remains constant, indicating a constant level in the RCS. The sump level increases significantly until the beginning of recirculation, indicating a significant amount of water level above the containment floor prior to recirculation. The transient steam mass profile follows the containment pressure profile, indicating a containment pressure response due to variation in containment atmosphere steam mass.

The total mass, steam mass, RCS mass, and air mass are all graphically identical between GOTHIC and COPATTA. The sump mass calculated in COPATTA is equal to the mass of sump water plus droplet water output by GOTHIC and these are graphically identical.

Figure 3.0-12: Transient Mass Profiles GOTHIC vs. COPATTA



4.0 CONCLUSIONS OF THE BENCHMARK

Comparison of containment pressure and temperatures in this benchmark evaluation shows good agreement between the COPATTA and GOTHIC results, as discussed in Section 3.2.

Comparison of containment vapor space and sump liquid temperatures generally shows good agreement, except for a peak temperature spike exhibited in the COPATTA results and the temperature difference for the sump liquid temperature early in the event. These temperature differences are the result of the different approaches taken by the codes in modeling the liquid phase introduced to the containment atmosphere from the break as discussed in Section 3.2.2. The GOTHIC methodology provides a more sophisticated representation of modeling suspended droplets in the vapor space.

Comparison of containment liner temperature profiles shows graphically identical results between the COPATTA and GOTHIC runs.

Comparison of transient energy and mass profiles shows nearly identical results between the COPATTA and GOTHIC runs.

Through benchmarking and sensitivity studies, it is demonstrated that the use of the GOTHIC computer code for the post-LOCA containment pressure and temperature response analysis application produces results that are acceptable. The difference between the current licensing basis COPATTA code and GOTHIC are as expected based on the more sophisticated GOTHIC modeling.

5.0 REFERENCES

5.1 GOTHIC, “Thermal Hydraulic Analysis Package Technical Manual,” Version 8.4(QA), Electric Power Research Institute, Palo Alto, CA, May 2022.

5.2 GOTHIC, “Generation of Thermal Hydraulic Information for Containments,” Version 8.4(QA), Electric Power Research Institute, Palo Alto, CA, 2022.

5.3 GOTHIC, “Thermal Hydraulic Analysis Package User Manual,” Version 8.4(QA), Electric Power Research Institute, Palo Alto, CA, May 2022.

5.4 Westinghouse Topical Report WCAP-16608-NP, Revision 0, “Westinghouse Containment Analysis Methodology”, August 2006, ADAMS Accession No. ML062430591.

5.5 Bechtel Topical Report BN-TOP-3, Revision 1, “Performance and Sizing of Dry Pressure Containments,” January 1974.

ATTACHMENT 6

Technical Analysis of Changes to the SIT Pressure Bands

ATTACHMENT 6

TABLE OF CONTENTS

1.0 INTRODUCTION

Technical Specification (TS) 3.5.1 addresses a Limiting Condition for Operation (LCO) and related Surveillance Requirements (SRs) for the Safety Injection Tanks (SITs) during Operating Conditions. TS 3.5.1 is applicable in MODES 1 and 2, and MODES 3 and 4 with pressurizer pressure equal to or greater than 1837 psia. This technical analysis evaluates the proposed TS 3.5.1 changes that would revise the SR 3.5.1.3 SIT pressure band in each required SIT from "≥ 600 psig and ≤ 625 psig" to "≥ 602 psia and ≤ 675 psia."

Technical Specification 3.5.2 addresses an LCO and related SRs for the SITs during Shutdown Conditions. TS 3.5.2 is applicable in MODES 3 and 4 with pressurizer pressure less than 1837 psia. This technical analysis evaluates the proposed TS 3.5.2 changes that would revise the SR 3.5.2.3 SIT pressure band in each required SIT from "≥ 260 psig and ≤ 625 psig" to "≥ 250 psia and ≤ 675 psia."

The current SR 3.5.1.3 and SR 3.5.2.3 list the nitrogen cover pressure for each SIT as values in units of psig with instrument uncertainties included. The proposed changes will list the SIT nitrogen cover pressure as design values in units of psia with no instrument uncertainties included. Listing the SIT SRs for nitrogen cover pressure in units of psia will match the PVNGS safety analyses that model SIT nitrogen cover pressure in units of psia. The accounting for instrument uncertainties will continue as part of surveillance procedures. All requirements and applicability of the SRs will remain the same.

The proposed changes to the SIT pressure bands impact Emergency Core Cooling System (ECCS) performance analyses, including analyses addressing the consequences of large break Loss of Coolant Accident (LOCA), small break LOCA, and post-LOCA long-term cooling analyses. The proposed changes to the SIT pressure bands also impact the post-LOCA containment pressure and temperature analyses since the nitrogen used to pressurize the four SITs is released to the containment atmosphere during a LOCA event.

Section 2.0 presents a description of the SITs, including their functions.

Section 3.0 presents the evaluations of the impact of the proposed changes in the SR 3.5.1.3 SIT pressure band on the ECCS performance analyses of record (AORs) applicable to a PVNGS reactor with Westinghouse CE16STD and CE16NGF fuel.

Section 4.0 presents the evaluations of the impact of the proposed changes in the SR 3.5.1.3 SIT pressure band on the ECCS performance AORs applicable to a PVNGS reactor with Framatome CE16HTP fuel.

Section 5.0 presents the evaluations of the impact of the proposed changes in the SR 3.5.1.3 SIT pressure band on the limiting post-LOCA containment pressure and temperature response AOR.

Section 6.0 presents an evaluation of the impact of the proposed changes in the SR 3.5.2.3 SIT pressure band.

2.0 SAFETY INJECTION TANK DESCRIPTION

In the unlikely event of a LOCA, the Safety Injection System (SIS), including high pressure and low pressure safety injection pumps and SITs, injects borated water into the Reactor Coolant System (RCS). This provides cooling to limit core damage and fission product release and ensures adequate shutdown margin. The SIS also provides continuous long-term, post-accident cooling of the core by recirculation of borated water from the containment sump.

The PVNGS Nuclear Steam Supply System (NSSS) design features four SITs containing borated water pressurized by a nitrogen cover. Each SIT is connected to its associated reactor coolant cold leg by a separate line containing check valves that isolate the SIT from the RCS during normal operation.

3.0 WESTINGHOUSE CE16STD AND CE16NGF FUEL – IMPACT OF CHANGES TO THE SR 3.5.1.3 SIT PRESSURE BAND

Other than the change in pressure units from psig to psia, no change is being made to the lower limit of the SR 3.5.1.3 SIT pressure band.

The proposed change to the upper limit of the SR 3.5.1.3 SIT pressure band impacts the ECCS performance analyses for Westinghouse CE16STD and CE16NGF fuel. The CE16STD and CE16NGF ECCS performance analyses are comprised of the large break LOCA, the small break LOCA, and the post-LOCA long-term cooling analyses.

3.1 Large Break LOCA

The CE16STD and CE16NGF large break LOCA AORs utilize the NRC approved 1999 EM version of the Westinghouse Electric Company LLC large break LOCA evaluation model for Combustion Engineering designed PWRs (Reference 8.1).

3.1.1 CE16STD Large Break LOCA

Per the CE16STD large break LOCA AOR, both limiting peak cladding temperature (PCT) and limiting peak local oxidation (PLO) cases model the maximum SIT pressure. The core wide cladding oxidation (CWO) cases are not limiting with respect to maximum SIT pressure.

The limiting break for a CE16STD large break LOCA, which results in the closest approach to the 10 CFR 50.46 ECCS performance acceptance criterion for PCT is a 0.6 DEG/PD (Double-Ended Guillotine in the Reactor Coolant Pump Discharge leg). The limiting break, which results in the closest approach to the 10 CFR 50.46 ECCS performance acceptance criterion for PLO is a 0.8 DEG/PD.

The CE16STD large break LOCA AOR limiting PCT and PLO cases have been re-evaluated for an increase in their modeled maximum SIT pressure to 675 psia. The only change to the CE16STD large break LOCA AOR input is the increase in the modeled maximum SIT pressure.

The increase in maximum SIT pressure has a 2°F impact on PCT, resulting in a peak clad temperature of 2108°F. The increase in maximum SIT pressure has a negligible impact on the PLO (i.e., unchanged at 11.9% after roundoff).

As noted in PVNGS UFSAR Section 6.3.3a.1, the CE16STD large break LOCA evaluation model includes a cladding swelling and rupture model that accounts for the effects of changes in core geometry, if such changes are predicted to occur. The ECCS performance analysis demonstrated adequate core cooling even with core geometry changes. The ECCS performance analysis was performed to a point in time where cladding temperatures were decreasing and the RCS was depressurized, thereby precluding any further cladding deformation.

The results of the CE16STD large break LOCA analysis continue to meet 10 CFR 50.46 ECCS performance acceptance criteria 1, 2, 3, and 4, thereby demonstrating the adequacy of the ECCS to support the proposed changes to the SIT pressure bands.

3.1.2 CE16NGF Large Break LOCA

Per the CE16NGF large break LOCA AOR, only the limiting PCT case models the maximum SIT pressure. The PLO and CWO cases are not limiting with maximum SIT pressure.

The limiting break for a CE16NGF large break LOCA, which results in the closest approach to the 10 CFR 50.46 ECCS performance acceptance criterion for PCT is a 0.8 DEG/PD.

The CE16NGF large break LOCA AOR limiting PCT case has been re-evaluated for an increase in its modeled maximum SIT pressure to 675 psia. The only change to the CE16NGF large break LOCA AOR input is the increase in the modeled maximum SIT pressure.

The increase in maximum SIT pressure has a 2°F impact on PCT, resulting in a peak clad temperature of 2132°F.

As noted in PVNGS UFSAR Section 6.3.3b.1, the CE16NGF large break LOCA evaluation model includes a cladding swelling and rupture model that accounts for the effects of changes in core geometry, if such changes are predicted to occur. The ECCS performance analysis demonstrated adequate core cooling even with core geometry changes. The ECCS performance analysis was performed to a point in time where cladding temperatures were decreasing and the RCS was depressurized, thereby precluding any further cladding deformation.

The results of the CE16NGF large break LOCA analysis continue to meet 10 CFR 50.46 ECCS performance acceptance criteria 1, 2, 3, and 4, thereby demonstrating the adequacy of the ECCS to support the proposed changes to the SIT pressure bands.

3.2 Small Break LOCA

The CE16STD and CE16NGF small break LOCA AORs are performed using the Supplement 2 Model (S2M) C-E small break LOCA evaluation model (Reference 8.2, Reference 8.3, and Reference 8.4).

3.2.1 CE16STD Small Break LOCA

The minimum SIT pressure is an input to the CE16STD small break LOCA analysis. The SIT pressure is set at a low value of 200 psia to prevent SIT water injection during a small break LOCA transient. The modeled pressure is less than the proposed revision to Technical Specification SR 3.5.1.3, specifying a SIT pressure band of "≥ 602 psia and ≤ 675 psia." This is a conservative approach by not taking any SIT credit. Consequently, the CE16STD small break LOCA AOR does not necessitate re-analysis for the proposed SIT pressure bands.

The results of the CE16STD small break LOCA analysis continue to meet 10 CFR 50.46 ECCS performance acceptance criteria 1, 2, and 3, thereby demonstrating the adequacy of the ECCS to support the proposed changes to the SIT pressure bands.

3.2.2 CE16NGF Small Break LOCA

The minimum SIT pressure of 602 psia during operating conditions is an input to the CE16NGF small break LOCA analysis, which aligns with the proposed revision to Technical Specification SR 3.5.1.3, specifying a SIT pressure band of "≥ 602 psia and ≤ 675 psia." Consequently, the CE16NGF small break LOCA AOR does not necessitate re-analysis for the proposed SIT pressure bands.

The results of the CE16NGF small break LOCA analysis continue to meet 10 CFR 50.46 ECCS performance acceptance criteria 1, 2, and 3, thereby demonstrating the adequacy of the ECCS to support the proposed changes to the SIT pressure bands.

3.3 Long-Term Cooling

Long-Term Cooling (LTC) is initiated when the core is quenched after a LOCA and is continued until the plant is secured. The objectives of LTC are to maintain the core at safe temperature levels and to avoid the precipitation of boric acid in the core region.

3.3.1 CE16STD and CE16NGF LTC Boric Acid Precipitation Control

The CE16STD and CE16NGF LTC Boric Acid Precipitation Control (BAPC) AOR calculates a SIT liquid mass of 487,812 lbm for a maximum SIT pressure of 651 psia. The SIT liquid mass modeled in the LTC BAPC AOR is rounded up to 488,000 lbm with discretionary conservatism. The LTC BAPC AOR demonstrates that sufficient flushing flow is available after the ECCS is realigned for simultaneous hot leg and cold leg recirculation to preclude reaching the solubility limit for boric acid in the assumed mixing volume.

Increasing the maximum SIT pressure from 651 psia to 675 psia (at 50°F) increases the SIT water density by 0.016%. Since the change in liquid mass is directly proportional to liquid density, the resulting SIT liquid mass for the increased maximum SIT pressure is calculated to be 487,890 lbm (= 487,812 lbm × 1.00016). This is less than the SIT liquid mass modeled in the LTC BAPC AOR. Consequently, the results of the CE16STD and CE16NGF LTC BAPC AOR remain applicable for a maximum SIT pressure of 675 psia.

The results of the CE16STD and CE16NGF LTC BAPC AOR will continue to meet 10 CFR 50.46 ECCS performance acceptance criterion 5, thereby demonstrating the adequacy of the ECCS to support the proposed changes to the SIT pressure bands.

3.3.2 CE16STD and CE16NGF LTC Decay Heat Removal

The CE16STD and CE16NGF LTC Decay Heat Removal (DHR) AOR models the rounded-up SIT mass of 488,000 lbm modeled in the LTC BAPC AOR. The LTC DHR AOR results meet 10 CFR 50.46 LTC performance acceptance criterion 5 requiring that after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

As noted in the Section 3.3.1, the increase in SIT water density associated with the increase in maximum SIT pressure results in a SIT mass of less than 488,000 lbm. Consequently, the results of the CE16STD and CE16NGF LTC DHR AOR remain applicable for a maximum SIT pressure of 675 psia.

The results of the CE16STD and CE16NGF LTC DHR AOR will continue to meet 10 CFR 50.46 ECCS performance acceptance criterion 5, thereby demonstrating the adequacy of the ECCS to support the proposed changes to the SIT pressure bands.

4.0 FRAMATOME CE16HTP FUEL– IMPACT OF CHANGES TO THE SR 3.5.1.3 SIT PRESSURE BAND

Other than the change in pressure units from psig to psia, no change is being made to the lower limit of the SR 3.5.1.3 SIT pressure band.

The proposed change to the upper limit of the SR 3.5.1.3 SIT pressure band impacts the ECCS performance analyses for Framatome CE16HTP fuel. The CE16HTP ECCS performance analyses are comprised of the realistic large break LOCA, the small break LOCA, and the post-LOCA long-term cooling analyses.

4.1 Large Break LOCA

The CE16HTP Realistic Large Break LOCA (RLBLOCA) AOR utilizes the Framatome NRC-approved RLBLOCA methodology documented in EMF-2103P-A (Reference 8.5). This methodology has been reviewed and approved by the NRC to perform RLBLOCA analyses. However, some differences from the approved RLBLOCA methodology were included in the PVNGS RLBLOCA analysis. These differences were present in the previously NRC-approved PVNGS RLBLOCA analysis as noted in the NRC Safety Evaluation for PVNGS Operating Licenses Amendment 212 implementation of Framatome CE16HTP fuel (Reference 8.6).

The CE16HTP RLBLOCA AOR has been re-performed to support the SIT pressure band expansion and demonstrate the adequacy of the ECCS for PVNGS. The only change to the CE16HTP RLBLOCA AOR input is modeling the proposed SIT pressure band. The CE16HTP RLBLOCA analysis is summarized in Framatome Licensing Report ANP-4101 (Reference 8.7). The CE16HTP RLBLOCA analysis assumes a SIT pressure band of ≥ 602 psia and ≤ 675 psia, which aligns with the proposed revision to Technical Specification SR 3.5.1.3.

The key RLBLOCA acceptance criterion variables are: (1) peak clad temperature (PCT), (2) total hydrogen production from the cladding oxidation, and (3) maximum local oxidization (MLO) of cladding. The fraction of total hydrogen generated is not calculated; however, it is conservatively bounded by the calculated core wide oxidation (CWO).

The proposed SIT pressure band has a 39°F impact on PCT, resulting in a peak clad temperature of 1791°F. The proposed SIT pressure band has a -0.40% impact on MLO, resulting in a maximum local oxidization of 1.97 percent. The proposed SIT pressure band has a -0.006% impact on CWO, resulting in a total core wide oxidation of 0.014 percent.

As noted in PVNGS UFSAR Section 6.3.3c.1, the CE16HTP realistic large break LOCA evaluation model includes a cladding swelling and rupture model that accounts for the effects of changes in core geometry, if such changes are predicted to occur. The ECCS performance analysis demonstrated adequate core cooling even with core geometry changes. The ECCS performance analysis was performed to a point where cladding temperatures were decreasing and the RCS was depressurized, thereby precluding any further cladding deformation.

The results of the CE16HTP realistic large break LOCA analysis continue to meet 10 CFR 50.46 ECCS performance acceptance criteria 1, 2, 3, and 4, thereby demonstrating the adequacy of the ECCS to support the proposed changes to the SIT pressure bands.

4.2 Small Break LOCA

The CE16HTP small break LOCA AOR, summarized in Framatome Licensing Report ANP-3640 (Reference 8.8), adheres to the NRC-approved S-RELAP5 methodology detailed in NRC-approved S-RELAP5 methodology as described in EMF-2328(P)(A) (Reference 8.9) and as modified by EMF-2328(P)(A) (Reference 8.10). ANP-3640 concludes that the small break LOCA scenario satisfies 10 CFR 50.46 ECCS performance acceptance criteria 1 through 4 related to peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, and coolable geometry.

The CE16HTP small break LOCA AOR assumes a SIT minimum pressure of 602 psia, which aligns with the proposed revision to Technical Specification SR 3.5.1.3, specifying a SIT pressure band of "≥ 602 psia and ≤ 675 psia." Consequently, the CE16HTP small break LOCA AOR does not necessitate re-analysis for the proposed SIT pressure bands.

The results of the CE16HTP small break LOCA analysis continue to meet 10 CFR 50.46 ECCS performance acceptance criteria 1, 2, 3, and 4, thereby demonstrating the adequacy of the ECCS to support the proposed changes to the SIT pressure bands.

4.3 Long-Term Cooling

As noted in the NRC Safety Evaluation for PVNGS Operating Licenses Amendment 212 implementation of Framatome CE16HTP fuel (Reference 8.6), the CE16NGF LTC BAPC and DHR analyses are applicable to Framatome HTP fuel.

As noted in Section 3.3.1 and Section 3.3.2, the results of the LTC BAPC and DHR AORs remain applicable for a maximum SIT pressure of 675 psia. The results of the CE16NGF LTC AORs will continue to meet 10 CFR 50.46 ECCS performance acceptance criterion 5.

Therefore, CE16HTP LTC fuel performance will continue to meet 10 CFR 50.46 ECCS performance acceptance criterion 5, thereby demonstrating the adequacy of the ECCS to support the proposed changes to the SIT pressure bands.

5.0 POST-LOCA CONTAINMENT PRESSURE AND TEMPERATURE RESPONSE – IMPACT OF CHANGES TO THE SR 3.5.1.3 SIT PRESSURE BAND

Other than the change in pressure units from psig to psia, no change is being made to the lower limit of the SR 3.5.1.3 SIT pressure band.

The proposed change to the upper limit of the SR 3.5.1.3 SIT pressure band impacts the post-LOCA containment pressure and temperature analyses since the nitrogen used to pressurize the four SITs is released to the containment atmosphere during a LOCA event.

As described in the PVNGS UFSAR, PVNGS currently uses the Bechtel Containment Pressure and Temperature Transient Analysis (COPATTA) code as part of the methodology for the calculation of the containment pressure and temperature response to various postulated pipe breaks. This methodology is not addressed in the PVNGS Technical Specifications. The proposed changes also include use of the industry standard Generation of Thermal Hydraulic Information for Containments (GOTHIC) code as a replacement to COPATTA as part of the methodology to perform calculations of the containment pressure and temperature response to various postulated pipe breaks.

The double-ended discharge leg slot (DEDLS) break LOCA with maximum ECCS (i.e., Safety Injection [SI]) flow was identified as the pipe break with the highest peak pressure. An evaluation has been performed using the GOTHIC code to document the impact of the proposed increase in the upper limit of the SIT pressure bands to 675 psia for the DEDLS break LOCA with maximum SI flow.

The GOTHIC benchmark run documented in Enclosure Attachment 5 is used as a base model for the GOTHIC evaluation. The following model enhancements have been made to the base model:

1. Updated the modeling of SIT cover gas to nitrogen at 675 psia and modeled nitrogen release from all four SITs. The COPATTA and GOTHIC benchmark runs documented in Enclosure Attachment 5 both model air injection at 652 psia from one SIT.
2. Updated the modeling of the Shutdown Cooling (SDC) heat exchanger to use the more intricate GOTHIC water-to-water heat exchanger model to account for the overall heat transfer coefficient (U-factor) as a function of tube-side flow and tube-side inlet temperature. The COPATTA and GOTHIC benchmark runs documented in Enclosure Attachment 5 both model a simplistic SDC heat exchanger using a constant U-factor with shell-and tube heat exchanger geometry.
3. Updated the Containment Spray (CS) mass flow rate model to a volumetric flow rate model. The COPATTA and GOTHIC benchmark runs documented in Enclosure Attachment 5 both model the CS recirculation phase volumetric flow rate converted to a mass flow rate based on a single containment sump water temperature.
4. Included the energy added to the fluid by the SI and CS pumps. The pump energy accounts for local temperature rise across the pump due to pump inefficiency and also the energy imparted to the fluid via an increase in pressure. The COPATTA and GOTHIC benchmark runs documented in Enclosure Attachment 5 do not address this energy contribution.
5. Modeled the liquid phase of the break effluent as a uniform drop field with a 100 microns (= 0.00394 inches) drop diameter consistent with other NRC approved containment analyses performed in GOTHIC such as WCAP-15667 (Reference 8.11 for the Kewaunee Nuclear Power Plant), WCAP-16219-NP (Reference 8.12 for the Prairie Island Nuclear Generating Plants), and as recommended by the GOTHIC User’s Manual (Reference 8.13, Section 27.5).

The proposed increase in the upper limit of the SR 3.5.1.3 SIT pressure band to 675 psia was evaluated for potential impact on containment peak pressure due to the resultant increase in the mass and energy release from the SITs. The evaluation determined that the total amount of injected mass and condensed steam is expected to remain the same and therefore there is a negligible impact on the integrated mass and energy released at the time of peak containment pressure.

The NRC Safety Evaluations addressing the Kewaunee and Prairie Island use of GOTHIC state that the NRC staff concludes that the mist formation model shall not be used for licensing calculations (Reference 8.14 and Reference 8.15). Consistent with the GOTHIC benchmark run documented in Enclosure Attachment 5, the PVNGS post-LOCA containment pressure and temperature analysis does not use the mist formation model. The PVNGS analysis uses the Tagami and Uchida condensing heat transfer correlations as described in the in PVNGS UFSAR Section 6.2.1.1.3.1.

The GOTHIC evaluation calculates a peak containment pressure of 71.46 psia, which is reported with inclusion of discretionary margin as 72.05 psia (equivalent to 57.85 psig based on a local atmospheric pressure of 14.2 psia). The reported peak containment pressure is equivalent to the value reported in the PVNGS UFSAR Table 6.2.1-9. This value is conservatively rounded to 58.0 psig for the Technical Specification 5.5.16b for the peak containment pressure (Pa) for the design basis loss of coolant accident. As such, the value of Pa in Technical Specification 5.5.16b remains unchanged, and is less than the containment design pressure of 60 psig, as specified in TS 5.5.16b for the Containment Leakage Rate Testing Program.

The GOTHIC evaluation calculates a peak containment vapor temperature of 280.06°F, which is reported with inclusion of discretionary margin as 308.41°F. The reported peak containment vapor temperature is equivalent to the value reported in the PVNGS UFSAR Table 6.2.1-9.

The GOTHIC evaluation calculates the containment pressure dropping to 34.2 psia at 24 hours, which is reported with inclusion of discretionary margin as 35.59 psia (equivalent to 21.39 psig based on a local atmospheric pressure of 14.2 psia). The calculated containment pressure of 21.39 psig at 24 hours is less than 50 percent of the peak calculated pressure of 57.85 psig as required by Standard Review Plan 6.2.1.1.A (Reference 8.16).

In compliance with 10 CFR 50 Appendix A General Design Criterion (GDC) 16, the GOTHIC evaluation confirms that in the event of the limiting DEDLS break LOCA with maximum ECCS flow, the Safety Injection System (SIS) and Containment Spray System (CSS) are actuated, cool the reactor core, and return the containment to near atmospheric pressure. The containment, SIS, CSS, and containment isolation system ensure the functional capability of containing any uncontrolled release of radioactivity.

In compliance with 10 CFR 50 Appendix A GDC 50, the GOTHIC evaluation confirms that the reactor containment structure and its internal compartments, including access openings, penetrations, and the containment heat removal system, accommodate the calculated pressure and temperature conditions resulting from the limiting DEDLS break LOCA with maximum ECCS flow, without exceeding the design leakage rate and with a sufficient margin.

The results of the post-LOCA containment pressure and temperature analysis will continue to meet 10 CFR 50 Appendix A General Design Criteria 16 and 50, thereby demonstrating the adequacy of the containment structure design to support the proposed changes to the

SIT pressure bands.

6.0 IMPACT OF CHANGES TO THE SR 3.5.2.3 SIT PRESSURE BAND

The analysis for ECCS – Shutdown uses minimum SIT pressure as the limiting input. The lower limit of the SR 3.5.2.3 SIT pressure band remains unchanged, except for the switch from psig to psia units. This lower limit is determined by the SIT water volume requirements outlined in SR 3.5.2.2. The minimum nitrogen cover pressure ensures that sufficient SIT water volume can be injected into the RCS during a LOCA event in Mode 3 and Mode 4 with a pressurizer pressure less than 1837 psia. This minimum SIT pressure remains unaffected by the increase in the maximum nitrogen cover pressure, as the SIT water volume requirements are not altered. Therefore, the 10 CFR 50.46 ECCS performance acceptance criteria will continue to be met during Shutdown conditions.

The upper limit of the SR 3.5.2.3 SIT pressure band is being adjusted to align with the proposed analytical value of 675 psia to be specified in SR 3.5.1.3. This adjustment maintains consistency with the historical precedent of setting identical upper limit values in SR 3.5.1.3 and SR 3.5.2.3. This change improves reactor operator human performance by eliminating the need to adjust SIT pressures during mode transitions since the upper nitrogen cover pressure limit remains the same.

7.0 RESULTS AND CONCLUSIONS

7.1 ECCS Performance Criteria

As demonstrated in Section 3.0, Section 4.0, and Section 6.0, the results of the CE16STD, CE16NGF, and CE16HTP ECCS performance analyses will continue to meet 10 CFR 50.46 ECCS performance acceptance criteria 1, 2, 3, 4, and 5, thereby demonstrating the adequacy of the ECCS to support the proposed changes to the SIT pressure bands.

7.2 Containment Structure Performance Criteria

As demonstrated in Section 5.0, the results of the post-LOCA containment pressure and temperature analysis will continue to meet 10 CFR 50 Appendix A General Design Criteria 16 and 50, thereby demonstrating the adequacy of the containment structure design to support the proposed changes to the SIT pressure bands.

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9.0 LIST OF TYPICAL ACRONYMS AND ABBREVIATIONS

Acronym	Meaning
ADAMS	NRC Agencywide Document Access Management System
AOR	Analysis of Record
BAPC	Boric Acid Precipitation Control
CE16NGF	Westinghouse supplied Next Generation Fuel
CE16HTP	Framatome supplied High Thermal Performance Fuel
CE16STD	Westinghouse supplied Standard fuel
CFR	Code of Federal Regulations
COPATTA	Containment Pressure and Temperature Transient Analysis (code)
CS	Containment Spray
CSS	Containment Spray System
CWO	Core-Wide Oxidation
DEDLS	Double-Ended Discharge Leg Slot
DEG/PD	Double-Ended Guillotine in the Reactor Coolant Pump Discharge leg
DHR	Decay Heat Removal
ECCS	Emergency Core Cooling System
EM	Evaluation Model (or Methodology)
GDC	General Design Criterion (or Criteria)
GOTHIC	Generation of Thermal Hydraulic Information for Containments (code)
LOCA	Loss of Coolant Accident
LTC	Long-Term Cooling
M5®	Designation for a Framatome Fuel Rod Cladding Material
MLO	Maximum Local Oxidation
NRC	United States Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PCT	Peak Cladding Temperature
PLO	Peak Local Oxidation

Acronym	Meaning
psia	pounds per square inch, absolute
psig	pounds per square inch, gauge
PVNGS	Palo Verde Nuclear Generating Station
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RLBLOCA	Realistic Large Break LOCA
RWT	Refueling Water Tank
SDC	Shutdown Cooling
SI	Safety Injection
SIS	Safety Injection System
SIT	Safety Injection Tank
SR	Surveillance Requirement
S2M	Supplement 2 Model
TS	Technical Specification
U-factor	overall heat transfer coefficient
UFSAR	Updated Final Safety Analysis Report
NRC	United States Nuclear Regulatory Commission
WCAP	Westinghouse Commercial Atomic Power
WEC	Westinghouse Electric Company

ATTACHMENT 7

Framatome Licensing Report

ANP-4101NP Revision 0 “Palo Verde Units 1, 2, and 3 Large Break LOCA Analysis Licensing Report”

[NON-PROPRIETARY VERSION]

Palo Verde Units 1, 2, and 3 Large Break LOCA Analysis

ANP-4101NP
Revision 0

Licensing Report

May 2024

FRAMATOME INC. TRADEMARKS

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Nature of Changes

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

Contents

List of Tables

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Nomenclature

Acronym	**Definition**
ASI	Axial Shape Index
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
EDG	Emergency Diesel Generator
EM	Evaluation Model
EMDAP	Evaluation Model Development and Assessment Process
F_r	Nuclear Enthalpy Rise Factor/Radial Peaking Factor
F_Q	Total Peaking Factor/Global Peaking Factor
Framatome	Framatome Inc.
FSRR	Fuel Swell Rupture and Relocation
Gd_2O_3	Gadolinia or Gad
GDC	General Design Criteria
HPSI	High Pressure Safety Injection
HTP	High Thermal Performance (grid spacer)
HMP	High Mechanical Performance (grid spacer)
LBLOCA	Large Break Loss-of-Coolant Accident
LCO	Limiting Condition of Operation
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
LPSI	Low Pressure Safety Injection

Acronym	**Definition**
MLO	Maximum Local Oxidation
No-LOOP	No Loss of Offsite Power
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PCT	Peak Clad Temperature
PIRT	Phenomena Identification and Ranking Table
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
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
TS	Technical Specification
UTL	Upper Tolerance Limit

1.0 INTRODUCTION

This report summarizes the Realistic Large Break Loss-of-Coolant Accident (RLBLOCA) analysis for Palo Verde Nuclear Generating Station (PVNGS). The purpose of the RLBLOCA analysis is to support the SIT pressure band expansion and demonstrate the adequacy of the Emergency Core Cooling System (ECCS) for PVNGS. This analysis was performed in accordance with the U.S. Nuclear Regulatory Commission (NRC)-approved S-RELAP5-based methodology described in Reference 1.

PVNGS Units 1, 2, and 3 are 2x4-loop, Combustion Engineering (CE)-designed Pressurized Water Reactors (PWRs) with a large dry containment. The Framatome Advanced CE16 Fuel Design with M5$_{\text{Framatome}}$ 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$_{\text{Framatome}}$ MONOBLOC guide tube design, M5$_{\text{Framatome}}$ 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$_2$O$_3$ rods included.

The analysis supports plant operation at a core power level of 4070 MWt (including measurement uncertainty), a total linear heat generation rate (LHGR) of 13.1 kW/ft, a radial peaking factor (Fr) of 1.81 (includes uncertainty), and up to 10% steam generator (SG) tube plugging per SG.

The Framatome RLBLOCA methodology addresses typical operational ranges or technical specification (TS) limits (whichever is applicable) regarding [ ] The analysis explicitly analyzes fresh and once-burned fuel assemblies. The plant parameter specification for this analysis is provided in Table 4-1. The analysis uses the Fuel Swelling, Rupture, and Relocation (FSRR) model to determine if cladding rupture occurs and evaluates the consequences of FSRR on the transient response.

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2.0 SUMMARY OF RESULTS

The UTL results providing 95/95 simultaneous coverage from this evaluation meet the 10 CFR 50.46(b) criteria with a PCT of 1791°F, a maximum local oxidation of 1.97 percent and a total core-wide oxidation of 0.014 percent. The PCT of 1791°F occurred in a fresh UO_2 rod. The results of the analysis demonstrate the adequacy of the ECCS to support the 10 CFR 50.46(b) Criteria 1-3 discussed in Section 3.0.

3.0 DESCRIPTION OF ANALYSIS

3.1 *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 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 in separate 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 imposed 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% (Section 3.3.3, NRC Final Safety Evaluation (SE) for Reference 1). The limitation is addressed in Table 3-1.

3.2 *Description of LBLOCA Event*

A Large Break Loss-of-Coolant Accident (LBLOCA) is initiated by a postulated rupture of the Reactor Coolant System (RCS) primary piping. The most challenging break location is in the cold leg piping between the reactor coolant pump and the reactor. The plant is assumed to be operating normally at full power prior to the accident and the break is assumed to open instantaneously. A worst case single-failure is also assumed to occur during the accident. The single-failure for this analysis, as defined in the EM, conservatively assumes the loss of one emergency diesel generator (EDG), which takes out one train of ECCS pumped injection without the loss of containment spray.

The LBLOCA event is typically described in three phases: blowdown, refill, and reflood. Following the initiation of the break, the blowdown phase is characterized by a sudden depressurization from operating pressure down to the saturation pressure of the hot leg fluid. For larger cold leg breaks, an immediate flow reversal and stagnation occurs in the core due to flow out the break, which causes the fuel rods to pass through critical heat flux (CHF), usually within one second following the break. Following this initial rapid depressurization, the RCS depressurizes at a more gradual rate. Reactor trip and emergency injection signals occur when either the low pressurizer pressure setpoint or the containment high-pressure setpoint are reached. However, for LBLOCA, reactor trip and scram are not modeled, and reactor shutdown is accomplished by the negative reactivity feedback produced by the coolant voiding in the core region. During blowdown, core cooling is supported by the natural evolution of the RCS flow pattern as driven by the break flow.

When the system pressure falls below the SIT pressure, flow from the SIT is injected into the cold legs ending the blowdown period and initiating the refill period. Once the system pressure falls below the respective shutoff heads of the safety injection systems and the system startup time delays are met, flow from the pumped safety injection systems is injected into the RCS. While some of the ECCS flow bypasses the core and goes directly out of the break, the downcomer and lower plenum gradually refill until the mixture in the lower head and lower plenum regions reaches the bottom of the active core and the reflood period begins. Core cooling is supported by the natural evolution of the RCS flow pattern as driven by the break flow and condensation in the RCS promoted by safety injection. Towards the end of the refill period, heat transfer from the fuel rods is relatively low, steam cooling and rod-to-rod radiation being the primary mechanisms of core heat removal.

Once the lower plenum is refilled to the bottom of the fuel rod heated length, refill ends and the reflood phase begins. Substantial ECCS fluid is retained in the downcomer during refill. This provides the driving head to move coolant into the core. As the mixture level moves up the core, steam is generated and liquid is entrained, providing cooling in the upper core regions. The two-phase mixture extends into the upper plenum and some liquid may de-entrain and flow downward back into the cooler core regions. The remaining entrained liquid passes into the steam generators where it vaporizes, adding to the steam that must be discharged through the break and out of the system. The difficulty of venting steam is, in general, referred to as steam binding. It acts to impede core reflood rates. With the initiation of reflood, a quench front starts to progress up the core. With the advancement of the quench front, the cooling in the upper regions of the core increases, eventually arresting the rise in fuel rod surface temperatures. Later the core is quenched and a pool cooling process is established that can maintain the cladding temperature near saturation, so long as the ECCS makes up for the core boil off.

3.3 *Description of Analytical Models*

The NRC-approved RLBLOCA methodology is documented in EMF-2103(P)(A) (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 Loss-of-Coolant Accident (LOCA) analysis.

The Framatome RLBLOCA methodology evaluation model 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 approved Reference 1 LBLOCA methodology were included in this analysis, as described below. These differences were present in the previously NRC-approved PVNGS RLBLOCA analyses (Reference 5) and are repeated here for consistency.

- [

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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 fission product 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 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. Specific parameters are discussed in Section 3.6. Additionally, the COPERNIC code provides initial conditions for the S-RELAP5 fuel models.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by inducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood is computed continuously using S-RELAP5. The COPERNIC module in S-RELAP5 provides the fuel properties that are needed to calculate the fuel rod thermal response during the transient. 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 values for the first three criteria of 10 CFR 50.46(b) 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(b) are met with a probability higher than 95 percent with 95 percent confidence.

3.4 *GDC-35 Limiting Condition Determination*

General Design Criteria (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. [

]

3.5 *Overall Statistical Compliance to Criteria*

[

]

3.6 *Plant Description*

The plants analyzed is the Palo Verde Unit (Units 1 through 3 are supported), CE-designed PWR, which has 2X4 loops arrangement. All three units at PVNGS are CE-designed PWRs with two hot legs, four cold legs, two U-tube steam generators, and an RCP in each cold leg. The Reactor Coolant System (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 RCS, reactor vessel, pressurizer, and ECCS are explicitly modeled in the S-RELAP5 model. The ECCS includes a SIT path and a LPSI/HPSI (LPSI feeds only 2 RCS loops as described above) path per RCS loop. The HPSI and LPSI feed into the SIT line which connects to each cold leg pipe downstream of the RCP discharge. The ECCS pumped injection is modeled as a table of flow versus backpressure. The model also includes the secondary-side steam generators which are instantaneously isolated (closed MSIV and feedwater trip) at the time of the break. A symmetric steam generator tube plugging level of 10 percent per steam generator is assumed.

The primary and secondary coolant systems for PVNGS were nodalized consistent with code input guidelines in Appendix A of Reference 1. Representative system nodalization details are shown in Figure 3-1 through Figure 3-3.

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 Section 3.3 many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges are sampled. Values for process or operational parameters, including ranges of sampled process parameters, and fuel design parameters used in this analysis are given in Table 4-1. A summary of the uncertainties used in the analysis is presented in Table 4-2. [

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3.7 *Safety Evaluation 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 limitations and conditions from the NRC SE for EMF-2103(P)(A), Revision 3 (Reference 1), are addressed in Table 3-1.

Table 3-1
EMF-2103(P)(A), Revision 3, SE Limitations Evaluation

Limitations (Section 4.0 of the SE in Reference 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 or LOCA blowdown loads, or for evaluating the effects of reactor coolant system boric acid transport. Such evaluations would be considered separate methods.	The 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 Combustion Engineering-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 was performed with $M5_{\text{Framatome}}$ cladding material.

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 were followed completely for the analysis described in this notebook. [ ]

Limitations (Section 4.0 of the SE in Reference 1)		Response
4		[ ]
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 burnup values applied in the analysis do not exceed the rod average burnup 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. Such a request would be tendered by a letter from the NRC to AREVA identifying the newly available data and requesting an update to the model, or an assessment to demonstrate that such an update is not needed.	The analysis uses the approved EMF-2103P-A, Revision 3 relocation packing factor application. [ ]

Limitations (Section 4.0 of the SE in Reference 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 C-P correlation, this limit shall be reduced to 13 percent, inclusive of pre-transient oxide layer thickness.	For this analysis the MLO UTL is less than 13%.
8	In conjunction with Limitation [7] above, C-P 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 (Section 4.0 of the SE in Reference 1)		**Response**
10	[ ]	This analysis uses [ ]
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.	The present analysis is the first statistical application intended to become the AOR for this plant.

Figure 3-1
Representative Primary System Noding

Figure 3-2
Representative Secondary System Noding

Figure 3-3
Representative Reactor Vessel Noding

4.0 RLBLOCA ANALYSIS

4.1 *RLBLOCA Results*

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For a simultaneous coverage/confidence level of 95/95, the UTL values, [ ] are a PCT of 1791°F, an MLO of 1.97 percent, and a CWO of 0.014 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.

A summary of the major input parameters for the demonstration case is provided in Table 4-4. The sequence of event times for the demonstration case is provided in Table 4-5. [

]

The analysis scatter plots for the case set are shown in Figure 4-1 through Figure 4-5. Figure 4-1 shows linear scatter plots of the key parameters sampled for all cases. 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 the values used is appropriate and within the uncertainty ranges listed in Table 4-2.

Figure 4-2 and Figure 4-3 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 4-4 and Figure 4-5, respectively.

Figure 4-2 shows about 57% of LOOP cases have PCTs during the blowdown phase (PCT time less than approximately 30 seconds). The next cluster of PCTs occurs during the early to late reflood period. Blowdown PCT cases are dominated by rapid RCS depressurization and stored energy content. Early reflood PCT cases are dominated by decay heat removal capacity. In general, plants with high pressure SITs inject early in the transient when the break flow is still high. The high pressure and high break flow drive some of this fluid to bypass the core, delaying the progression of the core reflood. This effect is counterbalanced by the relatively high volume of the SITs, which extends the injection time and allows the core recovery to begin relatively early. This results in cases with higher PCTs in the blowdown phase of the transient and relatively lower PCTs during reflood and late reflood.

The high PCT cases in the upper part of Figure 4-2 are mainly influenced by the area of the break. This is demonstrated in Figure 4-3 which shows a general increasing trend in PCT with break size and similarly smaller break sizes result in lower PCTs. From all sampled parameters, the break size is a dominant effect on PCT because of its influence on the rate of primary depressurization.

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

The demonstration case is a blowdown peak case with a PCT timing of 7.0 seconds. Figure 4-6 through Figure 4-17 show key parameters from the S-RELAP5 calculations for the demonstration case. The transient progression for the demonstration case follows that described in Section 3.2.

[

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4.2 *Conclusions*

This report describes and provides results from the RLBLOCA analysis for the Palo Verde Nuclear Generating Station with the Framatome Advanced CE16 HTP Fuel Design with $\text{M5}_{\text{Framatome}}$ cladding. The application of the 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 1791°F, an MLO of 1.97 percent, and a CWO of 0.014 percent.

Table 4-1
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	150.0 in
	g) Gd_2O_3 concentrations	2, 4, 6, 8 weight-percent
	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) Nominal reactor power	4070 MWt (1)
	b) LHGR	13.1 kW/ft
	c) F_Q	2.289 (2)
	d) F_r	1.81 (1)
	2.2 Fluid Conditions	
	a) Total Loop flow	$155.8 \text{ Mlbm/hr} \le M \le 190.3 \text{ Mlbm/hr}$
	b) RCS Cold Leg temperature	$548.0°F \le T \le 566.0°F$
	c) Upper head temperature	~RCS Hot Leg Temperature (3)
	d) Pressurizer pressure	$2100 \text{ psia} \le P \le 2325 \text{ psia}$
	e) Pressurizer liquid level	$24 \text{ percent} \le L \le 59 \text{ percent}$
	f) SIT pressure	$602 \text{ psia} \le P \le 675 \text{ psia}$
	g) SIT liquid volume	$1750 \text{ ft}^3 \le V \le 1950 \text{ ft}^3$
	h) SIT temperature	$50.0°F \le T \le 120.0°F$ (4)
	i) SIT resistance fL/D	As-built piping configuration
	j) Minimum SIT boron	2300 ppm

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1 Includes measurement uncertainty.

2 The value used for FQ is derived from the LHGR Technical Specification value.

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

4 Coupled with containment temperature.

Table 4-1
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 (w/ offsite power) 30 s (w/o offsite power)
	f) LPSI pump delay	30 s (w/ offsite power) 30 s (w/o offsite power)
	g) Initial Containment pressure	14.2 psia
	h) Initial Containment temperature	50.0°F ≤ T ≤ 120.0°F
	i) Containment sprays delay	30 s
	j) Containment spray water temperature	50.0°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	1872.0	1872.0	0.0	0.0
25.2	1801.0	1801.0	0.0	0.0
54.2	1603.0	1603.0	0.0	0.0
64.2	1510.0	1510.0	0.0	0.0
77.2	1403.0	1403.0	0.0	0.0
96.2	1215.0	1215.0	0.0	0.0
114.2	1012.5	1012.5	0.0	0.0
117.2	1000.0	1000.0	0.0	0.0
126.2	750.0	750.0	0.0	0.0
138.2	500.0	500.0	0.0	0.0
144.2	333.5	333.5	0.0	0.0
147.2	250.0	250.0	0.0	0.0
156.2	0.0	0.0	0.0	0.0
167.2	0.0	0.0	0.0	0.0

Table 4-1
RLBLOCA Analysis – Plant Parameter Values and Ranges (continued)

I) 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 4-2
Statistical Distribution Used for Process Parameters

Table 4-3
Compliance with 10 CFR 50.46(b)

UTL for 95/95 Simultaneous Coverage/Confidence

Parameter	Value	Case Number
PCT (°F)	1791	[ ]
MLO (%)	1.97	[ ]
CWO (%)	0.014	[ ]

Characteristics of Case Setting the PCT UTL

PCT (°F)	1791
PCT Rod Type	Fresh UO_2 Rod
Time of PCT (s)	7.00
Elevation within Core (ft)	3.25
Local Maximum Oxidation (%)	1.47
Total Core-Wide Oxidation (%)	0.004
PCT Rod Rupture Time (s)	No Rupture for Case
Rod Rupture Elevation within Core (ft)	No Rupture for Case

[ ]

Table 4-4
Summary of Major Parameters for the Demonstration Case

Parameter	Value
Core Power (MWt)	4070
Fresh Fuel Time in Cycle (hrs)	5760
Burned Fuel Time in Cycle (hrs)	18183
Fresh Fuel Assembly Avg. Burnup (GWd/mtU)	10.4
Burned Fuel Assembly Avg. Burnup (GWd/mtU)	30.2
Core Peaking Factor, LHGR (F_Q)	12.69
Radial Peaking Factor, F_r^T ($F_{\Delta H}$)	1.81
Fresh Fuel Axial Shape Index	0.100
Burned Fuel Axial Shape Index	0.112
Break Type	Split
Break Size (ft^2/side)	4.100
[ ]	**[ ]**

Table 4-5
Calculated Event Times for the Demonstration Case

Event	Time (sec)
Begin Analysis	0
Break Opened	0
RCP Tripped	0
SIAS Issued	0.8
PCT Occurred (1791°F)	7.0
Start of Broken Loop SIT Injection	12.4
Start of Intact Loop SIT Injection (Loop 1B, 2A and 2B, respectively)	13.9, 13.9 and 13.9
Beginning of Core Recovery (Beginning of Reflood)	25.2
HPSI Available	30.8
LPSI Available	30.8
Broken Loop LPSI Delivery Began	30.8
Intact Loop LPSI Delivery Began (Loop 1B, 2A and 2B, respectively)	30.8, N/A, and N/A
Broken Loop HPSI Delivery Began	30.8
Intact Loop HPSI Delivery Began (Loop 1B, 2A and 2B, respectively)	30.8, 30.8, and 30.8
Intact Loop SIT Emptied (Loop 1B, 2A and 2B, respectively)	57.8, 57.2, and 57.2
Broken Loop SIT Emptied	58.2
Transient Calculation Terminated	900.0

Table 4-6
Fuel Rod Rupture Ranges of Parameters

Figure 4-1
Scatter Plot Key Parameters

Figure 4-1
Scatter Plot Key Parameters (continued)

Figure 4-2
PCT versus PCT Time Scatter Plot

Figure 4-3
PCT versus Break Size Scatter Plot

Figure 4-4
MLO vs PCT Scatter Plot

Figure 4-5
CWO vs PCT Scatter Plot

Figure 4-6
Peak Cladding Temperature (Independent of Elevation)

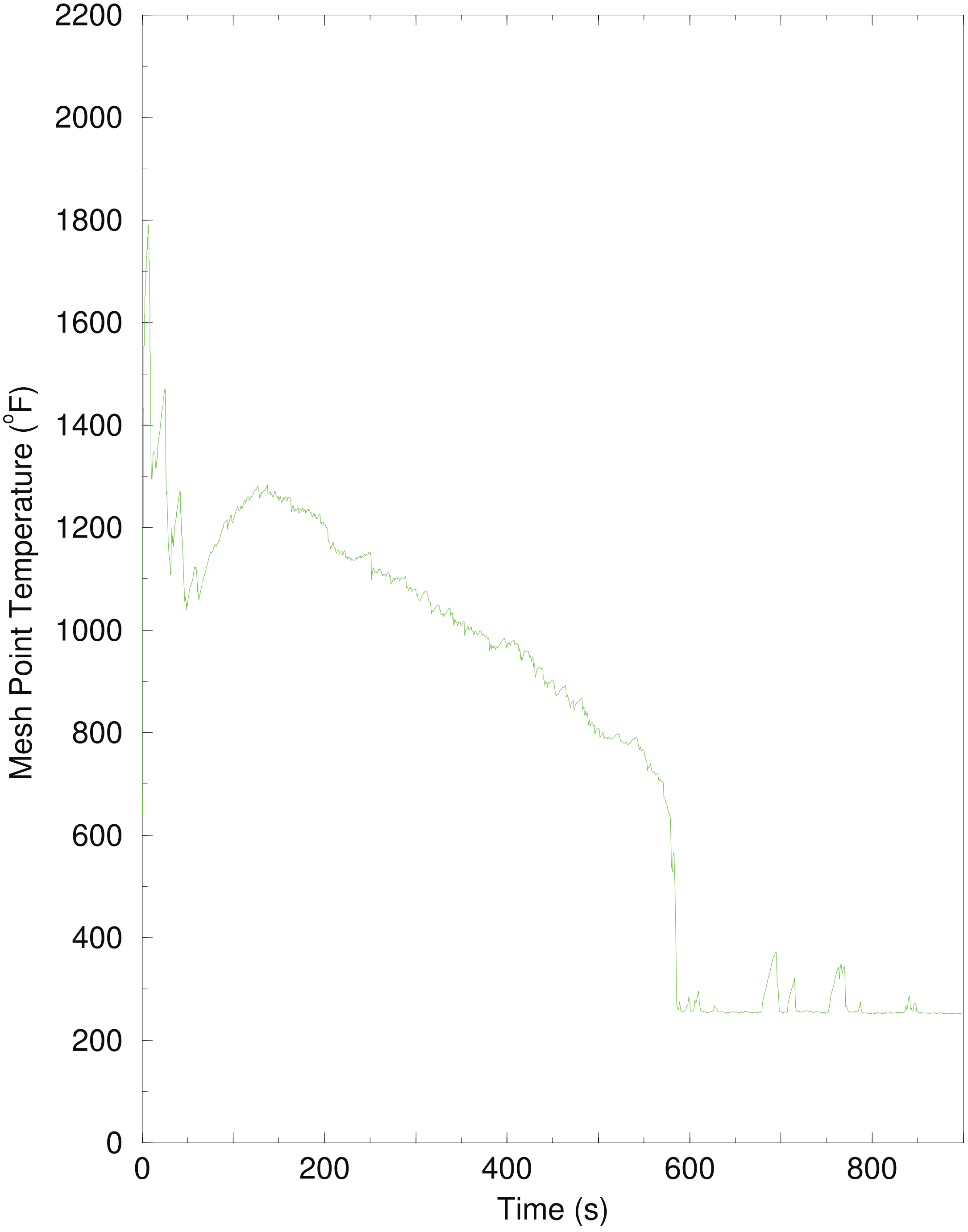

Figure 4-7
Break Flow

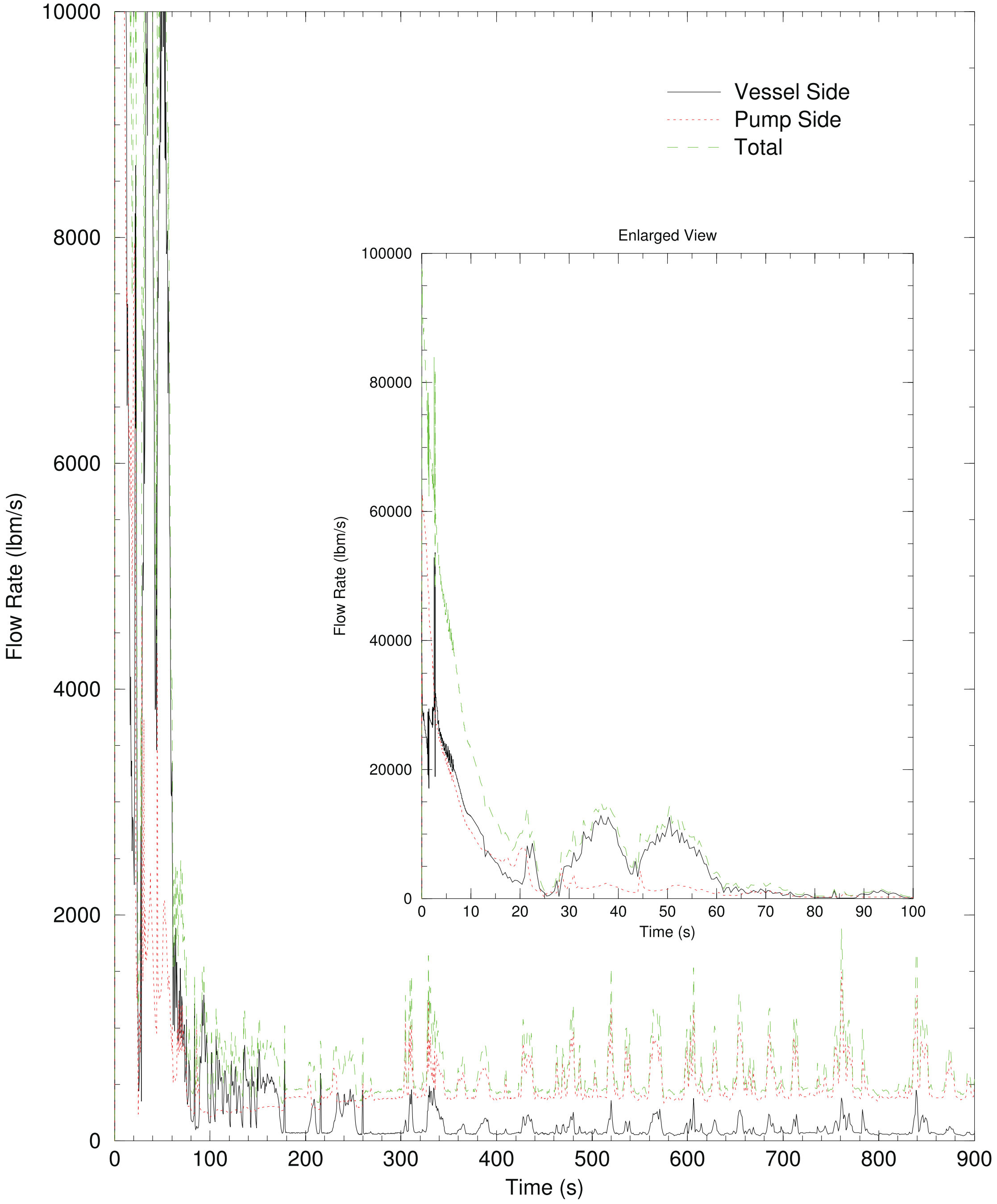

Figure 4-8
Core Inlet Mass Flux

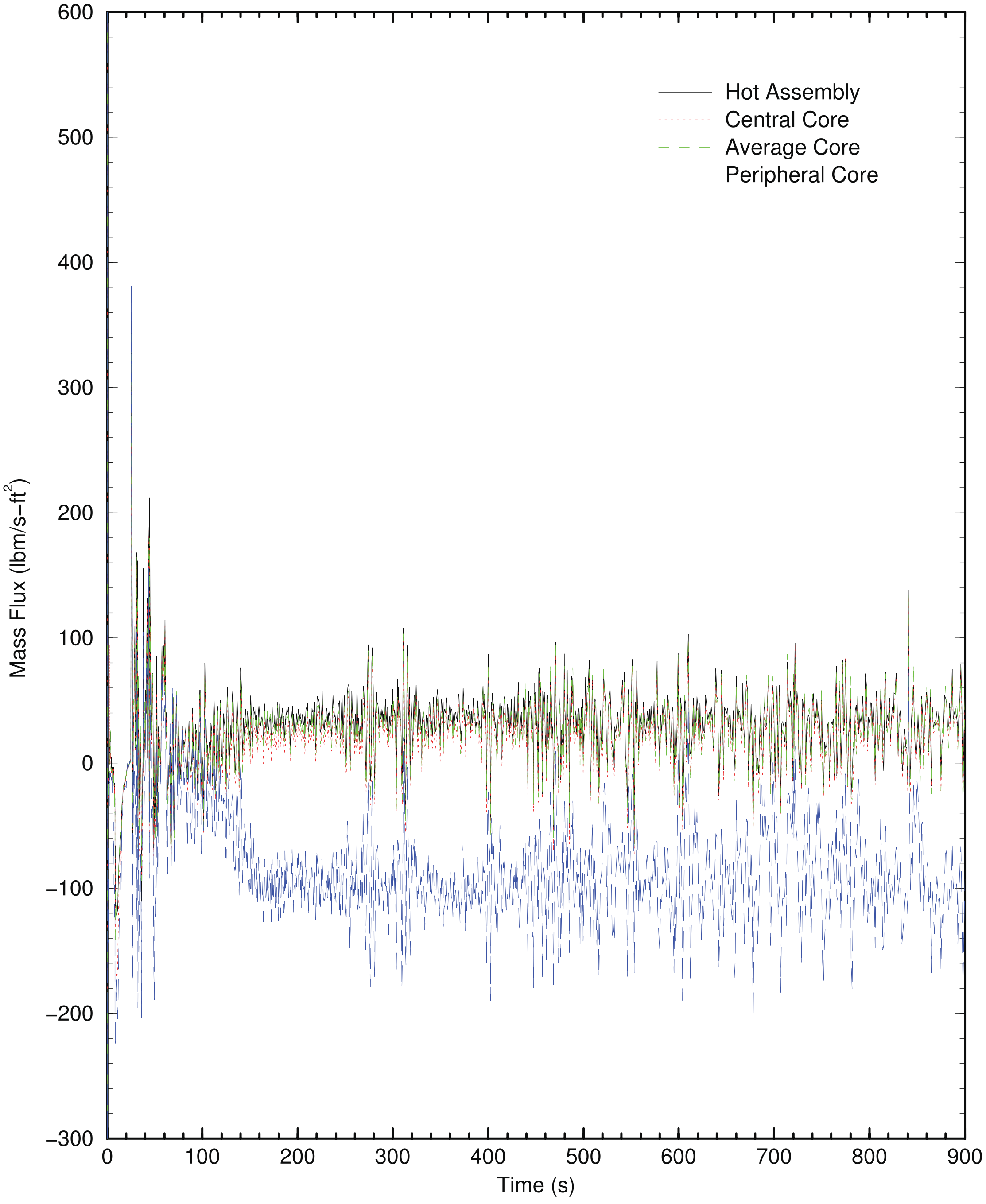

Figure 4-9
Core Outlet Mass Flux

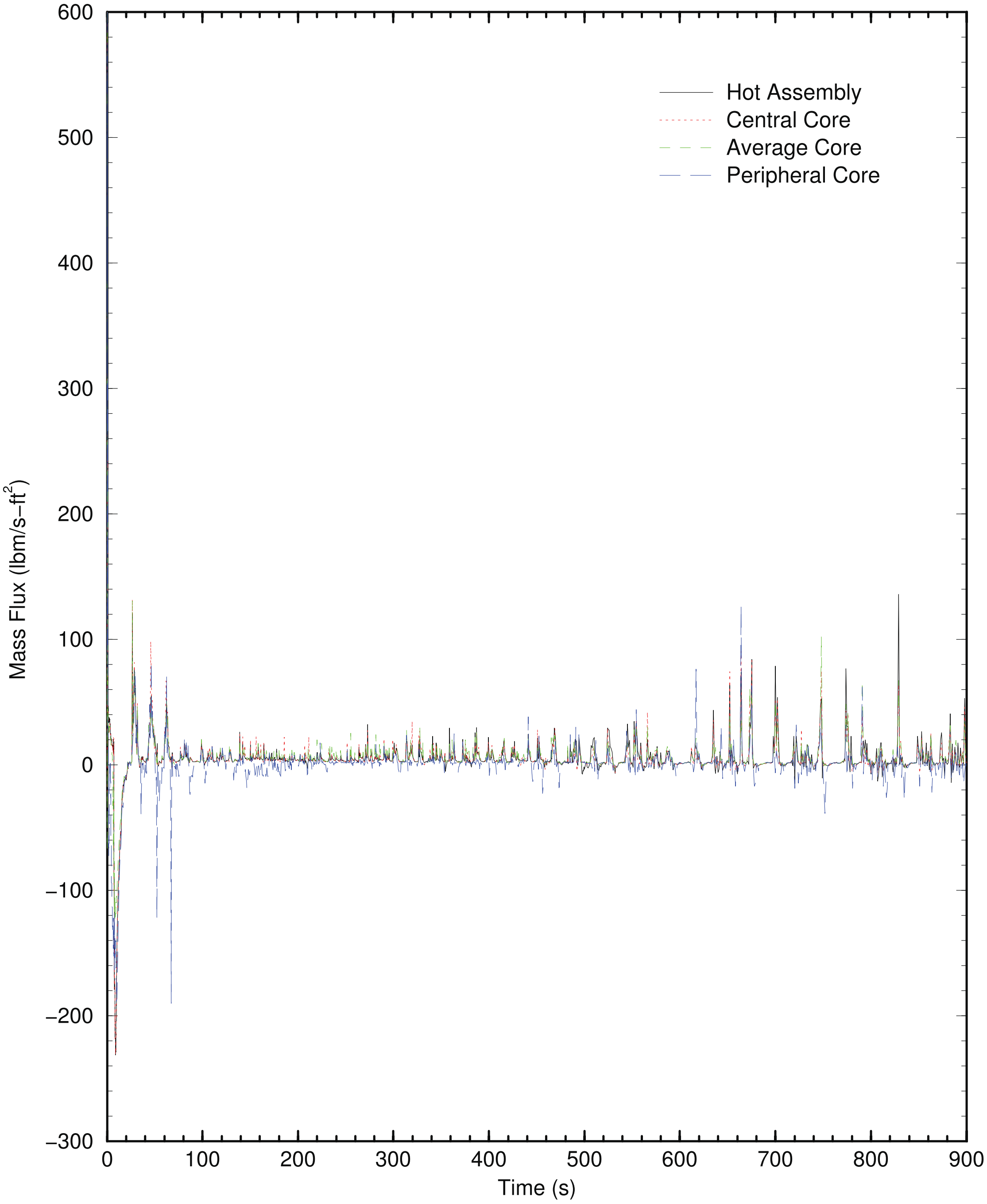

Figure 4-10
Void Fraction at RCS Pumps

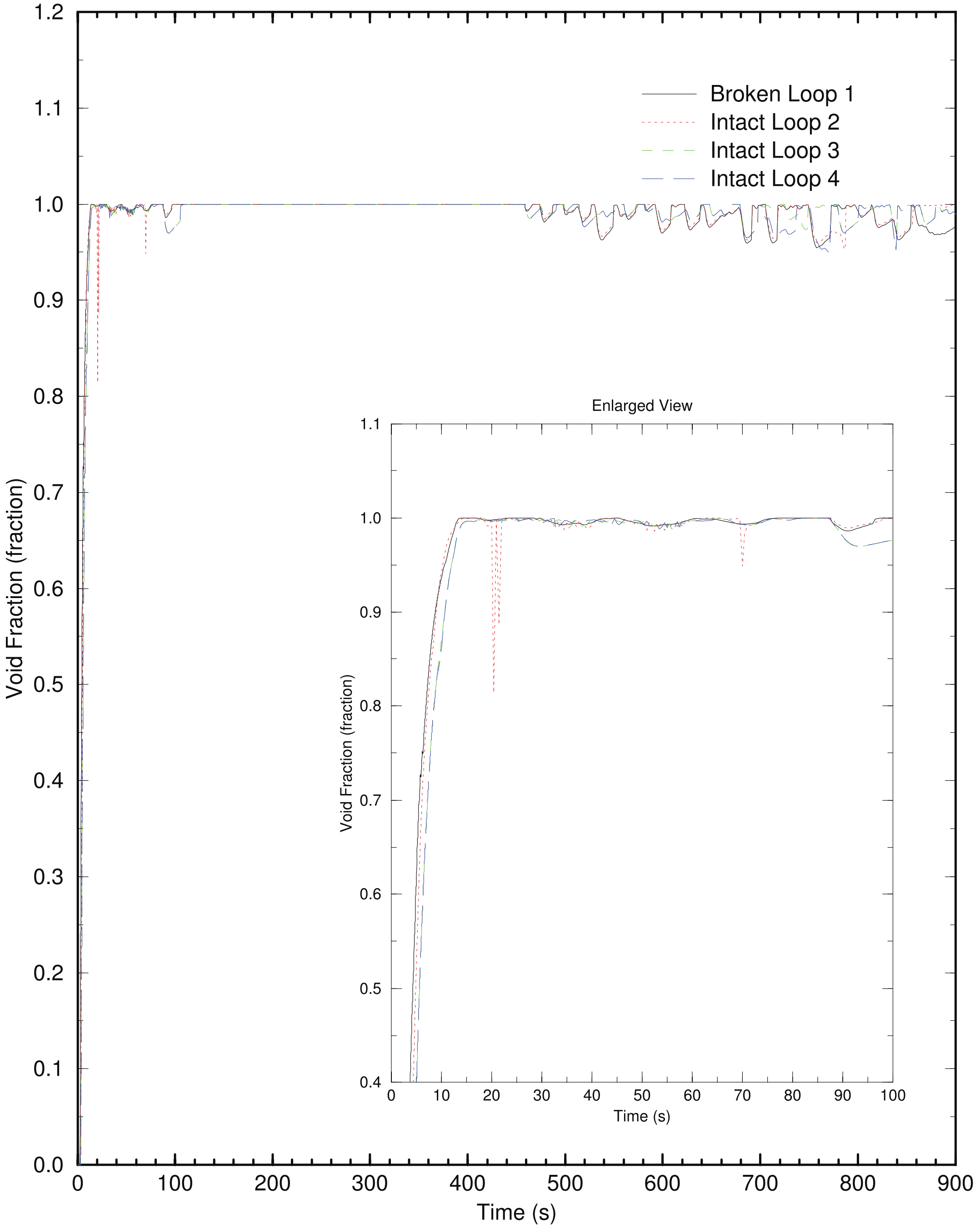

Figure 4-11
ECCS Flows (Includes SIT, HPSI, and LPSI)

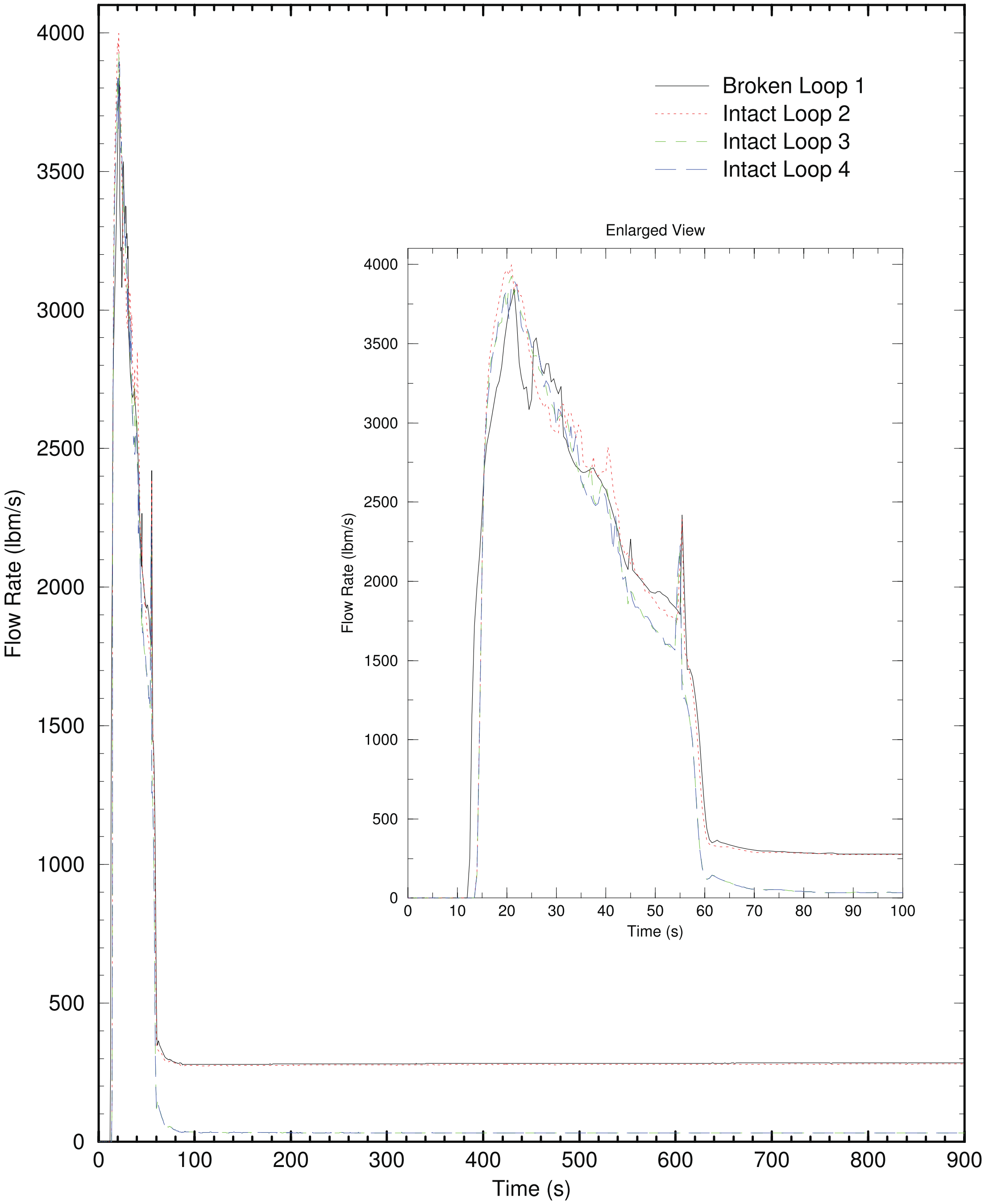

Figure 4-12
Upper Plenum Pressure

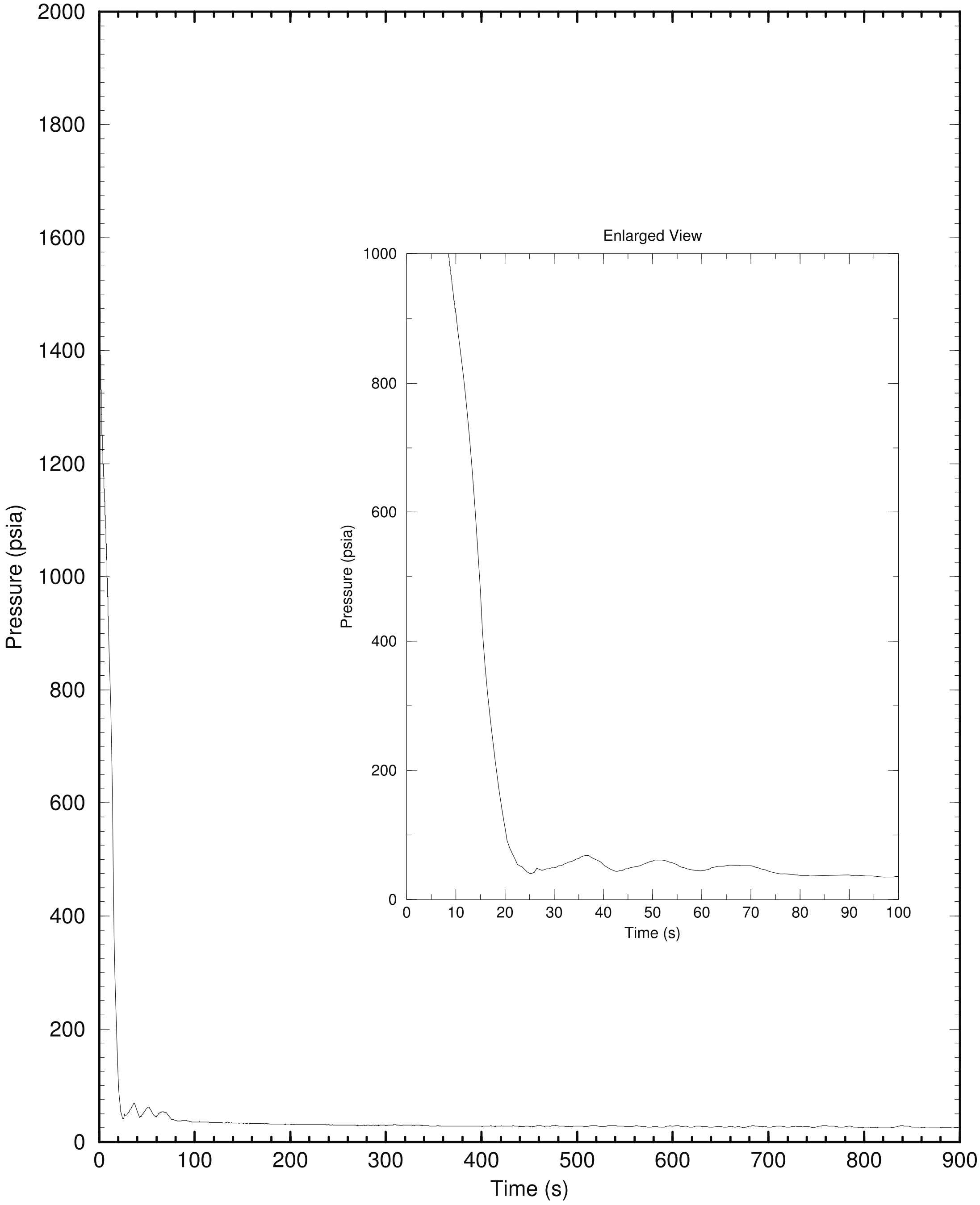

Figure 4-13
Collapsed Liquid Level in the Downcomer

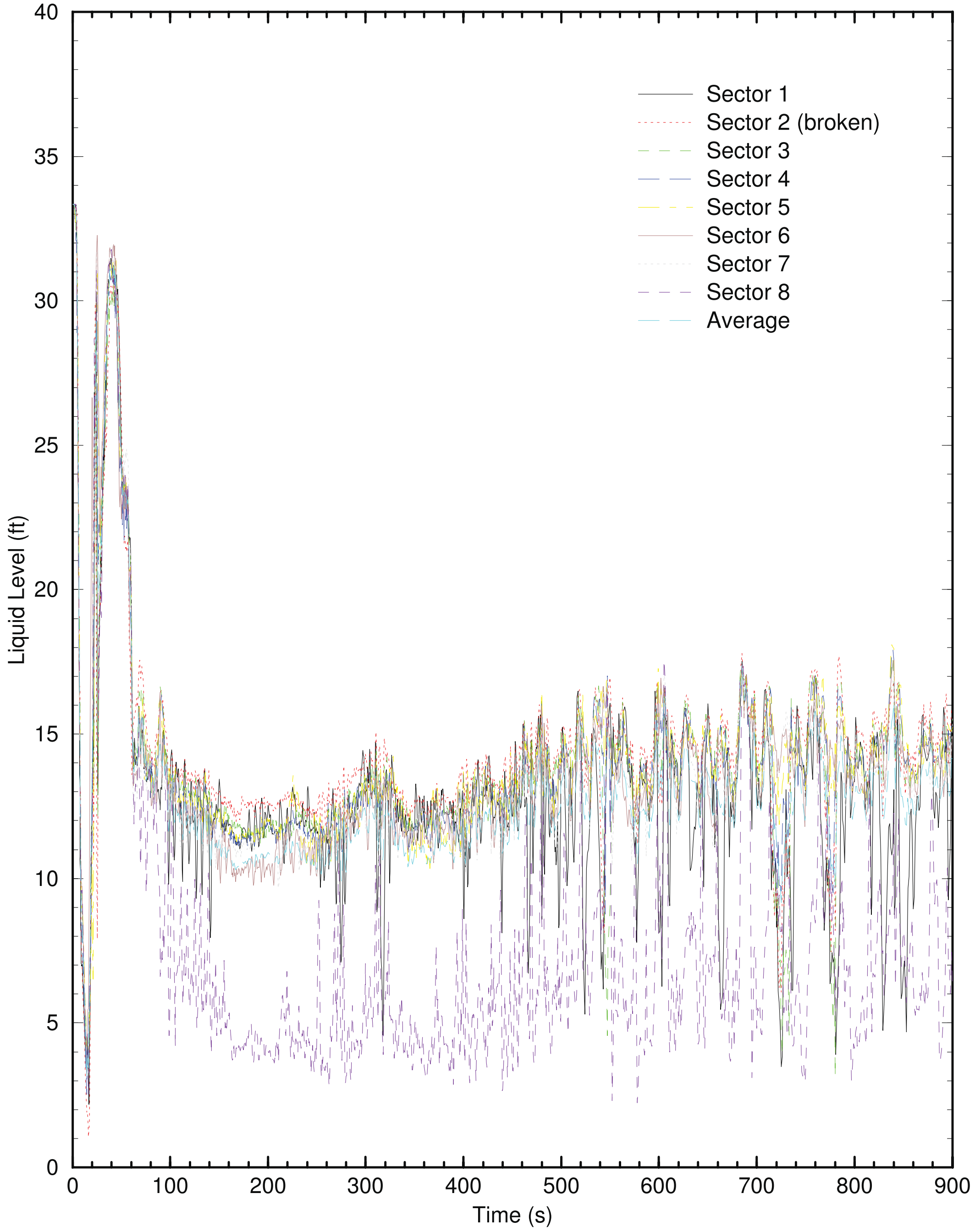

Figure 4-14
Collapsed Liquid Level in the Lower Plenum

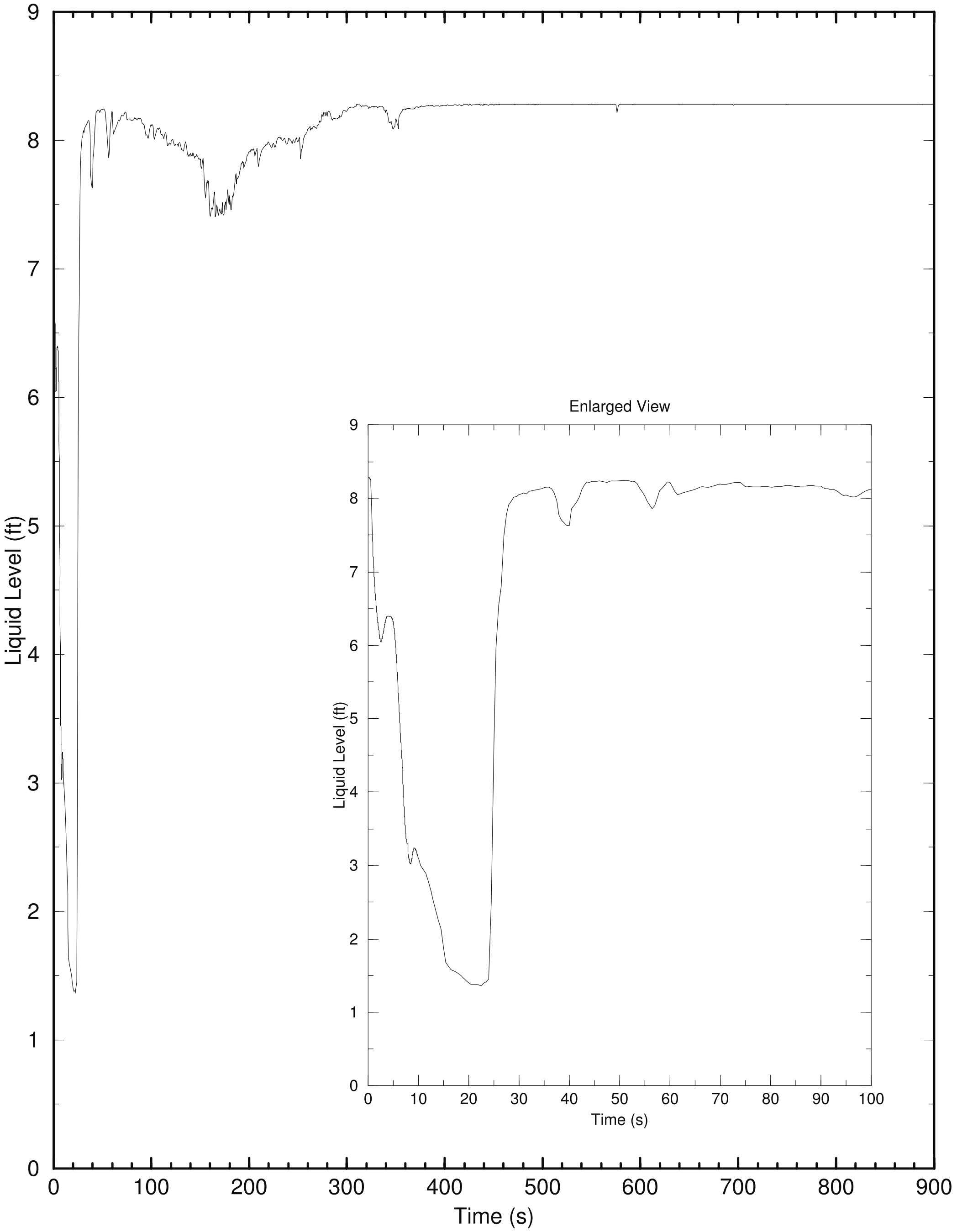

Figure 4-15
Core Collapsed Liquid Level

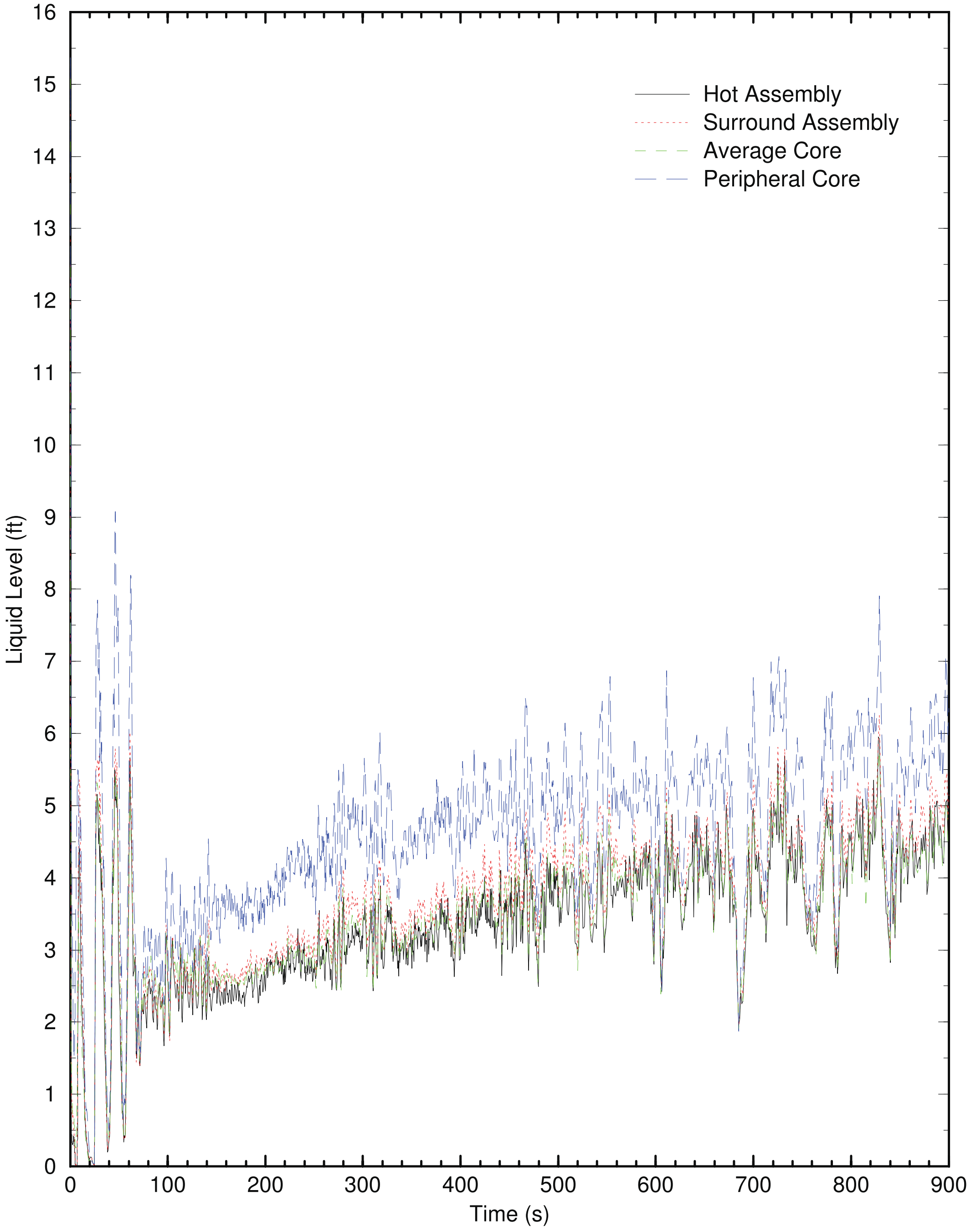

Figure 4-16
Containment and Loop Pressures

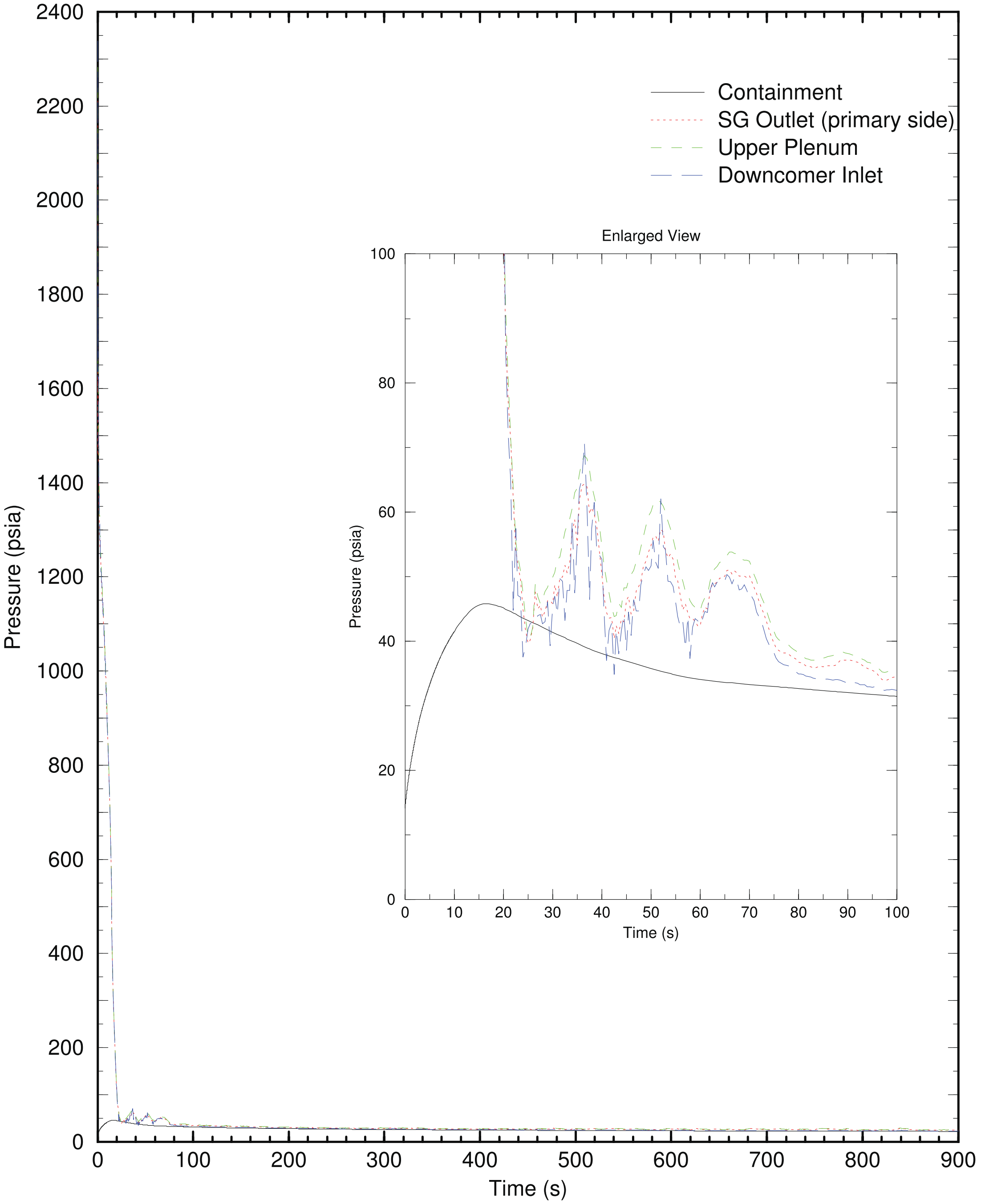

Figure 4-17
Pressure Differences between Upper Plenum and Downcomer

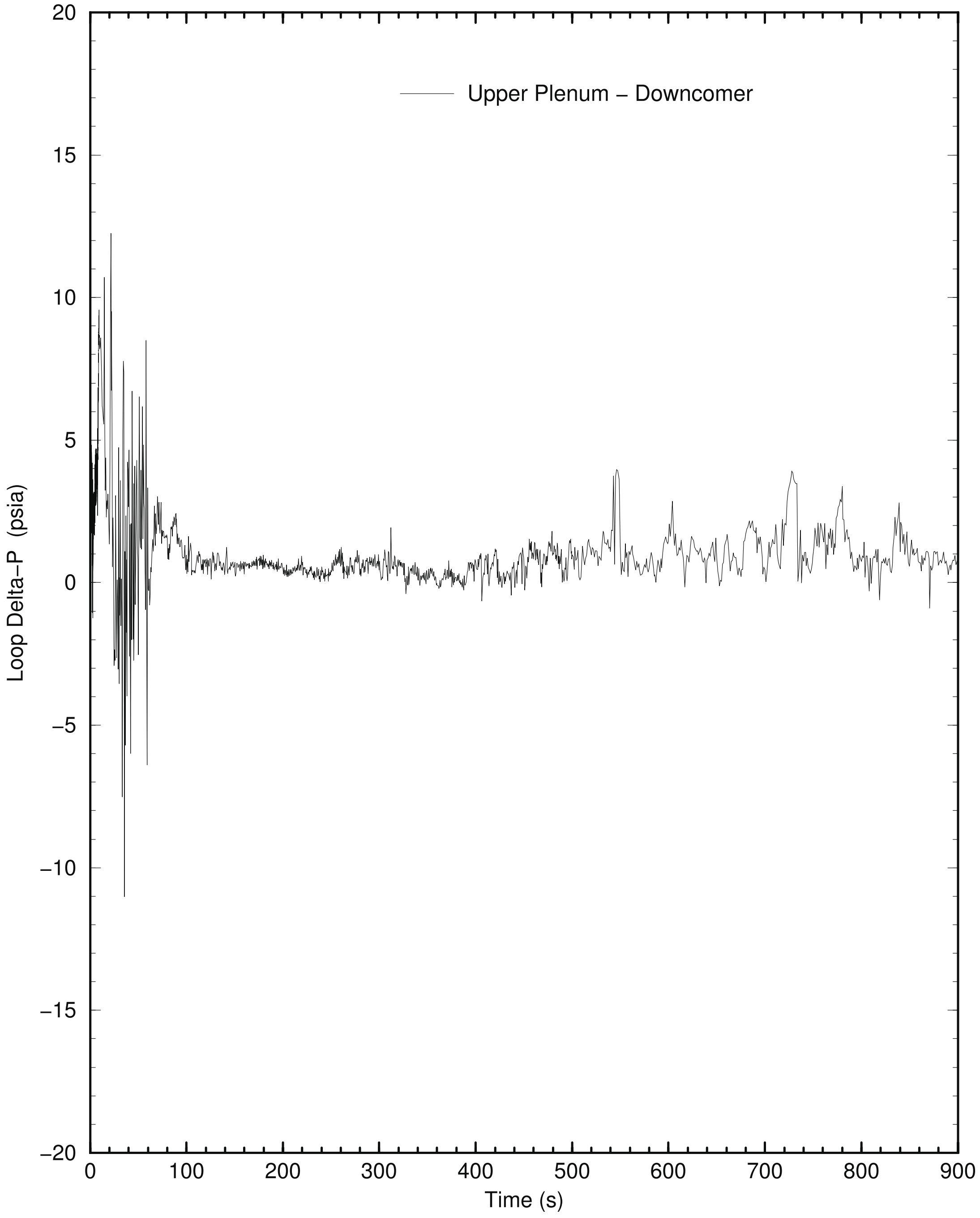

Figure 4-18

[ ]

5.0 REFERENCES

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