energy harbor

Energy Harbor Nuclear Corp.
Beaver Valley Power Station
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John J. Grabnar
Site Vice President, Beaver Valley Nuclear

724-682-5234

August 31, 2022
L-21-238

10 CFR 50.90

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

SUBJECT:
Beaver Valley Power Station, Unit Nos. 1 and 2
Docket No. 50-334, License No. DPR-66
Docket No. 50-412, License No. NPF-73
<u>License Amendment Request for Addition of Analytical Methodology to the Core Operating Limits Report for a Full Spectrum Loss of Coolant Accident</u>

Pursuant to 10 CFR 50.90, Energy Harbor Nuclear Corp. hereby requests an amendment to revise the Technical Specifications (TS) for Beaver Valley Power Station Units 1 and 2. The proposed amendment requests the addition of the Westinghouse Electric Company LLC (Westinghouse) Topical Report WCAP-16996-P-A, Rev.1, *Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)*, to the list of approved analytical methods used to determine the core operating limits as listed in TS 5.6.3, “Core Operating Limits Report (COLR).” To allow for a staggered implementation during refueling outages at each unit, a note would be added to the legacy loss-of-coolant accident methods listed in TS 5.6.3.b to restrict their future use. The proposed amendment also removes Zircalloy from the list of fuel rod cladding in TS 4.2.1, “Fuel Assemblies.”

An evaluation of the proposed amendment is attached. Westinghouse proprietary and non-proprietary versions of a supporting document referenced in the evaluation are provided in enclosures A and C. An affidavit from Westinghouse attesting to the proprietary nature of the information is provided in Enclosure B. Westinghouse proprietary and non-proprietary responses to potential requests for additional information are provided in enclosures D and F respectively, with an affidavit from Westinghouse attesting to the proprietary nature of the information provided in Enclosure E. Therefore, Energy Harbor Nuclear Corp. requests that enclosure A and D be withheld from public disclosure in accordance with 10 CFR 2.390.

Approval of the proposed amendment is requested by August 31, 2023. Once approved, the amendment shall be implemented within 90 days.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Phil H. Lashley, Manager – Fleet Licensing, at (330) 696-7208.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 31, 2022.

Sincerely,

John J Grabnar

Digitally signed by John J Grabnar
Date: 2022.08.31 12:09:24 -04'00'

John J. Grabnar

Attachments:
1. Evaluation of the Proposed Amendment
2. Proposed Technical Specification Page Markups
3. Proposed Technical Specification Bases Page Markups (for information only)

Enclosures:
A. Attachment 1 of LTR-LIS-21-67, Suggested Technical Evaluation Section of the Beaver Valley Power Station Unit 1 and 2 LAR Input (Proprietary)
B. Affidavit for Attachment 1 of LTR-LIS-21-67
C. Attachment 1 of LTR-LIS-21-67, Suggested Technical Evaluation Section of the Beaver Valley Power Station Unit 1 and 2 LAR Input (Non-Proprietary)
D. Attachment 1 to DLWM-LOCA-TM-LR-000001-NP, Revision 0, Responses to Potential RAIs on the Beaver Valley Units 1 and 2 Analysis with the Full Spectrum LOCA (FSLOCA) Methodology (Proprietary)
E. Affidavit for Attachment 1 to DLWM-LOCA-TM-LR-000001-NP, Revision 0
F. Attachment 1 to DLWM-LOCA-TM-LR-000001-NP, Revision 0, Responses to Potential RAIs on the Beaver Valley Units 1 and 2 Analysis with the Full Spectrum LOCA (FSLOCA) Methodology (Non-Proprietary)

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

Attachment 1
L-21-238

Evaluation of the Proposed Amendment
(9 pages follow)

Subject: <u>License Amendment Request for Addition of Analytical Methodology to the Core Operating Limits Report for a Full Spectrum Loss of Coolant Accident</u>

1.0 SUMMARY DESCRIPTION

The proposed amendment requests the addition of the Westinghouse Electric Company LLC (Westinghouse) Topical Report WCAP-16996-P-A, Revision 1, *Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)* [Reference 1], to the list of approved analytical methods used to determine the core operating limits as listed in TS 5.6.3, "Core Operating Limits Report (COLR)." The added methodology describes the FULL SPECTRUM™ loss-of-coolant accident (FSLOCA™) evaluation model (EM), and the FSLOCA EM analyses, which cover both small break (SBLOCA) and large break (LBLOCA) scenarios. To allow for a staggered implementation during refueling outages at each unit, a note would be added to the legacy loss-of-coolant accident methods listed in TS 5.6.3.b to restrict their future use. The proposed change also removes Zircalloy from the list of fuel rod cladding in TS 4.2.1, "Fuel Assemblies."

The FSLOCA EM has been generically approved by the Nuclear Regulatory Commission (NRC) for Westinghouse 3-loop and 4-loop plants with cold leg emergency core cooling system (ECCS) injection. Since BVPS Units 1 and 2 are Westinghouse designed 3-loop plants with cold leg ECCS injection, this approved analytical method is applicable.

The FSLOCA EM analyses for BVPS Units 1 and 2 were performed in compliance with the conditions and limitations discussed in WCAP-16996-P-A, Revision 1 [Reference 2]. Based on the FSLOCA EM analysis results, it is concluded that BVPS Units 1 and 2 continue to comply with the requirements in 10 CFR 50.46, *Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors*, criterion (b)(1) through (b)(4).

2.0 DETAILED DESCRIPTION

2.1. System Design and Operation

The primary function of the emergency core cooling system (ECCS) following a loss-of-coolant accident (LOCA) is to remove the stored and fission product decay heat from the reactor core such that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented.

The ECCS consists of the high head safety injection/charging pumps, the refueling water storage tank, low head safety injection pumps, recirculation spray pumps, and the safety injection accumulators with the associated valves, instrumentation, and piping.

The ECCS is designed to cool the reactor core as well as to provide additional shutdown capability following initiation of the following accident conditions:

1. A LOCA, including a pipe break or a spurious relief or safety valve opening in the reactor coolant system, which would result in a discharge larger than that which could be made up by the normal make-up system.
2. A rupture of a control rod drive mechanism causing a rod cluster control assembly ejection accident.
3. A steam or feedwater system break accident, including a pipe break or a spurious relief or safety valve opening in the secondary steam system which would result in an uncontrolled steam release or a loss of feedwater.
4. A steam generator tube rupture.

2.2. Current Technical Specification Requirements

TS 5.6.3 requires, in part, core operating limits be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and contains references to the approved analytical methods that are used to determine the core operating limits. The current methods listed in TS 5.6.3.b for LOCA analyses are WCAP-12945-P-A, Volumes 1 through 5, *Code Qualification Document for Best Estimate LOCA Analysis*; WCAP-16009-P-A, *Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM).*

2.3. Reason for the Proposed Change

By letter dated September 19, 2018, Energy Harbor Nuclear Corp. committed to submit LBLOCA analyses that apply NRC-approved methods including the effects of fuel pellet thermal conductivity degradation (TCD) to the NRC for review and approval within 36 months of the effective date of the planned 10 CFR 50.46 rulemaking. The Westinghouse FSLOCA EM includes the effects of TCD, and this LAR is being submitted to fulfill the commitment.

2.4. Description of the Proposed Change

Mark-ups of the following changes to TS 4.2.1 and TS 5.6.3.b are included in Attachment 2.

TS 4.2.1 currently states, in part, that each fuel assembly shall consist of a matrix of Zircalloy, ZIRLO®, or Optimized ZIRLO™ clad fuel rods. Zircalloy is being removed because it will no longer be utilized in future core designs.

TS 5.6.3.b currently includes approved analytical methods that would no longer be used to support BVPS reload cores. The proposed revision would add a note restricting the use of the following legacy analytical methods listed in TS 5.6.3.b as follows:

WCAP-12945-P-A, Volumes 1 through 5, “Code Qualification Document for Best Estimate LOCA Analysis,” [Shall not be used to determine core operating limits after December 2024]

(For Unit 1 only) WCAP-16009-P-A, “Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM),” [Shall not be used to determine core operating limits after December 2024]

The proposed revision would add WCAP-16996-P-A to the list of approved methods as follows:

WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016,

3.0 TECHNICAL EVALUATION

Reference 2 provides the technical evaluation for the application of the Westinghouse FSLOCA EM to BVPS Unit 1 and 2. This evaluation was performed in accordance with the NRC-approved FSLOCA EM in Westinghouse Topical Report WCAP-16996-P-A.

The application of this topical report to BVPS involves two separate analyses for Region I (SBLOCA) and Region II (LBLOCA) with the FSLOCA EM for BVPS Unit 1 and 2 due to plant differences.

The removal of Zircalloy from the list of fuel rod cladding in TS 4.2.1, addition of the Westinghouse Topical Report WCAP-16996-P-A and addition of a note to restrict future use of legacy loss-of-coolant accident methods listed in TS 5.6.3.b, would have no technical impact on the ability to meet COLR limits.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The FSLOCA EM in WCAP-16996-P-A satisfies the requirements of 10 CFR 50.46(b) paragraphs (1) through (4). This proposed amendment demonstrates that there is a high level of probability that the following criteria in 10 CFR 50.46 are met as follows:

Criterion (b)(1) requires that the analysis peak cladding temperature (PCT) corresponds to a bounding estimate of the 95th percentile PCT at the 95 percent confidence level. Since the resulting PCT is less than 2,200°F, the analysis with the FSLOCA EM confirms that 10 CFR 50.46 acceptance criterion (b)(1), "Peak Cladding Temperature does not exceed 2,200°F," is demonstrated.

Criterion (b)(2) requires that the analysis Maximum Local Oxidation (MLO) corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Since the resulting MLO is less than 17 percent when converting the time-at-temperature to an equivalent cladding reacted using the Baker-Just correlation and adding the pre-transient corrosion, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), "Maximum Local Oxidation of the cladding does not exceed 17 percent," is demonstrated.

Criterion (b)(3) requires that the analysis Core-Wide Oxidation (CWO) corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. Since the resulting CWO is less than 1 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3),"Core-Wide Oxidation does not exceed 1 percent," is demonstrated.

Criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains in a coolable geometry. This criterion is met by demonstrating compliance with criteria (b)(1) and (b)(2), and by assuring that fuel assembly grid deformation due to combined LOCA and seismic loads is specifically addressed.

The effects of LOCA and seismic loads on the core geometry is discussed in Section 32.1 of the NRC-approved FSLOCA EM (Reference 1) and do not need to be considered unless fuel assembly grid deformation extends beyond the core periphery, for example, deformation in a fuel assembly with no sides adjacent to the core baffle plates. Inboard grid deformation due to combined LOCA and seismic loads is not calculated to occur for Beaver Valley Unit 1 and Unit 2.

Based on the analysis results for Region I, which includes breaks that are typically defined as Small Break LOCAs (SBLOCAs) and Region II, which includes break sizes that are typically defined as Large Break LOCAs (LBLOCAs), it is concluded that Beaver Valley Unit 1 and Unit 2 comply with the criteria in 10 CFR 50.46.

The following regulatory requirements are applicable to ECCS functions:
General Design Criterion (GDC) 35 - Emergency Core Cooling, "A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere

with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Conformance with General Design Criterion 35, "Emergency Core Cooling," is described in more detail in Reference 1. The proposed change does not affect compliance with these regulations or guidance and will ensure that the lowest functional capabilities or performance levels of equipment required for safe operation are met.

4.2 Precedent

The proposed amendment requests the addition of the Westinghouse Electric Company LLC (Westinghouse) Topical Report WCAP-16996-P-A, Revision 1, to the list of approved analytical methods used to determine the core operating limits as listed in TS 5.6.3. Numerous previous requests have been approved for methodology reference changes in plant-specific TS COLR reference lists. Response to potential requests for additional information regarding the FSLOCA methodology have been provided by Westinghouse and are included as Enclosures within this amendment request. This proposed change is consistent with the issuance of a license amendment to Turkey Point Nuclear Plant, Unit 3 and 4, dated April 15, 2021, (ML21105A848).

4.3 No Significant Hazards Consideration Analysis

The proposed change removes Zircalloy from the list of fuel rod cladding in TS 4.2.1, “Fuel Assemblies.” The proposed amendment also requests the addition of the Westinghouse Electric Company LLC (Westinghouse) Topical Report WCAP-16996-P-A, Revision 1, *Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)* [Reference 1], to the list of approved analytical methods used to determine the core operating limits as listed in TS 5.6.3, “Core Operating Limits Report (COLR).” To allow for a staggered implementation during refueling outages at each unit, a note would be added to the legacy loss-of-coolant accident methods listed in TS 5.6.3.b to restrict their future use.

Energy Harbor Nuclear Corp. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   The FULL SPECTRUM™ loss-of-coolant accident (FSLOCA™) evaluation model (EM), and the FSLOCA EM analyses supports ZIRLO and Optimized ZIRLO cladding. Throughout the industry, ZIRLO and Optimized ZIRLO has replaced Zircalloy as the alloy of choice in PWR's and as a result, Zircalloy is not supported by the FSLOCA EM analysis, nor will it be utilized in future core designs. The cores of subsequent cycles will consist of ZIRLO and Optimized ZIRLO and will continue to meet the applicable design criteria and ensure that the pertinent licensing basis acceptance criteria are met.

   The proposed change to TS 5.6.3 permits the use of an NRC-approved methodology for analysis of the loss of coolant accidents to determine if Beaver Valley Power Station (BVPS) Unit 1 and 2, continue to meet the applicable design and safety analysis acceptance criteria. Restricting the future use of legacy loss-of-coolant accident methods listed in TS 5.6.3.b is required to allow for a staggered implementation during refueling outages at each unit and has no direct impact upon plant operation or configuration.

   The results of the BVPS loss of coolant accident analyses demonstrate Energy Harbor Nuclear Corp. continues to satisfy the 10 CFR 50.46(b)(1-4) emergency core cooling system (ECCS) performance acceptance criteria using an NRC-approved evaluation model.

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

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

   Response: No.

   The proposed change will not create the possibility of a new or different accident due to credible new failure mechanisms, malfunctions, or accident initiators not previously considered. There are no physical plant modifications being made; thus, the possibility of a new or different type of accident is not created.

Therefore, the proposed change does not create the possibility of a new or different kind of accident or malfunction from any previously evaluated.

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

   Response: No.

   The proposed FSLOCA EM used in the analysis would more realistically describe the expected behavior of plant systems during a postulated loss of coolant accident. Uncertainties have been accounted for as required by 10 CFR 50.46. It has been shown by analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46, Paragraph b are met. No design basis safety limits are exceeded or altered by this change. Approved methodologies would continue to be used to ensure that the plants continue to meet applicable design criteria and safety analysis acceptance criteria.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above information, Energy Harbor Nuclear Corp. concludes that 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.

4.4 Conclusions

In conclusion, 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 to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, 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.

6.0 REFERENCES

1. Submittal of WCAP-16996-P-A/WCAP-16996-NP-A, Volumes I, II, III and Appendices, Revision 1, “Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)" (TAC No. ME5244)(Proprietary/Non-Proprietary) ADAMS Accession No. ML17277A130

2. Attachment 1 of LTR-LIS-21-67, Suggested Technical Evaluation Section of the Beaver Valley Power Station Unit 1 and 2 LAR Input (Proprietary)

Attachment 2
L-21-238

Technical Specification Page Markups
(2 pages follow)

4.0 DESIGN FEATURES

4.1 Site Location

The Beaver Valley Power Station is located in Shippingport Borough, Beaver County, Pennsylvania, on the south bank of the Ohio River. The site is approximately 1 mile southeast of Midland, Pennsylvania, 5 miles east of East Liverpool, Ohio, and approximately 25 miles northwest of Pittsburgh, Pennsylvania. The Unit 1 exclusion area boundary has a minimum radius of 2000 feet from the center of containment. The Unit 2 exclusion area boundary has a minimum radius of 2000 feet around the Unit No. 1 containment building.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 157 fuel assemblies. Each assembly shall consist of a matrix of ~~Zircalloy,~~ ZIRLO® or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 48 control rod assemblies. The control material shall be silver indium cadmium as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment as specified in LCO 3.7.14, "Spent Fuel Pool Storage,"

b. Unit 1
$K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.12 of the UFSAR,

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

LCO 3.1.5.1, "Unit 1 Shutdown Bank Insertion Limits"

LCO 3.1.5.2, “Unit 2 Shutdown Bank Insertion Limits”

LCO 3.1.6.1, "Unit 1 Control Bank Insertion Limits"

LCO 3.1.6.2, “Unit 2 Control Bank Insertion Limits”

LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)"

LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)"

LCO 3.2.3, "Axial Flux Difference (AFD)"

LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation" - Overtemperature and Overpower ΔT Allowable Value parameter values

LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"

LCO 3.9.1, "Boron Concentration"

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

WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology,"

WCAP-8745-P-A, "Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT Trip Functions,"

WCAP-16996-P-A, Revision 1, “Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology),” November 2016,

WCAP-12945-P-A, Volumes 1 through 5, "Code Qualification Document for Best Estimate LOCA Analysis," [Shall not be used to determine core operating limits after December 2024]

(For Unit 1 only) WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," [Shall not be used to determine core operating limits after December 2024]

WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification,"

WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,"

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report,"

WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicating Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids,"

Attachment 3
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TS Bases Page Markups (for information only)
(15 pages follow)

BASES

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large or small break loss of coolant accident (LOCA), the ~~peak cladding temperature must not exceed 2200°F~~ 10 CFR 50.46 acceptance criteria must be met (Ref. 1),

b. During ~~a~~ there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition,

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2), and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. ~~Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.~~

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \le [CFQ / P]\ \ K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \le [CFQ / 0.5]\ \ K(Z) \qquad \text{for } P \le 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

P = THERMAL POWER / RTP

The actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of 2.40, and K(Z) is a function that looks like the one provided in Figure B 3.2.1-1. Figure B 3.2.1-1 is for illustration purposes only. The actual unit specific K(Z) as a function of core height figures are contained in the COLR.

FOR INFORMATION ONLY

BACKGROUND (continued)

cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^{N}_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition,

b. During a large or small break loss of coolant accident (LOCA), ~~peak cladding temperature (PCT) must not exceed 2200°F~~ (Ref. 3),

10 CFR 50.46 acceptance criteria must be met

c. During a [illegible] to the fuel must not exceed 280 cal/gm (Ref. 1), and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F^{N}_{\Delta H}$ are the core parameters of most importance. The limits on $F^{N}_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis ensures the probability that DNB will not occur on the most limiting fuel rod is at least 95% at a 95% confidence level. This is met by limiting the minimum DNBR to the 95/95 DNB criterion of 1.22 for typical and thimble cells using the WRB-2M CHF correlation, and 1.23 for the typical cell and 1.22 for the thimble cell using the WRB-1 CHF correlation. These values provide a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^{N}_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^{N}_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, DNB events in which the core limits are modeled implicitly use this variable value of $F^{N}_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^{N}_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^{N}_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are also indirectly modeled in the LOCA safety analyses that verify ~~the acceptability of the resulting peak cladding temperature~~ (Ref. 3).

the 10 CFR 50.46 acceptance criteria are met

FOR INFORMATION ONLY

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and ~~for limiting peak clad temperature to ≤ 2200°F~~ compliance with the 10 CFR 50.46 acceptance criteria), and
2. Boration to ensure recovery and maintenance of SDM (k_{eff} < 1.0).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Phase A Isolation,
- Reactor Trip,
- Turbine Trip,
- Feedwater Isolation,
- Start of auxiliary feedwater (AFW) pumps, and
- Enabling automatic switchover of Emergency Core Cooling Systems (ECCS) suction to containment sump.

These other functions ensure:

- Isolation of nonessential systems through containment penetrations,
- Trip of the turbine and reactor to limit power generation,
- Isolation of main feedwater (MFW) to limit secondary side mass losses,
- Start of AFW to ensure secondary side cooling capability, and
- Enabling ECCS suction switchover from the refueling water storage tank (RWST) to the containment sump on RWST Level Extreme Low to ensure continued cooling via use of the containment sump.

FOR INFORMATION ONLY

BASES

APPLICABLE SAFETY ANALYSES

The accumulators are assumed to be OPERABLE in both the large and small break LOCA analyses at full power and hot zero power (HZP) steam line break (SLB) analysis (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a large break LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg large break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The largest break area considered for a large break LOCA is a double ended guillotine break in the RCS cold leg.

~~The limiting large break LOCA is a double ended guillotine break in the cold leg for both Units 1 and 2.~~ During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

No credit is taken for ECCS pump flow in the analysis until full flow is available. If offsite power is not available, the analysis accounts for the diesels starting and the pumps being loaded and delivering full flow. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

intermediate

is assumed to inject into the reactor coolant system

into a small break LOCA

, charging pumps, and low head safety injection pumps

charging and low head safety injection pumps

The ~~worst case~~ small break LOCA analyses also assume a time delay before pumped flow ~~reaches the core~~. For ~~the larger range of small~~ breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators ~~and charging pumps both~~ play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the ~~charging pumps~~ become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F,

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BASES

APPLICABLE SAFETY ANALYSES (continued)

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation,

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 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, and

d. Core is maintained in a coolable geometry.

and the recovery phase of a small break LOCA

Since the accumulators discharge during the blowdown phase of a large break LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

The large and small break LOCA analyses use a range of accumulator water volumes per approved methods. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. Both large and small break analyses use a nominal accumulator line volume from the accumulator to the check valve.

~~For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The nominal water volume assumed in the analyses is within the range of accumulator volumes specified in Surveillance Requirement 3.5.1.2. The contained water volume is not the same as the usable volume of the accumulators, since the accumulators are not completely emptied after discharge. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. Therefore, the large break LOCA analyses use a range of accumulator volumes. The Unit 1 ASTRUM large break LOCA analysis statistically calculates the accumulator water volume over the range of accumulator volumes specified in Surveillance Requirement 3.5.1.2. For Unit 2, the large break LOCA analysis assumes values of 6898 gallons and 8019 gallons for accumulator volume. The large break LOCA analyses also credit the line water volume from the accumulator to the check valve~~.

The minimum boron concentration is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

and large break LOCA analyses

The small break ~~LOCA analysis is performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that a higher nitrogen cover pressure results in a computed peak clad~~

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APPLICABLE SAFETY ANALYSES (continued)

use a range of accumulator nitrogen cover pressures per approved methods.

~~temperature benefit~~. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity. The accumulators also discharge following a SLB; however, their impact is minor with respect to meeting the design basis DNB limit.

The specified Technical Specification values for the usable accumulator volume, boron concentration, and minimum nitrogen pressure are analysis values. Also, the values specified for nitrogen pressure and volume do not account for instrument uncertainty.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Ref 3).

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

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Three accumulators are required to ensure that 100% of the contents of two of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than two accumulators are injected during the blowdown phase of ~~a LOCA~~, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

large break LOCA and the recovery phase of a small break LOCA

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 2000 psig, and the limits established in the SRs for usable volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that ~~peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.~~

10 CFR 50.46 acceptance criteria are met.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The value specified for boron concentration is an analysis value. The boron concentration should be verified to be within required limits for each accumulator since the static design of the accumulators limits the ways in which the concentration can be changed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Sampling the affected accumulator within 6 hours after a ≥ 1% accumulator volume increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 5).

SR 3.5.1.5

Verification that power is removed from each accumulator isolation valve operator control circuit when the RCS pressure is > 2000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only one accumulator would be available for injection given a single failure coincident with a LOCA. Power is removed from the accumulator motor operated isolation valves control circuits by removing the plug in the lock out jack from the associated control circuits. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR allows power to be supplied to the motor operated isolation valves control circuits when RCS pressure is ≤ 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to remove control power during plant startups or shutdowns.

REFERENCES

1. UFSAR, Chapter 14 (Unit 1) and UFSAR, Chapter 15 (Unit 2).

2. 10 CFR 50.46.

3. UFSAR, Chapter 14 (Unit 1) and UFSAR, Chapter 6 (Unit 2).

4. WCAP-15049-A, Risk-Informed Evaluation of an Extension to Accumulator Completion Times, Rev. 1, April 1999.

5. NUREG-1366, February 1990.

6. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

FOR INFORMATION ONLY

APPLICABLE SAFETY ANALYSES (continued)

The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The minimum boron concentration limit is an important assumption in ensuring the required shutdown capability. The maximum boron concentration is an explicit assumption in "Spurious Operation of the Safety Injection System at Power" (Unit 1) and "Inadvertent Operation of the ECCS During Power Operation" (Unit 2), however, the results are very insensitive to boron concentration. The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum temperature is an assumption in both the MSLB analysis and the "Spurious Operation of the Safety Injection System at Power" (Unit 1) and "Inadvertent Operation of the ECCS During Power Operation" (Unit 2).

The RWST temperature impacts the large and small break LOCA peak cladding temperature (PCT) calculations, and the LOCA and MSLB containment peak pressure calculations.

ECCS

LOCA ~~PCT~~ Calculations:

In the large break and small break LOCA analyses, a range of RWST temperatures are used for the containment spray temperature and safety injection temperature per approved methods. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. If the higher temperature limit is violated there is less heat transfer from the core to the injected water and increases peak clad temperature.

~~The large break LOCA analysis assumes that the quench spray temperature is equal to the RWST lower limit of 45°F. The lower RWST temperature results in a reduced containment backpressure, which increases steam binding, reducing the flooding rate and results in an increased PCT. The small break LOCA analysis assumes an RWST temperature of 65°F.~~

Containment Integrity Calculations:

Both the LOCA and MSLB containment integrity analyses credit the quench spray to reduce the containment pressure following the accident. The LOCA and MSLB containment analyses assume that the quench spray temperature is greater than or equal to the upper RWST temperature limit of 65°F. A higher RWST temperature results in a reduced cooling and condensation spray capability, and therefore higher calculated containment pressures.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The assumed response times are provided in the Licensing Requirements Manual.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified to be within the limits assumed in the accident analyses band. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.

SR 3.5.4.2

The RWST water volume should be verified to be above the required usable level in order to ensure that a sufficient initial supply is available for injection and the Quench Spray System and to support continued ECCS and Recirculation Spray System pump operation on recirculation. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.4.3

The boron concentration of the RWST should be verified to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that boron precipitation in the core will not occur and that the resulting sump pH will be maintained in an acceptable range so the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Chapter 14 (Unit 1) and UFSAR, Chapter 6 and Chapter 15 (Unit 2).

2. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

FOR INFORMATION ONLY

BASES

APPLICABLE SAFETY ANALYSES (continued)

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with ~~10 CFR 50, Appendix K (Ref. 2).~~ approved methodologies.

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Quench Spray System. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit also ensures that sufficient net positive suction head will be available for the Unit 1 recirculation spray and low head safety injection pumps and the Unit 2 recirculation spray pumps.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

FOR INFORMATION ONLY

BASES

ACTIONS (continued)

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Chapter 14 (Unit 1), and UFSAR, Section 6.2 (Unit 2).

2. 10 CFR 50, Appendix K.

3. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

FOR INFORMATION ONLY

APPLICABLE SAFETY ANALYSES (continued)

EQ requirements (Ref. 2) for equipment inside containment. The EQ requirements provide assurance the equipment inside containment required to function during and after a DBA performs as designed during the adverse environmental conditions resulting from a DBA. Air temperature profiles (containment air temperature vs time) are calculated for each DBA to establish EQ design requirements for the equipment inside containment. The equipment inside containment required to function during and after a DBA is confirmed to be capable of performing its design function under the applicable EQ requirement (i.e., air temperature profile). Therefore, it is concluded that the calculated transient containment atmosphere temperatures resulting from various DBAs, including the most limiting temperature from a SLB, are acceptable.

The modeled QS System actuation from the containment analysis is based upon a response time associated with exceeding the Containment High-High pressure signal setpoint to achieving full flow through the quench spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The QS System total response time is specified in the Licensing Requirements Manual (LRM) and includes the signal delay, diesel generator startup time, and system startup time.

For certain aspects of accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with ~~10 CFR 50, Appendix K (Ref. 3).~~
approved methodologies.

Inadvertent actuation of the QS System is also evaluated, and the resultant reduction in containment pressure is calculated. The maximum calculated reduction in containment pressure does not reduce containment pressure below the minimum containment design pressure of 8.0 psia.

The QS System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.3 and SR 3.6.6.4

These SRs ensure that each QS automatic valve actuates to its correct position and each QS pump starts upon receipt of an actual or simulated containment spray actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.6.5

This SR is performed following maintenance when the potential for nozzle blockage has been determined to exist by an engineering evaluation. The required evaluation will also specify an appropriate test method for determining the spray header OPERABILITY. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Due to the passive nature of the design of the nozzle, a test following maintenance that results in the potential for nozzle blockage is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. UFSAR, Chapter 14 (Unit 1), and UFSAR, Section 6.2 (Unit 2).
2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.
4. ASME code for Operation and Maintenance of Nuclear Power Plants.
5. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

FOR INFORMATION ONLY

BASES

APPLICABLE SAFETY ANALYSES (continued)

The EQ requirements provide assurance the equipment inside containment required to function during and after a DBA performs as designed during the adverse environmental conditions resulting from a DBA. Air temperature profiles (containment air temperature vs time) are calculated for each DBA to establish EQ design requirements for the equipment inside containment. The equipment inside containment required to function during and after a DBA is confirmed to be capable of performing its design function under the applicable EQ requirement (i.e., air temperature profile). Therefore, it is concluded that the calculated transient containment atmosphere temperatures resulting from various DBAs, including the most limiting temperature from a SLB, are acceptable. The RS System is not credited in the SLB containment analysis.

The RS System actuation model from the containment analysis is based upon a response time between receipt of the RWST Level Low signal in coincidence with the Containment Pressure High High to achieving full flow through the RS System spray nozzles. A delay in response time initiation provides conservative analyses of peak calculated containment temperature and pressure. The RS System maximum time from coincidence of Containment Pressure High High and RWST Level Low to the start of effective RS spray is 65 seconds for Unit 1 and 77 seconds for Unit 2.

In the case of the Unit 2 RS System, the containment safety analysis models the operation of the system consistent with the system design. The Unit 2 analysis models the RS subsystems starting in the spray mode of operation. When the unit is shifted to the ECCS recirculation mode of operation the containment analysis models a reduction in recirculation spray flow to account for the Unit 2 RS subsystems used for the ECCS low head recirculation function.

For certain aspects of accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with ~~10 CFR 50, Appendix K (Ref. 3).~~ approved methodologies.

The RS System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

FOR INFORMATION ONLY

BASES

REFERENCES

1. UFSAR, Chapter 14 (Unit 1), and UFSAR, Section 6.2 (Unit 2).
2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.
4. ASME code for Operation and Maintenance of Nuclear Power Plants.

5. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

Enclosure B
L-21-238

Affidavit
(3 pages follow)

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

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

(2) I am requesting the proprietary portions of Attachment 1 of LTR-LIS-21-67, Revision 0 be withheld from public disclosure under 10 CFR 2.390.

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

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

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

(ii) The information sought to be withheld is being transmitted to the Commission in confidence and, to Westinghouse's knowledge, is not available in public sources.

(iii) Westinghouse notes that a showing of substantial harm is no longer an applicable criterion for analyzing whether a document should be withheld from public disclosure. Nevertheless, public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

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

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

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

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

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

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

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

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

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

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

Executed on: 16 Dec 2021

Camille Zozula

Camille T. Zozula, Manager
Regulatory Compliance & Corporate Licensing

Enclosure C
L-21-238

Attachment 1 of LTR-LIS-21-67, Suggested Technical Evaluation Section of the Beaver Valley Power Station Unit 1 and 2 LAR Input
(Non-Proprietary)
(88 pages follow)

Attachment 1

Suggested Technical Evaluation Section of the Beaver Valley Power Station Unit 1 and Unit 2 LAR Input

(Non-Proprietary)

(88 pages, including cover page)

APPLICATION OF WESTINGHOUSE FULL SPECTRUM LOCA EVALUATION MODEL TO THE BEAVER VALLEY POWER STATION UNIT 1 AND UNIT 2

1.0 INTRODUCTION

Analyses with the **FULL SPECTRUM™** loss-of-coolant accident (**FSLOCA™**) evaluation model (EM) have been completed for the Beaver Valley Power Station Unit 1 and Unit 2. This license amendment request (LAR) for Beaver Valley Unit 1 and Unit 2 requests approval to apply the Westinghouse FSLOCA EM.

The FSLOCA EM (Reference 1) was developed to address the full spectrum of loss-of-coolant accidents (LOCAs) which result from a postulated break in the reactor coolant system (RCS) of a pressurized water reactor (PWR). The break sizes considered in the Westinghouse FSLOCA EM include any break size in which break flow is beyond the capacity of the normal charging pumps, up to and including a double-ended guillotine (DEG) rupture of an RCS cold leg with a break flow area equal to two times the pipe area, including what traditionally are defined as Small and Large Break LOCAs.

The break size spectrum is divided into two regions. Region I includes breaks that are typically defined as Small Break LOCAs (SBLOCAs). Region II includes break sizes that are typically defined as Large Break LOCAs (LBLOCAs).

The FSLOCA EM explicitly considers the effects of fuel pellet thermal conductivity degradation (TCD) and other burnup-related effects by calibrating to fuel rod performance data input generated by the PAD5 code (Reference 2), which explicitly models TCD and is benchmarked to high burnup data in Reference 2. The fuel pellet thermal conductivity model in the WCOBRA/TRAC-TF2 code used in the FSLOCA EM explicitly accounts for pellet TCD.

Three of the Title 10 of the Code of Federal Regulations (CFR) 50.46 criteria (peak cladding temperature (PCT), maximum local oxidation (MLO), and core-wide oxidation (CWO)) are considered directly in the FSLOCA EM. A high probability statement is developed for the PCT, MLO, and CWO that is needed to demonstrate compliance with 10 CFR 50.46 acceptance criteria (b)(1), (b)(2), and (b)(3) (Reference 3) via statistical methods. The MLO is defined as the sum of pre-transient corrosion and transient oxidation consistent with the position in Information Notice 98-29 (Reference 4). The coolable geometry acceptance criterion, 10 CFR 50.46 (b)(4), is assured by compliance with acceptance criteria (b)(1) and (b)(2), and demonstrating that fuel assembly grid deformation due to combined seismic and LOCA loads does not extend to the in-board fuel assemblies such that a coolable geometry is maintained.

The FSLOCA EM has been generically approved by the Nuclear Regulatory Commission (NRC) for Westinghouse 3-loop and 4-loop plants with cold leg Emergency Core Cooling System (ECCS) injection (Reference 1). Since Beaver Valley Unit 1 and Unit 2 are Westinghouse designed 3-loop plants with cold leg ECCS injection, the approved method is applicable.

This report summarizes the application of the Westinghouse FSLOCA EM to Beaver Valley Unit 1 and Unit 2. The application of the FSLOCA EM to Beaver Valley Unit 1 and Unit 2 is consistent with the NRC-approved methodology (Reference 1), with exceptions identified under Limitation and Condition Number 2 in Section 2.3. The application of the FSLOCA EM to Beaver Valley Unit 1 and Unit 2 is consistent with the conditions and limitations as identified in the NRC's Safety Evaluation Report (SER) for Reference 1, and is also applicable to the Beaver Valley Unit 1 and Unit 2 plant design and operating conditions.

Two separate analyses with the FSLOCA EM were performed for Beaver Valley Unit 1 and Unit 2 due to plant design differences. For example, the Beaver Valley Unit 1 reactor vessel has thermal shields and the Beaver Valley Unit 2 reactor vessel has neutron pads. In addition, the steam generator (SG) designs of the two units are different. The FSLOCA EM analyses are summarized in Reference 14 for Beaver Valley Unit 1 and Reference 15 for Beaver Valley Unit 2.

Both Energy Harbor and its analysis vendor (Westinghouse) have interface processes which identify plant configuration changes potentially impacting safety analyses. These interface processes, along with Westinghouse internal processes for assessing EM changes and errors, are used to identify the need for LOCA analysis impact assessments.

The major plant parameter and analysis assumptions used in the Beaver Valley Unit 1 analysis with the FSLOCA EM are provided in Tables 1a through 6a. The major plant parameter and analysis assumptions used in the Beaver Valley Unit 2 analysis with the FSLOCA EM are provided in Tables 1b through 6b.

2.0 METHOD OF ANALYSIS

2.1 FULL SPECTRUM LOCA Evaluation Model Development

In 1988, the NRC Staff amended the requirements of 10 CFR 50.46 (Reference 3 and Reference 6) and Appendix K, "ECCS Evaluation Models," to permit the use of a realistic EM to analyze the performance of the ECCS during a hypothetical LOCA. Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. After the rule change, Westinghouse developed and received approval for a best-estimate LBLOCA EM, which is discussed in Reference 8. The EM is referred to as the Code Qualification Document (CQD), and was developed following Regulatory Guide (RG) 1.157 (Reference 7). Subsequently, Westinghouse developed and received approval for another best-estimate LBLOCA EM which is discussed in Reference 12. This subsequent EM is referred to as the Automated Statistical Treatment of Uncertainty Method (ASTRUM), and was also developed following Regulatory Guide (RG) 1.157 (Reference 7).

When the FSLOCA EM was being developed, the NRC issued RG 1.203 (Reference 9) which expands on the principles of RG 1.157, while providing a more systematic approach to the development and assessment process of a PWR accident and safety analysis EM. Therefore, the development of the FSLOCA EM followed the Evaluation Model Development and Assessment Process (EMDAP), which is documented in RG 1.203. While RG 1.203 expands upon RG 1.157, there are certain aspects of RG 1.157 which are more detailed than RG 1.203; therefore, both RGs were used for the development of the FSLOCA EM.

2.2 WCOBRA/TRAC-TF2 Computer Code

The FSLOCA EM (Reference 1) uses the WCOBRA/TRAC-TF2 code to analyze the system thermal-hydraulic response for the full spectrum of break sizes. WCOBRA/TRAC-TF2 was created by combining a 1D module (TRAC-P) with a 3D module (based on Westinghouse modified COBRA-TF). The 1D and 3D modules include an explicit non-condensable gas transport equation. The use of TRAC-P allows for the extension of a two-fluid, six-equation formulation of the two-phase flow to the 1D loop components. This new code is WCOBRA/TRAC-TF2, where "TF2" is an identifier that reflects the use of a three-field (TF) formulation of the 3D module derived by COBRA-TF and a two-fluid (TF) formulation of the 1D module based on TRAC-P.

This best-estimate computer code contains the following features:

1. Ability to model transient three-dimensional flows in different geometries inside the reactor vessel
2. Ability to model thermal and mechanical non-equilibrium between phases
3. Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes
4. Ability to represent important reactor and plant components such as fuel rods, SGs, reactor coolant pumps (RCPs), etc.

A detailed assessment of the computer code WCOBRA/TRAC-TF2 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 for a LOCA. Modeling of a LOCA introduces additional uncertainties which are identified and quantified in the plant-specific analysis. The reactor vessel and loop noding scheme used in the FSLOCA EM is consistent with the noding scheme used for the experiment simulations that form the validation basis for the physical models in the code. Such noding choices have been justified by assessing the model against large and full scale experiments.

2.3 Compliance with FSLOCA EM Limitations and Conditions

The NRC's SER for Reference 1 contains 15 limitations and conditions on the NRC-approved FSLOCA EM. A summary of each limitation and condition and how it was met is provided below.

Limitation and Condition Number 1

Summary

The FSLOCA EM is not approved to demonstrate compliance with 10 CFR 50.46 acceptance criterion (b)(5) related to the long-term cooling.

Compliance

The analyses for Beaver Valley Unit 1 and Unit 2 with the FSLOCA EM is only being used to demonstrate compliance with 10 CFR 50.46 (b)(1) through (b)(4).

Limitation and Condition Number 2

Summary

The FSLOCA EM is approved for the analysis of Westinghouse-designed 3-loop and 4-loop PWRs with cold-side injection. Analyses should be executed consistent with the approved method, or any deviations from the approved method should be described and justified.

Compliance

Beaver Valley Unit 1 and Unit 2 are Westinghouse-designed 3-loop PWRs with cold-side injection, so they are within the NRC-approved methodology. The analyses for Beaver Valley Unit 1 and Unit 2 utilized the NRC-approved FSLOCA methodology, with the following three exceptions: the changes which were previously transmitted to the NRC pursuant to 10 CFR 50.46 in LTR-NRC-18-30 (Reference 5) were incorporated into the analyses, the modeled fuel average temperatures bound the PAD5 data, and the blowdown energy release assumption was modified to use a plant-specific bounding value.

After completion of the analyses for Beaver Valley Unit 1 and Unit 2, two errors were discovered in the WCOBRA/TRAC-TF2 code that can occur under certain conditions. These errors were found to have a negligible impact on analysis results with the FSLOCA EM as described in LTR-NRC-19-6 (Reference 13).

The treatment for the uncertainty in the gamma energy redistribution is discussed on pages 29-75 and 29-76 of WCAP-16996-P-A, Revision 1 (Reference 1), and the equation for the assumed increase in hot rod and hot assembly relative power is presented on page 29-76. The power increase in the hot rod and hot assembly due to energy redistribution in the application of the FSLOCA EM to Beaver Valley Unit 1 and Unit 2, was calculated incorrectly. This error resulted in a 0% to 5% underestimation of the modeled hot rod and hot assembly rod linear heat rates on a run-specific basis, depending on the as-sampled value for the uncertainty. The effect of the error correction was evaluated against the results of the application of the FSLOCA EM to Beaver Valley Unit 1 and Unit 2.

The error correction has only a limited impact on the power modeled for a single assembly in the core. As such, there is a negligible impact of the error correction on the system thermal-hydraulic response during the postulated LOCA.

For Region I, the primary impact of the error correction is on the rate of cladding heatup above the two-phase mixture level in the core during the boiloff phase. The PCT impact was assessed using run-specific PCT versus linear heat rate relationships and the run-specific hot rod and hot assembly linear heat rate increase that would result from the error correction. Using this approach, the correction of the error was estimated to increase the Region I analysis PCT by 1°F for Beaver Valley Unit 1, leading to a final result of 1236°F for the Region I analysis. Using the same approach, the correction of the error was estimated to increase the Region I analysis PCT by 5°F for Beaver Valley Unit 2, leading to a final result of 1277°F for the Region I analysis.

For Region II, parametric PWR sensitivity studies, derived from a subset of uncertainty analysis simulations covering various design features and fuel arrays, were examined to determine the sensitivity of the analysis results to the error correction. The PCT impact from the error correction was found to be different for the different transient phases (i.e., blowdown versus reflood) based on the PWR sensitivity studies and existing power distribution sensitivity studies. Based on the results from the PWR sensitivity studies, the correction of the error is estimated to increase the Region II analysis PCT by 31°F for Beaver Valley Unit 1, leading to an analysis result of 2001°F for the Region II analysis assuming offsite power available (OPA) and 2006°F for the Region II analysis assuming loss-of-offsite power (LOOP). Based on the results from the PWR sensitivity studies, the correction of the error is estimated to increase the Region II analysis PCT by 31°F for Beaver Valley Unit 2, leading to an analysis result of 1856°F for the Region II analysis assuming OPA and 1854°F for the Region II analysis assuming LOOP.

All of the analysis results including the effects of the gamma energy redistribution error correction continue to maintain compliance with the 10 CFR 50.46 acceptance criteria.

Limitation and Condition Number 3

Summary

For Region II, the containment pressure calculation will be executed in a manner consistent with the approved methodology (i.e., the COCO or LOTIC2 model will be based on appropriate plant-specific design parameters and conditions, and engineered safety features which can reduce pressure are modeled). This includes utilizing a plant-specific initial containment temperature, and only taking credit for containment coatings which are qualified and outside of the break zone-of-influence.

Compliance

The containment pressure calculations for the Beaver Valley Unit 1 and Unit 2 analyses use the COCO model and were performed consistent with the NRC-approved methodology. Appropriate design parameters and conditions were modeled, as were the engineered safety features which can reduce the containment pressure. A plant-specific initial temperature associated with normal full-power operating conditions was modeled for each unit, and no coatings were credited on any of the containment structures.

Limitation and Condition Number 4

Summary

The decay heat uncertainty multiplier will be **[**
]^{a,c} The analysis simulations for the FSLOCA EM will not be executed for longer than 10,000 seconds following reactor trip unless the decay heat model is appropriately justified. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region I and Region II analysis results will be provided in the analysis submittal in units of sigma and absolute units.

Compliance

Consistent with the NRC-approved methodology, the decay heat uncertainty multiplier was **[**
]^{a,c} for the Beaver Valley Unit 1 and Unit 2 analyses. The analysis simulations were all executed for no longer than 10,000 seconds following reactor trip. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region I and Region II analysis results have been provided in units of sigma and approximate absolute units in Table 10a for Unit 1 and Table 10b for Unit 2.

Limitation and Condition Number 5

Summary

The maximum assembly and rod length-average burnup is limited to [ ][a,c] respectively.

Compliance

The maximum analyzed assembly and rod length-average burnup were less than or equal to [ ][a,c] respectively, for Beaver Valley Unit 1 and Unit 2.

Limitation and Condition Number 6

Summary

The fuel performance data for analyses with the FSLOCA EM should be based on the PAD5 code (at present), which includes the effect of thermal conductivity degradation. The nominal fuel pellet average temperatures and rod internal pressures should be the maximum values, and the generation of all the PAD5 fuel performance data should adhere to the NRC-approved PAD5 methodology.

Compliance

PAD5 fuel performance data were utilized in the Beaver Valley Unit 1 and Unit 2 analyses with the FSLOCA EM. The analyzed fuel pellet average temperatures are bounding compared to the upper bound maximum values calculated in accordance with Section 7.5.1 of Reference 2, and the analyzed rod internal pressures were calculated in accordance with Section 7.5.2 of Reference 2.

Limitation and Condition Number 7

Summary

The YDRAG uncertainty parameter should be [ ][a,c]

Compliance

Consistent with the NRC-approved methodology, the YDRAG uncertainty parameter was [ ][a,c] for the Beaver Valley Unit 1 and Unit 2 Region I analyses.

Limitation and Condition Number 8

Summary

The [ ][a,c]

Compliance

Consistent with the NRC-approved methodology, the [ ][a,c] for the Beaver Valley Unit 1 and Unit 2 Region I analyses.

Limitation and Condition Number 9

Summary

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to confirm that the **[** **]**^{a,c} for the plant design being analyzed. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.

Compliance

Beaver Valley Unit 1 and Unit 2 are Westinghouse-designed 3-loop PWRs, so this Limitation and Condition is not applicable.

Limitation and Condition Number 10

Summary

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to: 1) demonstrate that no unexplained behavior occurs in the predicted safety criteria across the region boundary, and 2) ensure that the **[** **]**^{a,c} must cover the equivalent 2 to 4-inch break range using RCS-volume scaling relative to the demonstration plant. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.

Additionally, the minimum sampled break area for the analysis of Region II should be 1 ft^2.

Compliance

Beaver Valley Unit 1 and Unit 2 are Westinghouse-designed 3-loop PWRs, so this part of the Limitation and Condition is not applicable.

The minimum sampled break area for the Beaver Valley Unit 1 and Unit 2 Region II analyses was 1 ft^2.

Limitation and Condition Number 11

Summary

There are various aspects of this Limitation and Condition, which are summarized below:

1. The **[** **]**^{a,c} the Region I and Region II analysis seeds, and the analysis inputs will be declared and documented prior to performing the Region I and Region II uncertainty analyses. The **[** **]**^{a,c} and the Region I and Region II analysis seeds will not be changed throughout the remainder of the analysis once they have been declared and documented.
2. If the analysis inputs are changed after they have been declared and documented, for the intended purpose of demonstrating compliance with the applicable acceptance criteria, then the changes and associated rationale for the changes will be provided in the analysis submittal. Additionally, the preliminary values for PCT, MLO, and CWO which caused the input changes will be provided. These preliminary values are not subject to Appendix B verification, and archival of the supporting information for these preliminary values is not required.

3. Plant operating ranges which are sampled within the uncertainty analysis will be provided in the analysis submittal for both regions.

Compliance

This Limitation and Condition was met for the Beaver Valley Unit 1 and Unit 2 analyses as follows:

1. The **[** **]**^{a,c} the Region I and Region II analysis seeds, and the analysis inputs were declared and documented prior to performing the Region I and Region II uncertainty analyses. The **[** **]**^{a,c} and the Region I and Region II analysis seeds were not changed once they were declared and documented.
2. The analysis inputs were not changed once they were declared and documented.
3. The plant operating ranges which were sampled within the uncertainty analyses are provided for Beaver Valley Unit 1 in Table 1a and for Unit 2 in Table 1b.

Limitation and Condition Number 12

Summary

The plant-specific dynamic pressure loss from the steam generator secondary-side to the main steam safety valves must be adequately accounted for in analysis with the FSLOCA EM.

Compliance

A bounding plant-specific dynamic pressure loss from the steam generator secondary-side to the main steam safety valves (MSSVs) was modeled in the Beaver Valley Unit 1 and Unit 2 analyses.

Limitation and Condition Number 13

Summary

In plant-specific models for analysis with the FSLOCA EM: 1) the **[** **]**^{a,c} and 2) the **[** **]**^{a,c}

Compliance

The **[** **]**^{a,c} in the analyses for Beaver Valley Unit 1 and Unit 2. The **[** **]**^{a,c} in the analyses.

Limitation and Condition Number 14

Summary

For analyses with the FSLOCA EM to demonstrate compliance against the current 10 CFR 50.46 oxidation criterion, the transient time-at-temperature will be converted to an equivalent cladding reacted (ECR) using either the Baker-Just or the Cathcart-Pawel correlation. In either case, the pre-transient corrosion will be added to the LOCA transient oxidation. If the Cathcart-Pawel correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 13 percent limit. If the Baker-

Just correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 17 percent limit.

<u>Compliance</u>

For the Beaver Valley Unit 1 and Unit 2 analyses, the Baker-Just correlation was used in each transient calculation to convert the LOCA transient time-at-temperature to an ECR. The resulting LOCA transient ECR was then added to the pre-existing corrosion for comparison against the 10 CFR 50.46 local oxidation acceptance criterion of 17%.

<u>Limitation and Condition Number 15</u>

<u>Summary</u>

The Region II analysis will be executed twice; once assuming LOOP and once assuming OPA. The results from both analysis executions should be shown to be in compliance with the 10 CFR 50.46 acceptance criteria.

The **[** **]**^{a,c}

<u>Compliance</u>

The Region II uncertainty analyses for Beaver Valley Unit 1 and Unit 2 were performed twice; once assuming a LOOP and once assuming OPA. The results from both analyses that were performed are in compliance with the 10 CFR 50.46 acceptance criteria (see Section 5.0).

The **[**
]^{a,c}

3.0 REGION I ANALYSIS

3.1 Description of Representative Transient

The small break LOCA transient can be divided into time periods in which specific phenomena are occurring, as discussed below.

Blowdown

The rapid depressurization of the RCS coincides with liquid flow through the break. Following the reactor trip on the low pressurizer pressure setpoint, the pressurizer drains, and safety injection is initiated on the low pressurizer pressure SI setpoint. After reaching this setpoint and applying the safety injection delays, high pressure safety injection flow begins. Phase separation begins in the upper head and upper plenum near the end of this period until the entire RCS eventually reaches saturation, ending the rapid depressurization slightly above the steam generator secondary side pressure near the modeled MSSV setpoint.

Natural Circulation

This quasi-equilibrium phase persists while the RCS pressure remains slightly above the secondary side pressure. The system drains from the top down, and while significant mass is continually lost through the break, the vapor generated in the core is trapped in the upper regions by the liquid remaining in the crossover leg loop seals. Throughout this period, the core remains covered by a two-phase mixture and the fuel cladding temperatures remain at the saturation temperature level.

Loop Seal Clearance

As the system drains, the liquid levels in the downhill side of the pump suction (crossover leg) become depressed all the way to the bottom elevations of the piping, allowing the steam trapped during the natural circulation phase to vent to the break (i.e., a process called loop seal clearance). The break flow and the flow through the RCS loop that cleared become primarily vapor. Relief of a static head imbalance allows for a quick but temporary recovery of liquid levels in the inner portion of the reactor vessel.

Boil-Off

With a vapor vent path established after the loop seal clearance, the RCS depressurizes at a rate controlled by the critical flow, which continues to be a primarily high quality mixture of water and steam. The RCS pressure remains high enough such that safety injection flow cannot make up for the primary system fluid inventory lost through the break, leading to core uncovery and a fuel rod cladding temperature heatup.

Core Recovery

The RCS pressure continues to decrease, and once it reaches that of the accumulator gas pressure, the introduction of additional ECCS water from the accumulators replenishes the reactor vessel inventory and recovers the core mixture level. The transient is considered over as the break flow is compensated by the injected flow.

3.2 Analysis Results

The Beaver Valley Unit 1 and Unit 2 Region I analyses were performed in accordance with the NRC-approved methodology in Reference 1, with exceptions identified under Limitation and Condition Number 2 in Section 2.3. The transient that produced the analysis PCT result for both Beaver Valley Unit 1 and Unit 2 is a cold leg break with a break diameter of 2.6-inches. The most limiting ECCS single failure of one ECCS train is assumed in both analyses as identified in Item 5.0 in Tables 1a and 1b. For both units, control rod drop is modeled for breaks less than 1 square foot assuming a 2 second signal delay time and a 2.7 second rod drop time, and RCP trip is modeled coincident with reactor trip on the low pressurizer pressure setpoint for LOOP transients. When the low pressurizer pressure SI setpoint is reached, there is a delay to account for emergency diesel generator start-up, filling headers, etc., after which safety injection is initiated into the reactor coolant system.

The results of the Beaver Valley Region I uncertainty analyses are summarized in Table 7a (Unit 1) and Table 7b (Unit 2). These tables show the analysis-of-record PCT result, which is the sum of the uncertainty analysis result plus the impact of the gamma energy redistribution uncertainty error correction. The MLO and CWO were confirmed to maintain compliance with the 10 CFR 50.46 acceptance criteria, including the effects of the error correction. The sampled decay heat uncertainty multipliers for the Region I analysis cases are provided in Table 10a (Unit 1) and Table 10b (Unit 2).

Table 8a (Unit 1) and Table 8b (Unit 2) contain sequences of events for the transients that produced the Region I analysis PCT result. Figures 1a through 13a (Unit 1) and Figures 1b through 13b (Unit 2) illustrate the calculated key transient response parameters for these transients. Note that the figures presenting the analysis results correspond to the uncertainty analysis results (not the gamma energy redistribution error correction results).

4.0 REGION II ANALYSIS

4.1 Description of Representative Transient

A large-break LOCA transient can be divided into phases in which specific phenomena are occurring. A convenient way to divide the transient is in terms of the various heatup and cooldown phases that the fuel assemblies undergo. For each of these phases, specific phenomena and heat transfer regimes are important, as discussed below.

Blowdown – Critical Heat Flux (CHF) Phase

In this phase, the break flow is subcooled, the discharge rate of coolant from the break is high, the core flow reverses, the fuel rods go through departure from nucleate boiling (DNB), and the cladding rapidly heats up and the reactor is shut down due to the core voiding.

The regions of the RCS with the highest initial temperatures (upper core, upper plenum, and hot legs) begin to flash during this period. This phase is terminated when the water in the lower plenum and downcomer begins to flash. The mixture level swells and a saturated mixture is pushed into the core by the intact loop RCPs, still rotating in single-phase liquid. As the fluid in the cold leg reaches saturation conditions, the discharge flow rate at the break decreases significantly.

Blowdown – Upward Core Flow Phase

Heat transfer is increased as the two-phase mixture is pushed into the core. The break discharge rate is reduced because the fluid becomes saturated at the break. This phase ends as the lower plenum mass is depleted, the fluid in the loops become two-phase, and the RCP head degrades.

Blowdown – Downward Core Flow Phase

The break flow begins to dominate and pulls flow down through the core as the RCP head degrades due to increased voiding, while liquid and entrained liquid flows also provide core cooling. Heat transfer in this period may be enhanced by liquid flow from the upper head. Once the system has depressurized to less than the accumulator cover pressure, the accumulators begin to inject cold water into the cold legs. During this period, due to steam upflow in the downcomer, a portion of the injected ECCS water is bypassed around the downcomer and sent out through the break. As the system pressure continues to decrease, the break flow and consequently the downward core flow are reduced. The system pressure approaches the containment pressure at the end of this last period of the blowdown phase.

During this phase, the core begins to heat up as the system approaches containment pressure, and the phase ends when the reactor vessel begins to refill with ECCS water.

Refill Phase

The core continues to heat up as the lower plenum refills with ECCS water. This phase is characterized by a rapid increase in fuel cladding temperature at all elevations due to the lack of liquid and steam flow in the core region. The water completely refills the lower plenum and the refill phase ends. As ECCS water enters the core, the fuel rods in the lower core region begin to quench and liquid entrainment begins, resulting in increased fuel rod heat transfer.

Reflood Phase

During the early reflood phase, the accumulators begin to empty and nitrogen is discharged into the RCS. The nitrogen surge forces water into the core, which is then evaporated, causing system re-pressurization and a temporary reduction of pumped ECCS flow; this re-pressurization is illustrated by a brief and oscillatory increase in RCS pressure in early reflood. During this time, core cooling may increase due to vapor generation and liquid entrainment, but conversely the early reflood pressure spike results in a brief increase in loss of mass out through the broken cold leg.

The pumped ECCS water aids in the filling of the downcomer throughout the reflood period. As the quench front progresses further into the core, the PCT elevation moves increasingly higher in the fuel assembly.

The axial power distributions for the analysis cases are presented in Figures 27a and 27b for Beaver Valley Units 1 and 2, respectively. The axial power distribution for the Unit 2 analysis case (Figure 27b), which is sampled to be a beginning-of-cycle case, contains small spikes at the top and bottom of the core that are not present on the Unit 1 analysis case (Figure 27a). These spikes represent the effects of the burnable absorber, which is still significant at the beginning of the cycle. As the sampled time-in-cycle progresses, the effect of the burnable absorber is dampened.

As the transient progresses, continued injection of pumped ECCS water refloods the core, effectively removes the reactor vessel metal mass stored energy and core decay heat, and leads to an increase in the reactor vessel fluid mass. Eventually the core inventory increases enough that liquid entrainment is able to quench all the fuel assemblies in the core.

4.2 Analysis Results

The Beaver Valley Unit 1 and Unit 2 Region II analyses were performed in accordance with the NRC-approved methodology in Reference 1, with exceptions identified under Limitation and Condition Number 2 in Section 2.3. The analyses for Beaver Valley Unit 1 and Unit 2 were performed assuming both LOOP and OPA, and the results of both of the LOOP and OPA analyses are compared to the 10 CFR 50.46 acceptance criteria. The most limiting ECCS single failure of one ECCS train is assumed in both analyses as identified in Item 5.0 in Tables 1a and 1b.

The results of the Beaver Valley Region II LOOP and OPA uncertainty analyses are summarized in Table 7a (Unit 1) and Table 7b (Unit 2). These tables show the analysis-of-record PCT result, which is the sum of the uncertainty analysis result plus the impact of the gamma energy redistribution uncertainty error correction. The MLO and CWO were confirmed to maintain compliance with the 10 CFR 50.46 acceptance criteria, including the effects of the error correction. The sampled decay heat uncertainty multipliers for the Region II analysis cases are provided in Table 10a (Unit 1) and Table 10b (Unit 2).

Table 9a (Unit 1) and Table 9b (Unit 2) contain the sequences of events for the transients that produced the more limiting analysis PCT result relative to the offsite power assumption. Figures 14a through 28a (Unit 1) and Figures 14b through 28b (Unit 2) illustrate the key response parameters for these transients. Note that the figures presenting the analysis results correspond to the uncertainty analysis results (not the gamma energy redistribution error correction results).

The containment pressure is calculated for each LOCA transient in the analysis using the COCO code (References 10 and 11). The COCO containment code is integrated into the WCOBRA/TRAC-TF2 thermal-hydraulic code. The transient-specific mass and energy releases calculated by the thermal-hydraulic code at the end of each timestep are transferred to COCO. COCO then calculates the containment pressure based on the containment model (the inputs are summarized in Tables 2a and 3a (Unit 1) and Tables 2b and 3b (Unit 2)) and the mass and energy releases, and transfers the pressure back to the thermal-hydraulic code as a boundary condition at the break, consistent with the methodology in Reference 1. The containment model for COCO calculates a conservatively low containment pressure, including the effects of all the installed pressure reducing systems and processes such as assuming all trains of containment spray are operable. The containment backpressure for the transient that produced the analysis PCT result is provided in Figure 22a (Unit 1) and Figure 22b (Unit 2).

5.0 COMPLIANCE WITH 10 CFR 50.46

It must be demonstrated that there is a high level of probability that the following criteria in 10 CFR 50.46 are met:

(b)(1) The analysis PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT is less than 2,200°F, the analysis with the FSLOCA EM confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Cladding Temperature does not exceed 2,200°F," is demonstrated.

The results for Beaver Valley are shown in Table 7a (Unit 1) and Table 7b (Unit 2).

(b)(2) The analysis MLO corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Since the resulting MLO is less than 17 percent when converting the time-at-temperature to an equivalent cladding reacted using the Baker-Just correlation and adding the pre-transient corrosion, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Maximum Local Oxidation of the cladding does not exceed 17 percent," is demonstrated.

The results for Beaver Valley are shown in Table 7a (Unit 1) and Table 7b (Unit 2).

(b)(3) The analysis CWO corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. Since the resulting CWO is less than 1 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation does not exceed 1 percent," is demonstrated.

The results for Beaver Valley are shown in Table 7a (Unit 1) and Table 7b (Unit 2).

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains in a coolable geometry.

This criterion is met by demonstrating compliance with criteria (b)(1) and (b)(2), and by assuring that fuel assembly grid deformation due to combined LOCA and seismic loads is specifically addressed. Criteria (b)(1) and (b)(2) have been met for Beaver Valley as shown in Table 7a (Unit 1) and Table 7b (Unit 2).

It is discussed in Section 32.1 of the NRC-approved FSLOCA EM (Reference 1) that the effects of LOCA and seismic loads on the core geometry do not need to be considered unless fuel assembly grid deformation extends beyond the core periphery (i.e., deformation in a fuel assembly with no sides adjacent to the core baffle plates). Inboard grid deformation due to combined LOCA and seismic loads is not calculated to occur for Beaver Valley Unit 1 and Unit 2.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS.

Long-term cooling is dependent on the demonstration of the continued delivery of cooling water to the core. The actions that are currently in place to maintain long-term cooling are not impacted by the application of the NRC-approved FSLOCA EM (Reference 1).

Based on the analysis results for Region I and Region II presented in Table 7a (Unit 1) and Table 7b (Unit 2) for Beaver Valley, it is concluded that Beaver Valley Unit 1 and Unit 2 comply with the criteria in 10 CFR 50.46.

6.0 REFERENCES

1. “Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology),” WCAP-16996-P-A, Revision 1, November 2016.
2. “Westinghouse Performance Analysis and Design Model (PAD5),” WCAP-17642-P-A, Revision 1, November 2017.
3. “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors,” 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 1974.
4. “Information Notice 98-29: Predicted Increase in Fuel Rod Cladding Oxidation,” USNRC, August 1998.
5. “U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2017,” LTR-NRC-18-30, July 2018.
6. “Emergency Core Cooling Systems: Revisions to Acceptance Criteria,” Federal Register, V53, N180, pp. 35996-36005, September 1988.
7. “Best Estimate Calculations of Emergency Core Cooling System Performance,” Regulatory Guide 1.157, USNRC, May 1989.
8. “Code Qualification Document for Best Estimate LOCA Analysis,” WCAP-12945-P-A Volume 1, Revision 2 and Volumes 2-5, Revision 1, March 1998.
9. “Transient and Accident Analysis Methods,” Regulatory Guide 1.203, USNRC, December 2005.
10. “Westinghouse Emergency Core Cooling System Evaluation Model – Summary,” WCAP-8339, June 1974.
11. “Containment Pressure Analysis Code (COCO),” WCAP-8327, June 1974.
12. “Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment Of Uncertainty Method (ASTRUM),” WCAP-16009-P-A, January 2005.
13. “U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2018,” LTR-NRC-19-6, February 2019.
14. WCAP-18612-P, “Engineering Summary Report of the Beaver Valley Unit 1 Loss-of-Coolant Accident (LOCA) Analysis with the FULL SPECTRUM LOCA (FSLOCA) Methodology,” March 2021.
15. WCAP-18611-P, “Engineering Summary Report of the Beaver Valley Unit 2 Loss-of-Coolant Accident (LOCA) Analysis with the FULL SPECTRUM LOCA (FSLOCA) Methodology,” January 2021.

Table 1a. Plant Operating Range Analyzed and Key Parameters for Beaver Valley Unit 1

Parameter		As-Analyzed Value or Range
1.0	**Core Parameters**	
	a) Core power	≤ 2900 MWt ± 0.6% Uncertainty
	b) Fuel type	17x17 RFA-2 Fuel with Intermediate Flow Mixers (IFMs), Integral Fuel Burnable Absorbers (IFBA) or Non-IFBA, ZIRLO® and Optimized ZIRLO™ Cladding (see Note 1)
	c) Maximum total core peaking factor (F_Q), including uncertainties	2.4
	d) Maximum hot channel enthalpy rise peaking factor ($F_{\Delta H}$), including uncertainties	1.7
	e) Axial flux difference (AFD) band at 100% power	± 10 %
	f) Maximum transient operation fraction	1.0
2.0	**Reactor Coolant System Parameters**	
	a) Thermal design flow (TDF)	87,200 gpm/loop
	b) Vessel average temperature (T_{AVG})	561.7°F ≤ T_{AVG} ≤ 583.9°F
	c) Pressurizer pressure (P_{RCS})	2200 psia ≤ P_{RCS} ≤ 2300 psia
	d) Reactor coolant pump (RCP) model and power	Model 93A, 6000 hp
3.0	**Containment Parameters**	
	a) Containment modeling	Region I: Constant pressure equal to initial containment pressure Region II: Calculated for each transient using transient-specific mass and energy releases and the containment model built using information in Tables 2a and 3a
4.0	**Steam Generator (SG) and Secondary Side Parameters**	
	a) Steam generator tube plugging level	≤ 15%
	b) Main steam safety valve (MSSV) nominal set pressures, uncertainty and accumulation	Table 6a
	c) Main feedwater temperature	Nominal (427.5°F)
	d) Auxiliary feedwater temperature	Nominal (80.0°F)
	e) Auxiliary feedwater flow rate	98 gpm/SG

5.0	**Safety Injection (SI) Parameters**	
	a) Single failure configuration	ECCS: Loss of one train of pumped ECCS Region II containment pressure: All containment spray trains are available
	b) Safety injection temperature (T_{SI})	45°F ≤ (T_{SI}) ≤ 65°F
	c) Low pressurizer pressure safety injection safety analysis limit	1745 psia
	d) Initiation delay time from low pressurizer pressure SI setpoint to full SI flow	≤ 22 seconds (OPA) or ≤ 32 seconds (LOOP)
	e) Safety injection flow	Minimum flows in Table 4a (Region I) or Table 5a (Region II)
6.0	**Accumulator Parameters**	
	a) Accumulator temperature (T_{ACC})	70°F ≤ T_{ACC} ≤ 108°F
	b) Accumulator water volume (V_{ACC})	893 ft^3 ≤ V_{ACC} ≤ 1022 ft^3
	c) Accumulator pressure (P_{ACC})	575 psia ≤ P_{ACC} ≤ 716 psia
	d) Accumulator boron concentration	≥ 2300 ppm
7.0	**Reactor Protection System Parameters**	
	a) Low pressurizer pressure reactor trip signal processing time	≤ 2 seconds
	b) Low pressurizer pressure reactor trip setpoint	1934.7 psia

Notes:

1. The analysis is applicable to a full core of ZIRLO fuel cladding, a full core of Optimized ZIRLO fuel cladding, and a mixed core of ZIRLO and Optimized ZIRLO fuel cladding.

Table 1b. Plant Operating Range Analyzed and Key Parameters for Beaver Valley Unit 2

Parameter		As-Analyzed Value or Range
1.0	**Core Parameters**	
	a) Core power	≤ 2900 MWt ± 0.6% Uncertainty
	b) Fuel type	17x17 RFA Fuel with Intermediate Flow Mixers (IFMs), Integral Fuel Burnable Absorbers (IFBA) or Non-IFBA, ZIRLO® and Optimized ZIRLO™ Cladding (see Note 1)
	c) Maximum total core peaking factor (F_Q), including uncertainties	2.4
	d) Maximum hot channel enthalpy rise peaking factor ($F_{\Delta H}$), including uncertainties	1.7
	e) Axial flux difference (AFD) band at 100% power	± 10%
	f) Maximum transient operation fraction	1.0
2.0	**Reactor Coolant System Parameters**	
	a) Thermal design flow (TDF)	87,200 gpm/loop
	b) Vessel average temperature (T_{AVG})	561.8°F ≤ T_{AVG} ≤ 583.8°F
	c) Pressurizer pressure (P_{RCS})	2200 psia ≤ P_{RCS} ≤ 2300 psia
	d) Reactor coolant pump (RCP) model and power	Model 93A, 6000 hp
3.0	**Containment Parameters**	
	a) Containment modeling	Region I: Constant pressure equal to initial containment pressure Region II: Calculated for each transient using transient-specific mass and energy releases and the containment model built using information in Tables 2b and 3b
4.0	**Steam Generator (SG) and Secondary Side Parameters**	
	a) Steam generator tube plugging level	≤ 20%
	b) Main steam safety valve (MSSV) nominal set pressures, uncertainty and accumulation	Table 6b
	c) Main feedwater temperature	Nominal (427.5°F)
	d) Auxiliary feedwater temperature	Nominal (70.0°F)
	e) Auxiliary feedwater flow rate	98 gpm/SG
5.0	**Safety Injection (SI) Parameters**	
	a) Single failure configuration	ECCS: Loss of one train of pumped ECCS Region II containment pressure: All containment spray trains are available

Table 1b. Plant Operating Range Analyzed and Key Parameters for Beaver Valley Unit 2

Parameter		As-Analyzed Value or Range
	b) Safety injection temperature (T_{SI})	45°F ≤ (T_{SI}) ≤ 65°F
	c) Low pressurizer pressure safety injection safety analysis limit	1760 psia
	d) Initiation delay time from low pressurizer pressure SI setpoint to full SI flow	≤ 22 seconds (OPA) or ≤ 32 seconds (LOOP)
	e) Safety injection flow	Minimum flows in Table 4b (Region I) or Table 5b (Region II)
6.0	**Accumulator Parameters**	
	a) Accumulator temperature (T_{ACC})	70°F ≤ T_{ACC} ≤ 108°F
	b) Accumulator water volume (V_{ACC})	921 ft^3 ≤ V_{ACC} ≤ 1072 ft^3
	c) Accumulator pressure (P_{ACC})	575 psia ≤ P_{ACC} ≤ 716 psia
	d) Accumulator boron concentration	≥ 2300 ppm
7.0	**Reactor Protection System Parameters**	
	a) Low pressurizer pressure reactor trip signal processing time	≤ 2 seconds
	b) Low pressurizer pressure reactor trip setpoint	1935 psia

Notes:

1. The analysis is applicable to a full core of ZIRLO fuel cladding, a full core of Optimized ZIRLO fuel cladding, and a mixed core of ZIRLO and Optimized ZIRLO fuel cladding.

Table 2a. Containment Data Used for Region II Calculation of Containment Pressure for Beaver Valley Unit 1

Parameter	Value
Maximum containment net free volume	1,800,000 ft^3
Minimum initial containment temperature at full power operation	70°F
Refueling water storage tank (RWST) temperature for containment spray (T_{RWST})	45°F ≤ T_{RWST} ≤ 65°F
Minimum RWST temperature for broken loop spilling SI	45°F
Minimum containment outside air / ground temperature	-20°F (outside air) 32°F (outside ground)
Minimum initial containment pressure at normal full power operation	12.8 psia
Minimum containment spray pump initiation delay from containment high pressure signal time	≥ 23 seconds (OPA) or ≥ 38 seconds (LOOP)
Maximum containment spray flow rate from all pumps	Pressure Dependent Flows 0.0 psig, 4982 gpm 45.0 psig, 4040 gpm
Maximum number of containment fan coolers in operation during LOCA transient	0 (see Note 1)
Maximum number of containment venting lines (including purge lines, pressure relief lines or any others) which can be OPEN at onset of transient at full power operation	0
Containment walls / heat sink properties	Table 3a
SI spilling flows	230.81 lbm/sec

Notes:

1. The purpose of the fan coolers is to maintain the containment temperature during normal operation below 108°F and they are not required to reduce the containment pressure following an accident. They may be operational during the first several seconds of the large-break LOCA transient before being stopped on a containment high-high pressure signal. Due to the limited time the fan coolers would operate during a LOCA transient, as well as their very small heat removal capacity compared to the initial energy discharge into the containment they are considered to have a negligible impact on the LOCA transient and are not modeled in the analysis.

Table 2b. Containment Data Used for Region II Calculation of Containment Pressure for Beaver Valley Unit 2

Parameter	Value
Maximum containment net free volume	1,800,814 ft^3
Minimum initial containment temperature at full power operation	70°F
Refueling water storage tank (RWST) temperature for containment spray (T_{RWST})	45°F ≤ T_{RWST} ≤ 65°F
Minimum RWST temperature for broken loop spilling SI	45°F
Minimum containment outside air / ground temperature	-20°F (outside air) 32°F (outside ground)
Minimum initial containment pressure at normal full power operation	12.8 psia
Minimum containment spray pump initiation delay from containment high pressure signal time	≥ 32.2 seconds (OPA) or ≥ 56.4 seconds (LOOP)
Maximum containment spray flow rate from all pumps	4399 gpm
Maximum number of containment fan coolers in operation during LOCA transient	0 (see Note 1)
Maximum number of containment venting lines (including purge lines, pressure relief lines or any others) which can be OPEN at onset of transient at full power operation	0
Containment walls / heat sink properties	Table 3b
SI spilling flows	240.0 lbm/sec

Notes:

1. The purpose of the fan coolers is to maintain the containment temperature during normal operation below 108°F and they are not required to operate during accident conditions. They may be operational during the first several seconds of the large-break LOCA transient before being stopped on SI signal. Due to the limited time the fan coolers would operate during a LOCA transient, as well as their very small heat removal capacity compared to the initial energy discharge into the containment they are considered to have a negligible impact on the LOCA transient and are not modeled in the analysis.

Table 3a. Containment Heat Sink Data Used for Region II Calculation of Containment Pressure for Beaver Valley Unit 1

Wall	Area (ft^2)	Thickness (in)	Material
1	133,078	12	Concrete
2	18,097	0.179	Stainless Steel
3	22,041	0.0625	Galvanized Steel
4	15,856	0.125	Galvanized Steel
5	7,034	0.0647	Carbon Steel
6	67,479	0.125	Carbon Steel
7	42,469	0.375 54	Carbon Steel Concrete
8	19,638	0.5 30	Carbon Steel Concrete
9	9,036	1 30	Carbon Steel Concrete
10	11,251	24 0.25 120	Concrete Carbon Steel Concrete
11	28,514	0.1883	Carbon Steel
12	45,689	0.2565	Carbon Steel
13	21,484	0.3233	Carbon Steel
14	6,697	0.25 48	Stainless Steel Concrete
15	1,674	1 48	Stainless Steel Concrete
16	32,214	0.438	Carbon Steel
17	11,269	0.6064	Carbon Steel
18	2,615	1.032	Carbon Steel
19	3,803	1.4683	Carbon Steel
20	7,648	4.593	Carbon Steel
21	4,258	0.1875 54	Carbon Steel Concrete
22	33,465	0.121	Galvanized Steel

Table 3b. Containment Heat Sink Data Used for Region II Calculation of Containment Pressure for Beaver Valley Unit 2

Wall	Area (ft^2)	Thickness (in)	Material
1	116,295	12	Concrete
2	19,973	0.166	Stainless Steel
3	21,935	0.0625	Galvanized Steel
4	15,836	0.125	Galvanized Steel
5	6,406	0.0625	Carbon Steel
6	67,776	0.125	Carbon Steel
7	42,469	0.375 54	Carbon Steel Concrete
8	19,638	0.5 30	Carbon Steel Concrete
9	9,036	1 30	Carbon Steel Concrete
10	11,251	24 0.25 120	Concrete Carbon Steel Concrete
11	24,920	0.1879	Carbon Steel
12	45,938	0.2565	Carbon Steel
13	20,767	0.3158	Carbon Steel
14	6,697	0.25 48	Stainless Steel Concrete
15	1,674	1 48	Stainless Steel Concrete
16	32,484	0.438	Carbon Steel
17	11,524	0.6065	Carbon Steel
18	2,615	1.032	Carbon Steel
19	3,853	1.4683	Carbon Steel
20	9,243	4.593	Carbon Steel
21	4,223	0.1875 54	Carbon Steel Concrete
22	35,405	0.121	Galvanized Steel
23	37,699	0.0018	Stainless Steel
24	17,626	12	Concrete
25	3	0.719	Aluminum

Table 4a. Safety Injection Flow Used for Region I Calculation for Beaver Valley Unit 1

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm)	Low Head Safety Injection (LHSI) Flow (gpm)
14.7	287.9	2110.7
24.7	287.1	2028.6
34.7	286.6	1944.7
64.7	284.7	1666.1
114.7	281.6	1080.1
164.7	278.3	61.5
164.71	278.32	0
214.7	274.9	0
414.7	261	0
614.7	246	0
814.7	230.3	0
1014.7	214.4	0
1214.7	198.2	0
1414.7	179.6	0
1614.7	159.2	0
1814.7	138.1	0
2014.7	111.6	0
2014.71	0	0

Note: The data included in this table only shows the safety injection flows into the intact loops. Safety injection flow into the broken loop is not provided.

Table 4b. Safety Injection Flow Used for Region I Calculation for Beaver Valley Unit 2

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm)	Low Head Safety Injection (LHSI) Flow (gpm)
14.7	287.2	2453.5
34.7	286.1	2232.3
54.7	285.1	1985
74.7	284.1	1699
94.7	283.1	1349.5
114.7	282.1	735.7
114.71	282.1	0
134.7	281	0
214.7	275.8	0
414.7	262.2	0
614.7	248	0
814.7	232.9	0
1014.7	217.2	0
1214.7	200	0
1414.7	181.8	0
1614.7	161.4	0
1814.7	137.6	0
2014.7	107	0
2014.71	0	0
Note: The data included in this table only shows the safety injection flows into the intact loops. Safety injection flow into the broken loop is not provided.		

Table 5a. Safety Injection Flow Used for Region II Calculation for Beaver Valley Unit 1

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm)	Low Head Safety Injection (LHSI) Flow (gpm)
14.7	288.2	2110.7
24.7	286.6	1953.2
34.7	285.2	1786.6
64.7	281.3	1247.1
104.7	276	294.8
104.71	276.02	0
114.7	274.5	0
164.7	264.7	0
214.7	254.9	0
414.7	215.1	0
614.7	174.4	0
614.71	0	0
Note: The data included in this table only shows the safety injection flows into the intact loops. Safety injection flow into the broken loop is not provided.		

Table 5b. Safety Injection Flow Used for Region II Calculation for Beaver Valley Unit 2

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm)	Low Head Safety Injection (LHSI) Flow (gpm)
14.7	287.2	2453.5
24.7	286.5	2293.5
34.7	285.6	2126.6
64.7	283.3	1559.4
104.7	280.1	469.2
109.7	279.7	253.4
109.71	279.7	0
114.7	279.2	0
164.7	272	0
214.7	264.6	0
414.7	235.2	0
614.7	204.4	0
814.7	172.2	0
1014.7	137.8	0
1214.7	100.8	0
1414.7	59.2	0
1414.71	0	0
Note: The data included in this table only shows the safety injection flows into the intact loops. Safety injection flow into the broken loop is not provided.		

Table 6a. Steam Generator Main Steam Safety Valve Parameters for Beaver Valley Unit 1

Stage	Set Pressure (psig)	Uncertainty (%)	Accumulation (%)
1	1075	±3	±3
2	1085	±3	±3
3	1095	±3	±3
4	1110	±3	±3
5	1125	±3	±3

Table 6b. Steam Generator Main Steam Safety Valve Parameters for Beaver Valley Unit 2

Stage	Set Pressure (psig)	Uncertainty (%)	Accumulation (%)
1	1075	±3	±3
2	1085	±3	±3
3	1095	±3	±3
4	1110	±3	±3
5	1125	±3	±3

Table 7a. Beaver Valley Unit 1 Analysis Results with the FSLOCA EM

Outcome	Region I Value	Region II Value (OPA)	Region II Value (LOOP)
95/95 PCT	1235+1=1236°F	1970+31=2001°F	1975+31=2006°F
95/95 MLO	10.3%	12.6%	12.5%
95/95 CWO	0.0%	0.92%	0.78%

Table 7b. Beaver Valley Unit 2 Analysis Results with the FSLOCA EM

Outcome	Region I Value	Region II Value (OPA)	Region II Value (LOOP)
95/95 PCT	1272+5 = 1277°F	1825+31 = 1856°F	1823+31 = 1854°F
95/95 MLO	10.3%	11.6%	11.3%
95/95 CWO	0.0%	0.34%	0.33%

Table 8a. Beaver Valley Unit 1 Sequence of Events for the Region I Analysis PCT Case

Event	Time after Break (sec)
Start of Transient	0.0
Reactor Trip Signal	16.5
Safety Injection Signal	29.5
Safety Injection Begins	61.5
Loop Seal Clearing Occurs	620
Top of Core Uncovered	1450
Accumulator Injection Begins	2740
PCT Occurs	2753
Top of Core Recovered	~5000

Table 8b. Beaver Valley Unit 2 Sequence of Events for the Region I Analysis PCT Case

Event	Time after Break (sec)
Start of Transient	0.0
Reactor Trip Signal	12.7
Safety Injection Signal	24.1
Safety Injection Begins	56.1
Loop Seal Clearing Occurs	635
Top of Core Uncovered	1650
Accumulator Injection Begins	3260
PCT Occurs	3276
Top of Core Recovered	~5000

Table 9a. Beaver Valley Unit 1 Sequence of Events for the Region II Analysis PCT Case (LOOP)

Event	Time after Break (sec)
Start of Transient	0.0
Safety Injection Signal	3.8
Fuel Rod Burst Occurs	3.8
Accumulator Injection Begins	8.5
End of Blowdown	18.5
Safety Injection Begins	35.8
Accumulator Empty	41
PCT Occurs	207
All Rods Quenched	485

Table 9b. Beaver Valley Unit 2 Sequence of Events for the Region II Analysis PCT Case (OPA)

Event	Time after Break (sec)
Start of Transient	0.0
Safety Injection Signal	3.3
Fuel Rod Burst Occurs	7.5
Accumulator Injection Begins	9.0
End of Blowdown	16.5
Safety Injection Begins	25.3
Accumulator Empty	48.0
PCT Occurs	104
All Rods Quenched	327

Table 10a. Beaver Valley Unit 1 Sampled Value of Decay Heat Uncertainty Multiplier, DECAY_HT, for Region I and Region II Analysis Cases

Region	Case	DECAY_HT (units of σ)	DECAY_HT (absolute units)(1)
Region I	PCT	+0.8204	4.17%
	MLO	+0.5458	2.80%
	CWO	N/A(2)	N/A(2)
Region II (OPA)	PCT	+1.2826	6.21%
	MLO	+0.5893	2.98%
	CWO	+1.7965	8.73%
Region II (LOOP)	PCT	+1.6585	8.13%
	MLO	+0.3649	1.84%
	CWO	+0.7146	3.52%

Notes:

1. Approximate uncertainty in total decay heat power at 1 second after shutdown as defined by the ANSI/ANS-5.1-1979 decay heat standard for ^{235}U, ^{239}Pu, and ^{238}U assuming infinite operation.
2. No decay heat uncertainty value is provided for the SBLOCA (Region I) CWO case since the analysis result for all runs is 0.0%.

Table 10b. Beaver Valley Unit 2 Sampled Value of Decay Heat Uncertainty Multiplier, DECAY_HT, for Region I and Region II Analysis Cases

Region	Case	DECAY_HT (units of σ)	DECAY_HT (absolute units)(1)
Region I	PCT	+0.6279	3.16%
	MLO	+0.2579	1.32%
	CWO	N/A(2)	N/A(2)
Region II (OPA)	PCT	+0.5475	2.59%
	MLO	+0.4662	2.34%
	CWO	+0.1645	0.79%
Region II (LOOP)	PCT	+1.8191	9.06%
	MLO	+0.3936	1.98%
	CWO	+1.0983	5.58%

Notes:

1. Approximate uncertainty in total decay heat power at 1 second after shutdown as defined by the ANSI/ANS-5.1-1979 decay heat standard for ^{235}U, ^{239}Pu, and ^{238}U assuming infinite operation.
2. No decay heat uncertainty value is provided for the SBLOCA (Region I) CWO case since the analysis result for all runs is 0.0%.

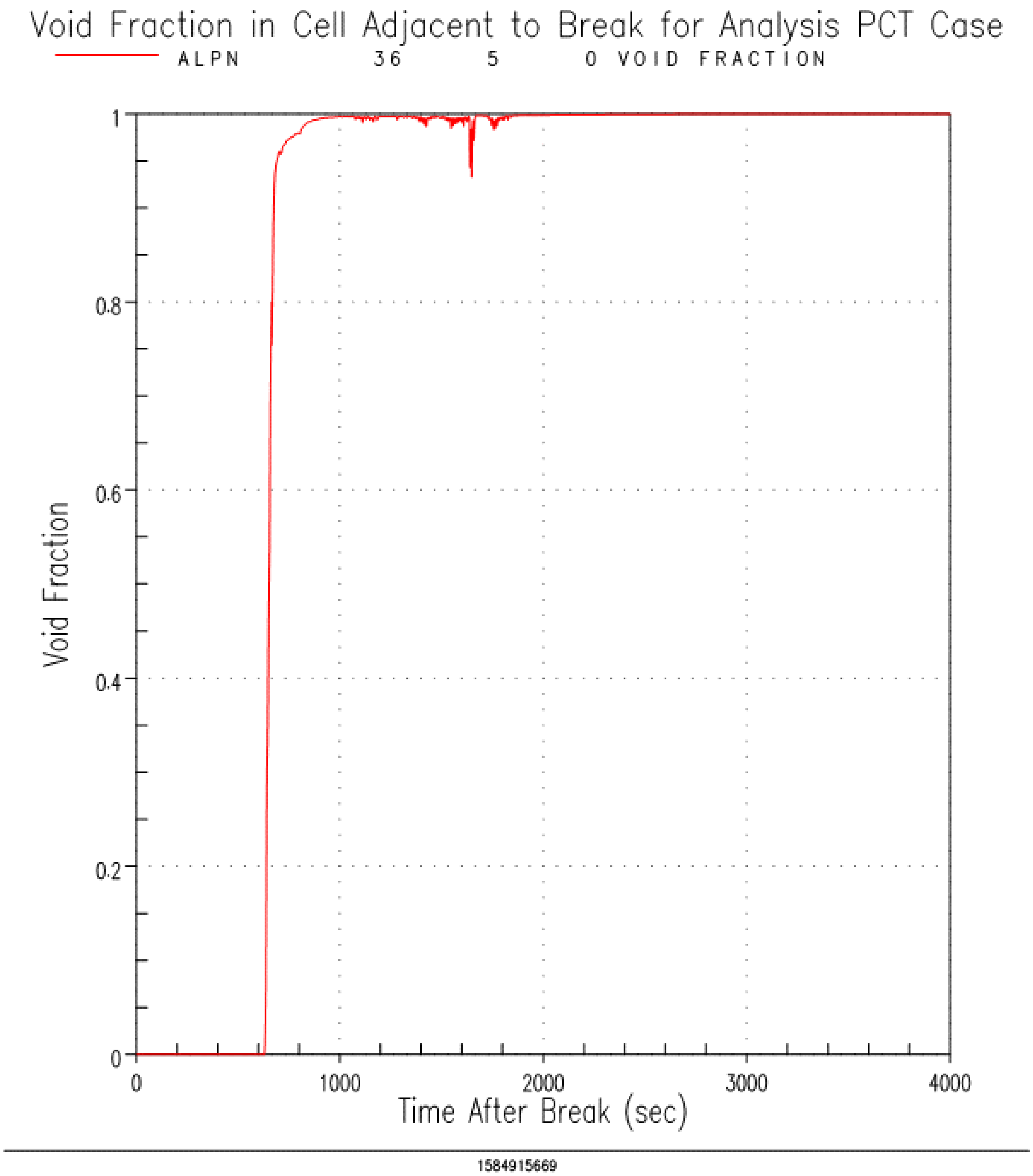

Figure 1a: Beaver Valley Unit 1 Break Flow Void Fraction for the Region I Analysis PCT Case

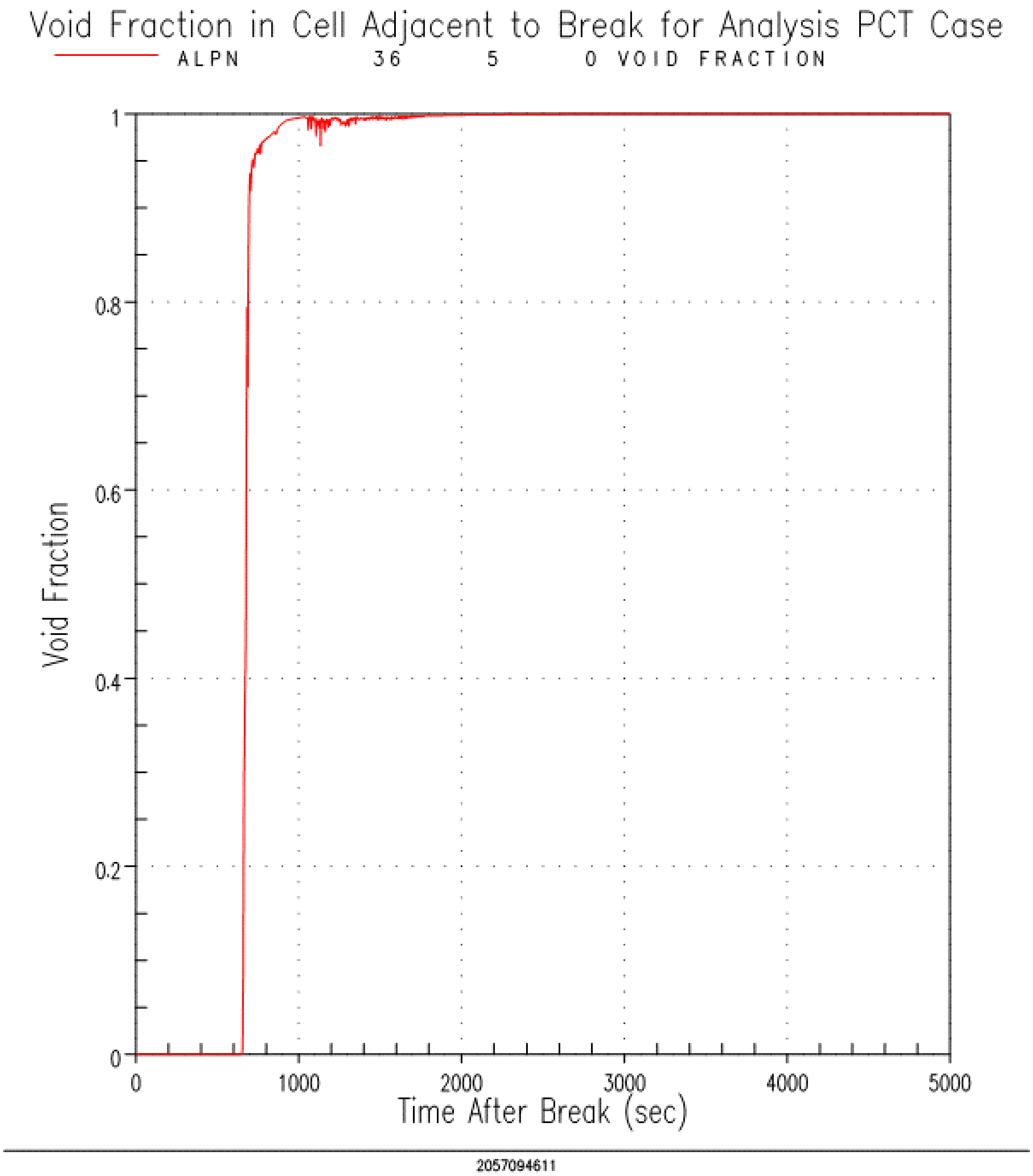

Figure 1b: Beaver Valley Unit 2 Break Flow Void Fraction for the Region I Analysis PCT Case

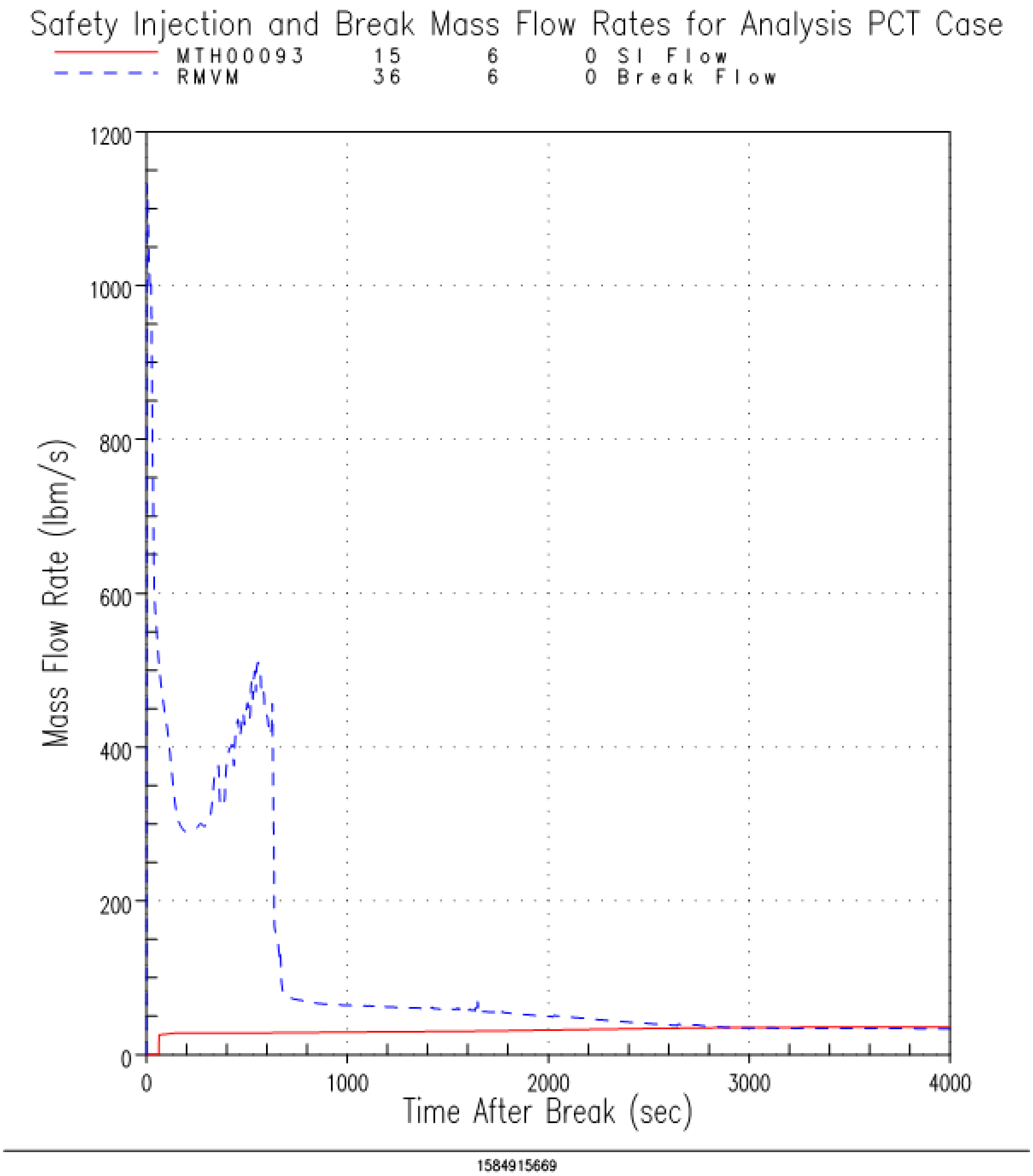

Figure 2a: Beaver Valley Unit 1 Total Safety Injection Flow (not including Accumulator Injection Flow) and Total Break Flow for the Region I Analysis PCT Case

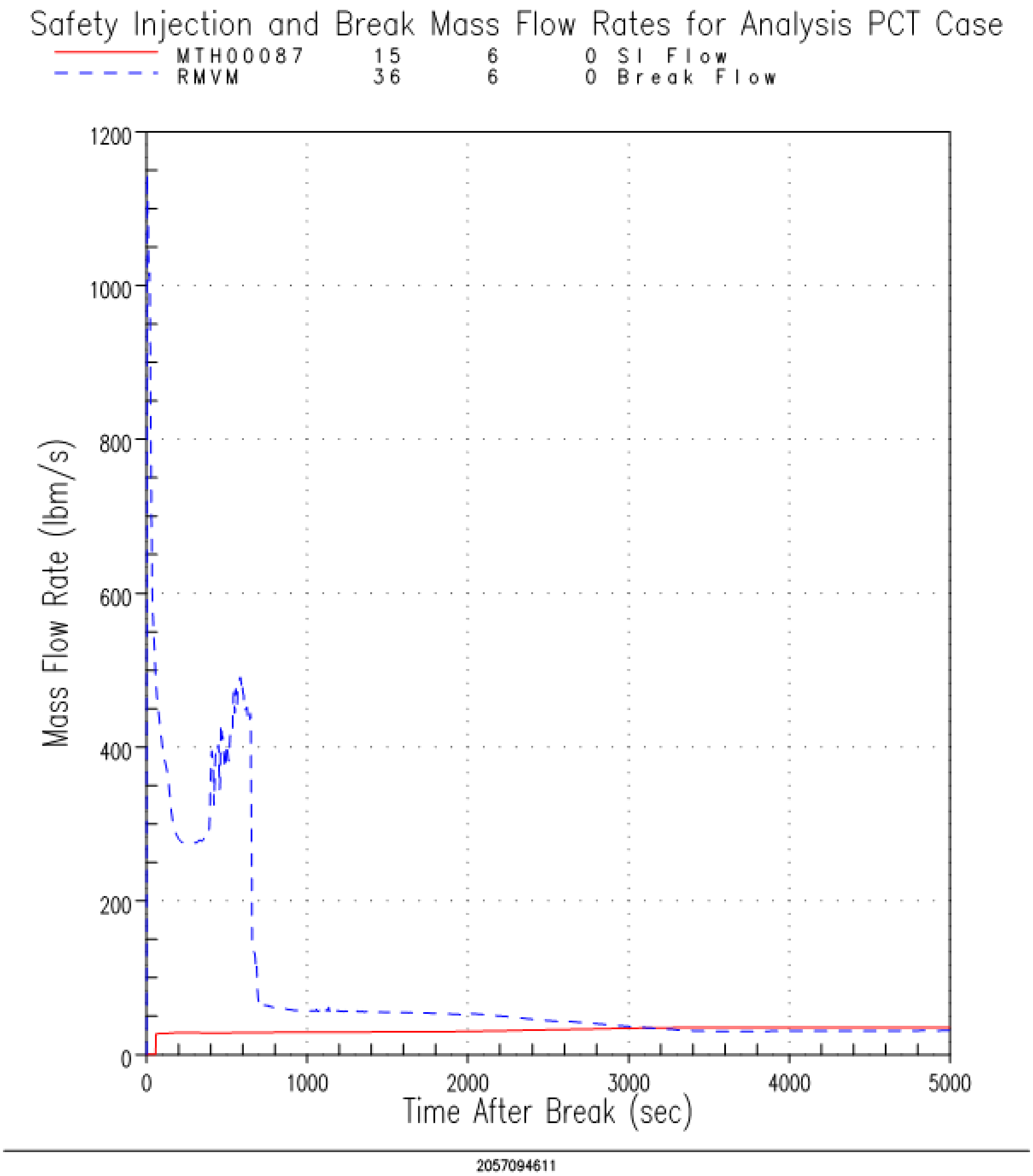

Figure 2b: Beaver Valley Unit 2 Total Safety Injection Flow (not including Accumulator Injection Flow) and Total Break Flow for the Region I Analysis PCT Case

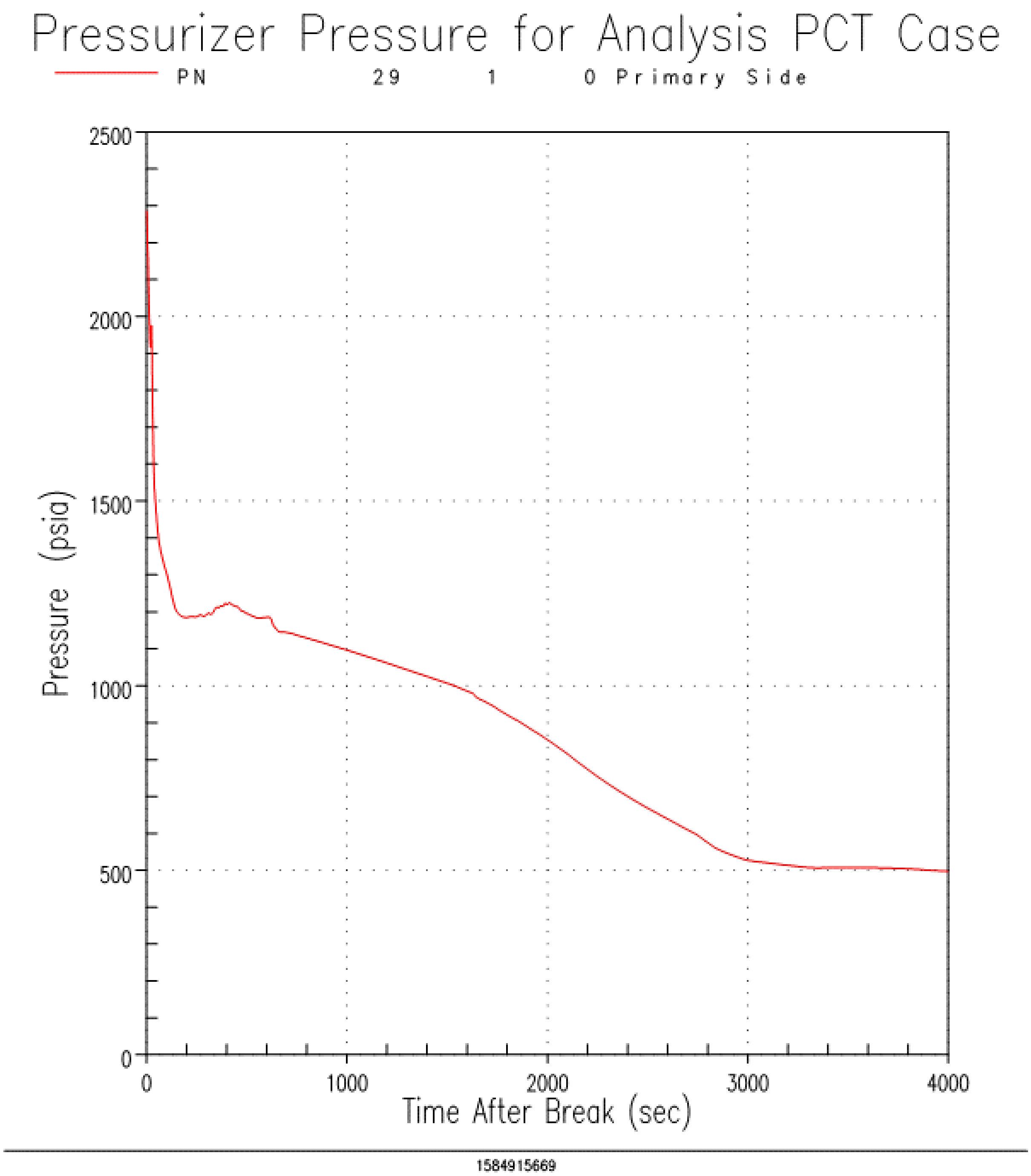

Figure 3a: Beaver Valley Unit 1 RCS Pressure for the Region I Analysis PCT Case

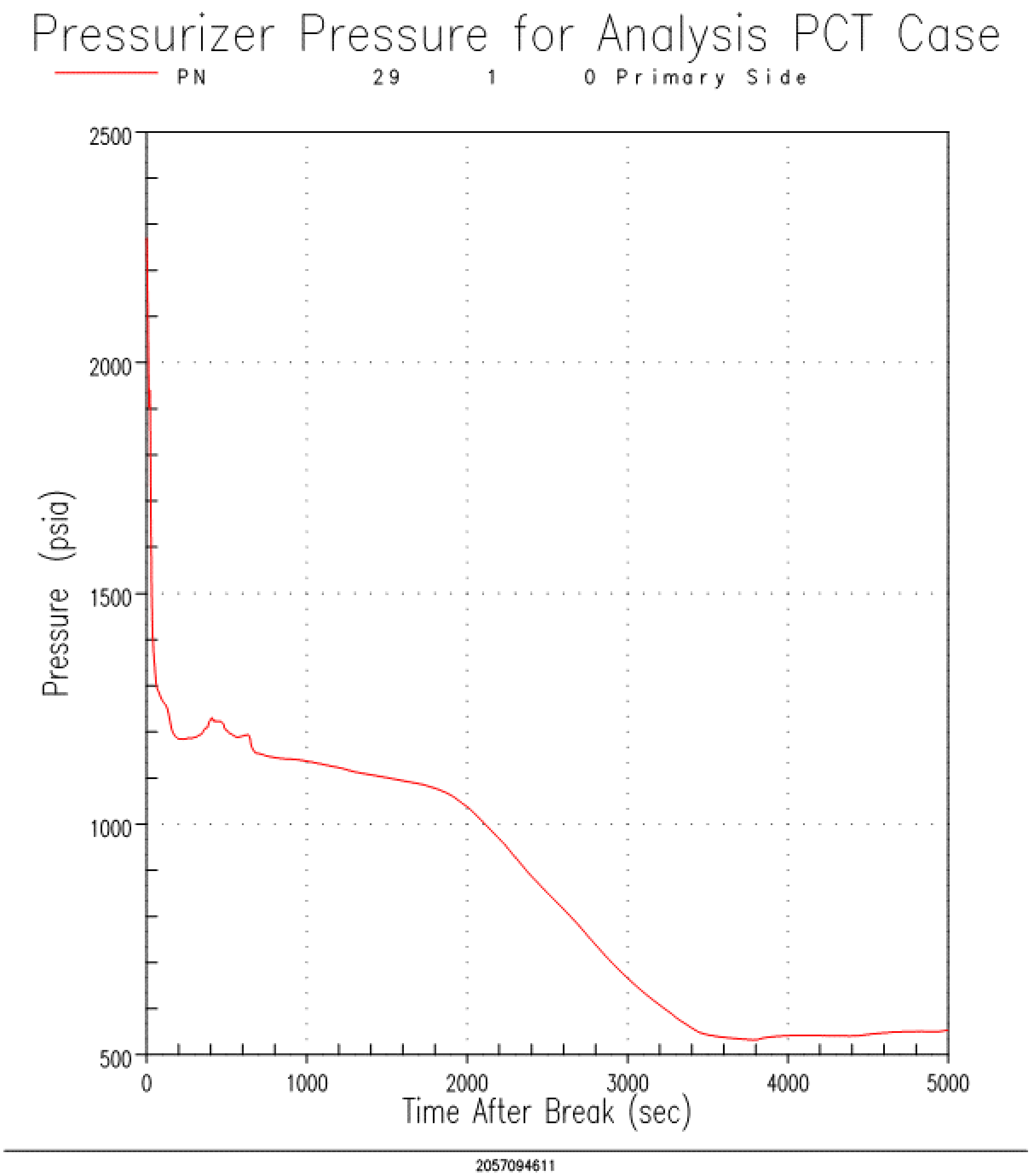

Figure 3b: Beaver Valley Unit 2 RCS Pressure for the Region I Analysis PCT Case

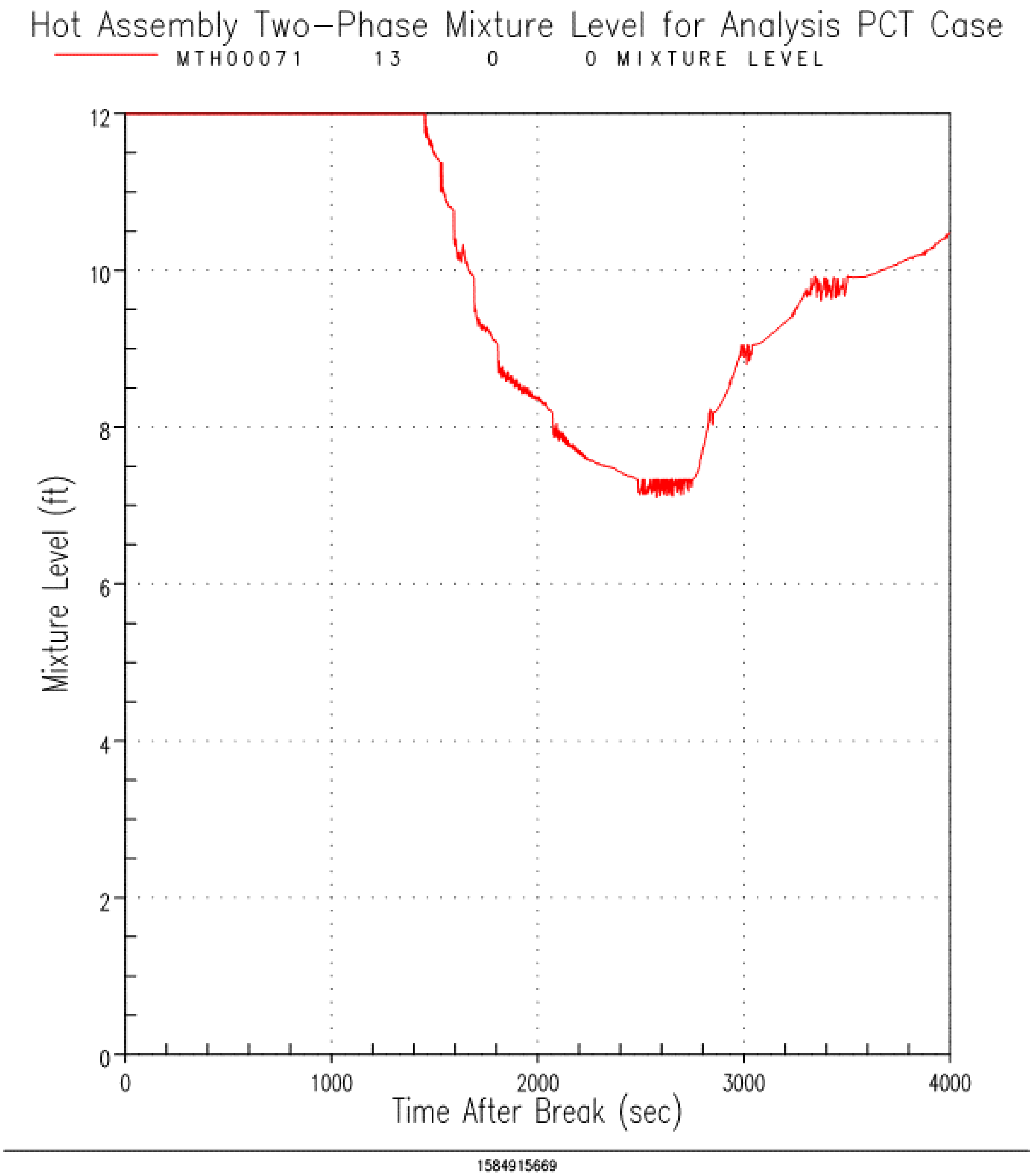

Figure 4a: Beaver Valley Unit 1 Hot Assembly Two-Phase Mixture Level (Relative to Bottom of Active Fuel) for the Region I Analysis PCT Case

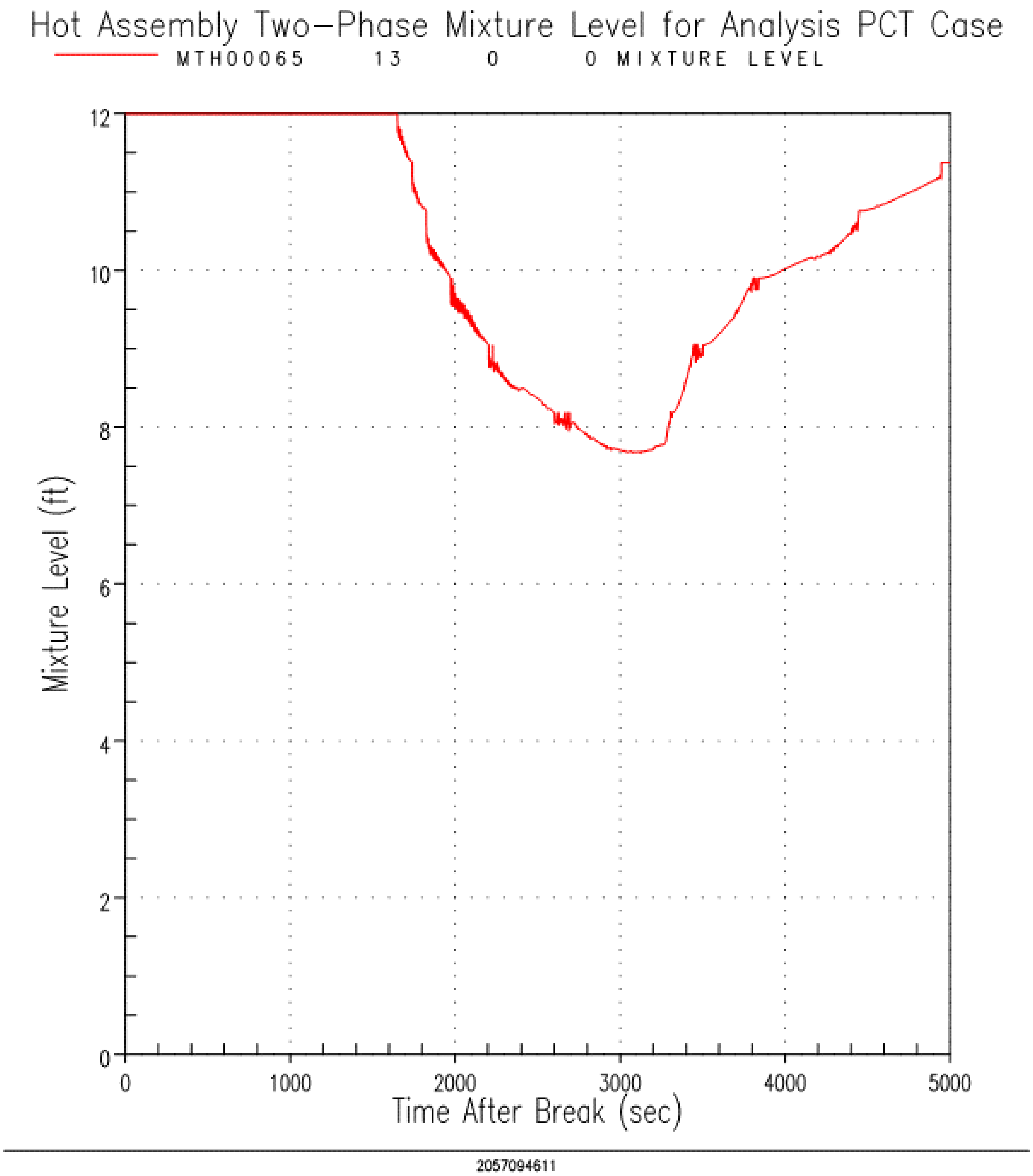

Figure 4b: Beaver Valley Unit 2 Hot Assembly Two-Phase Mixture Level (Relative to Bottom of Active Fuel) for the Region I Analysis PCT Case

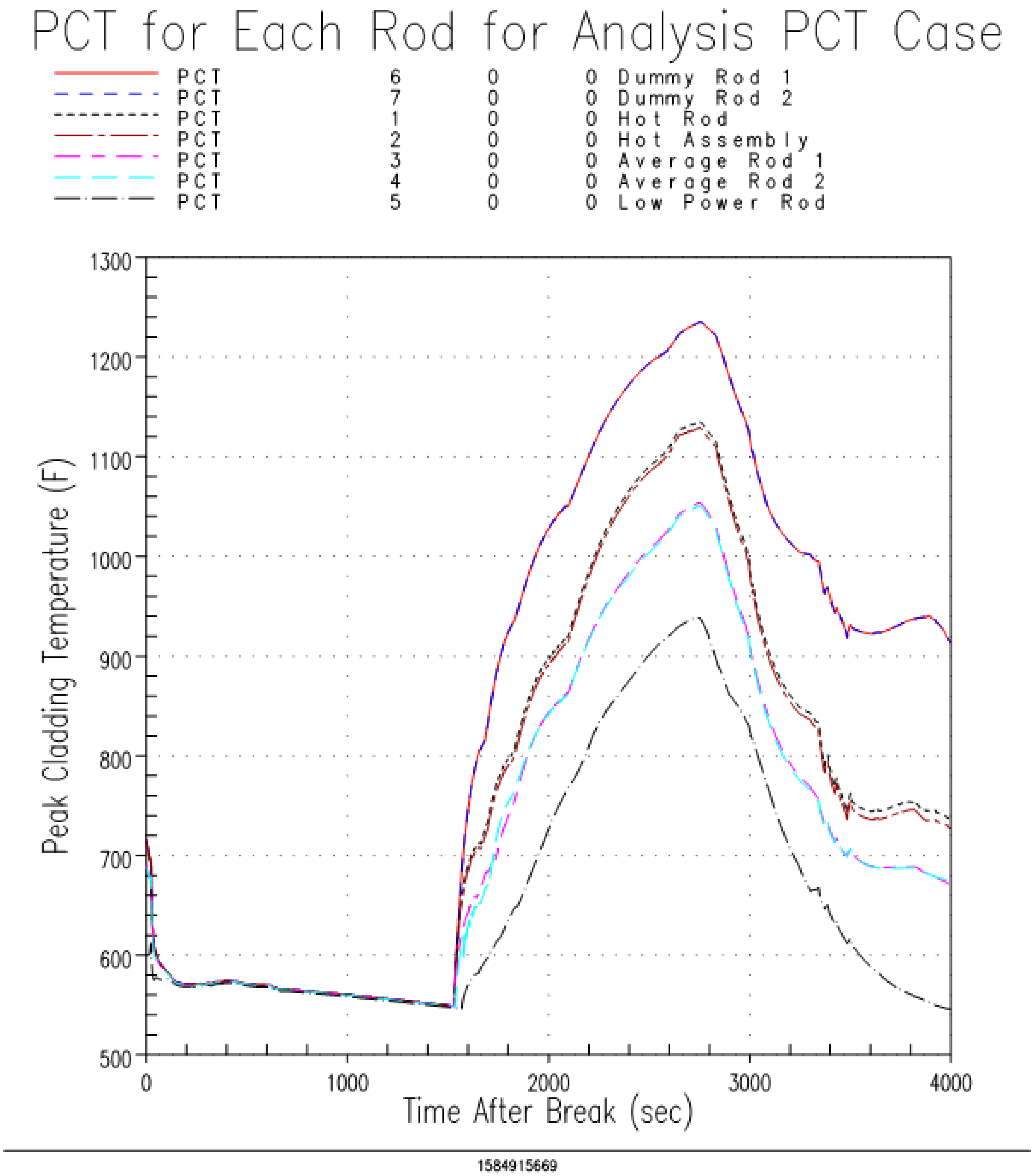

Figure 5a: Beaver Valley Unit 1 Peak Cladding Temperature for all Rods for the Region I Analysis PCT Case

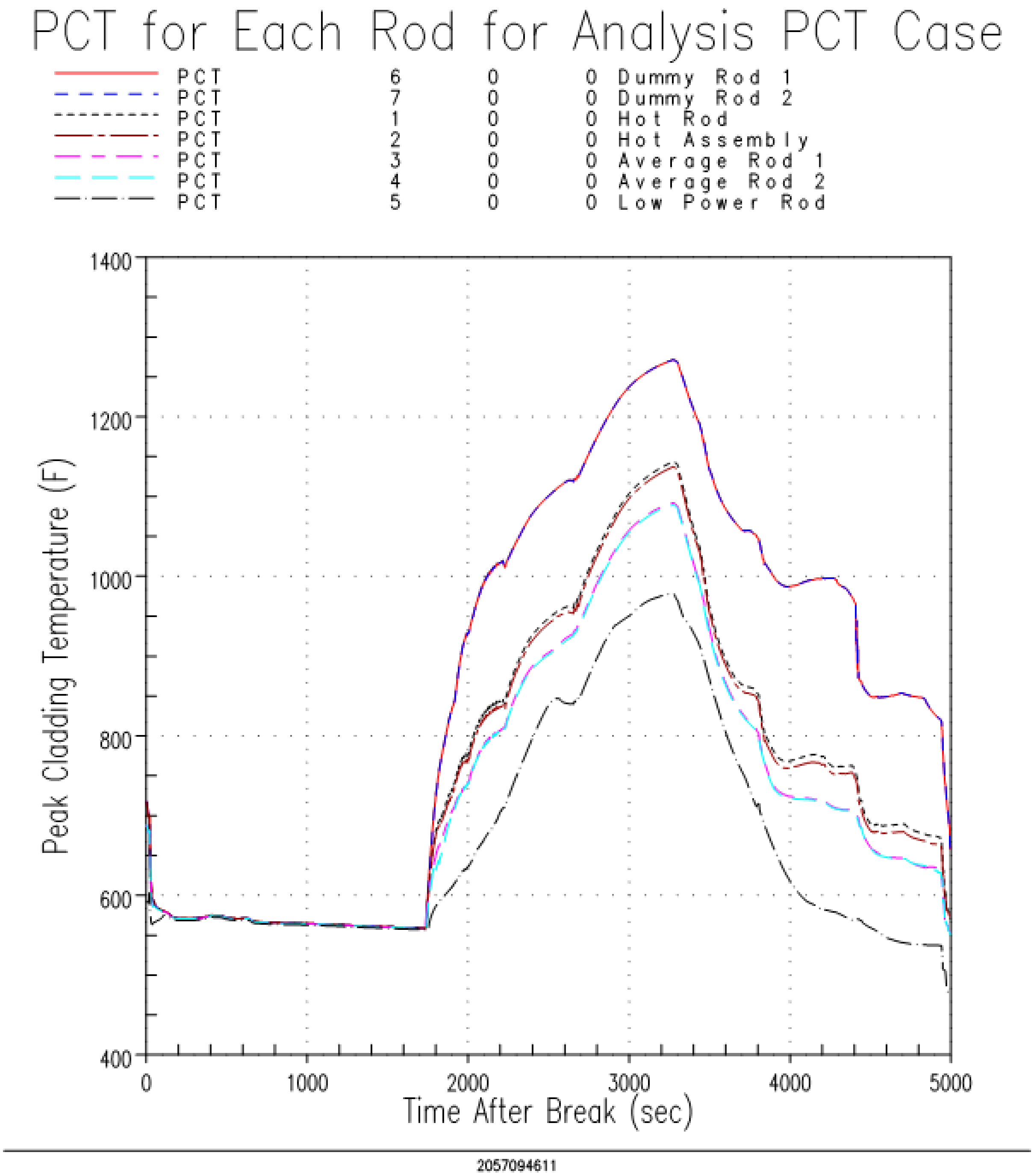

Figure 5b: Beaver Valley Unit 2 Peak Cladding Temperature for all Rods for the Region I Analysis PCT Case

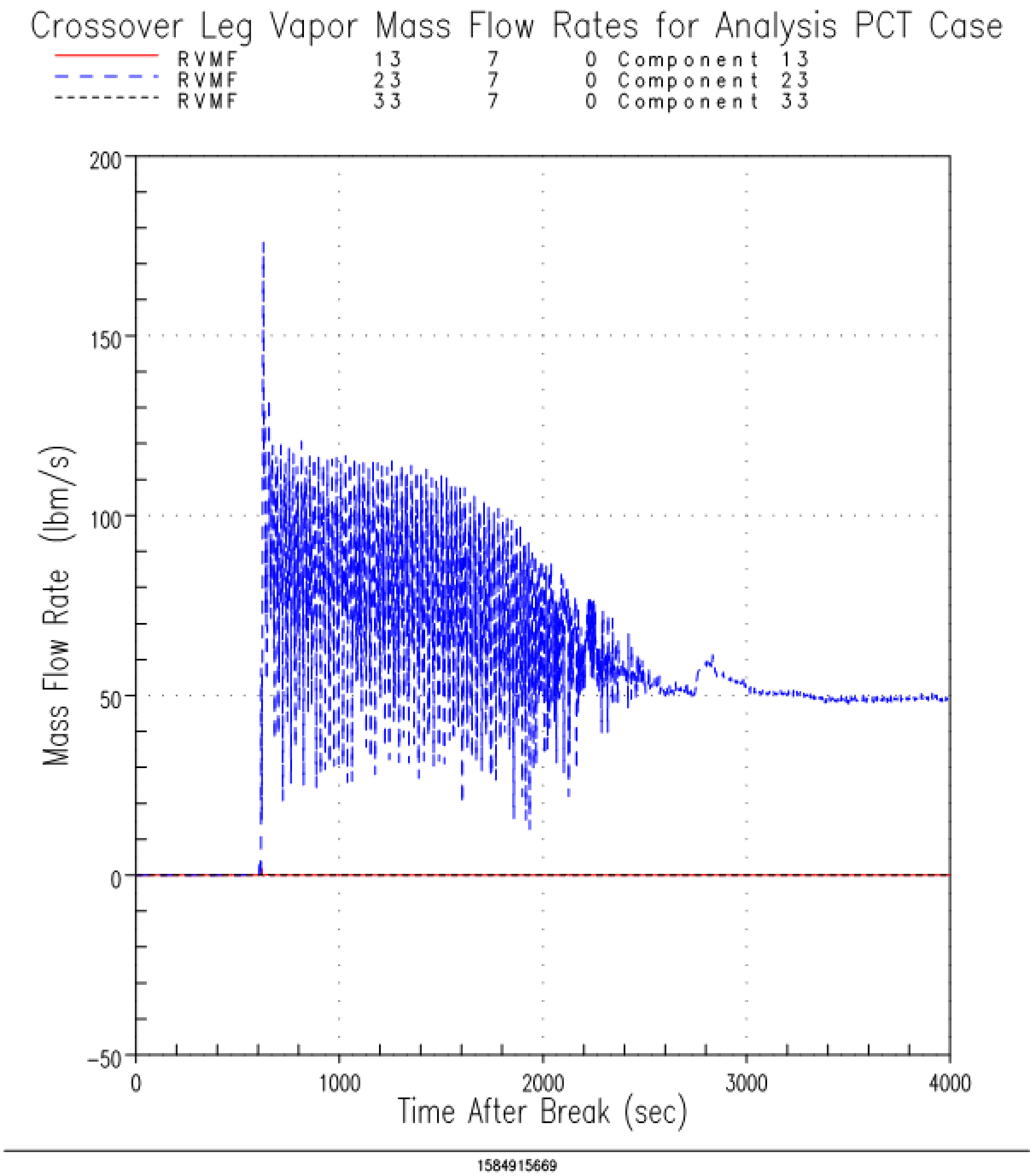

Figure 6a: Beaver Valley Unit 1 Vapor Mass Flow Rate through the Crossover Legs for the Region I Analysis PCT Case

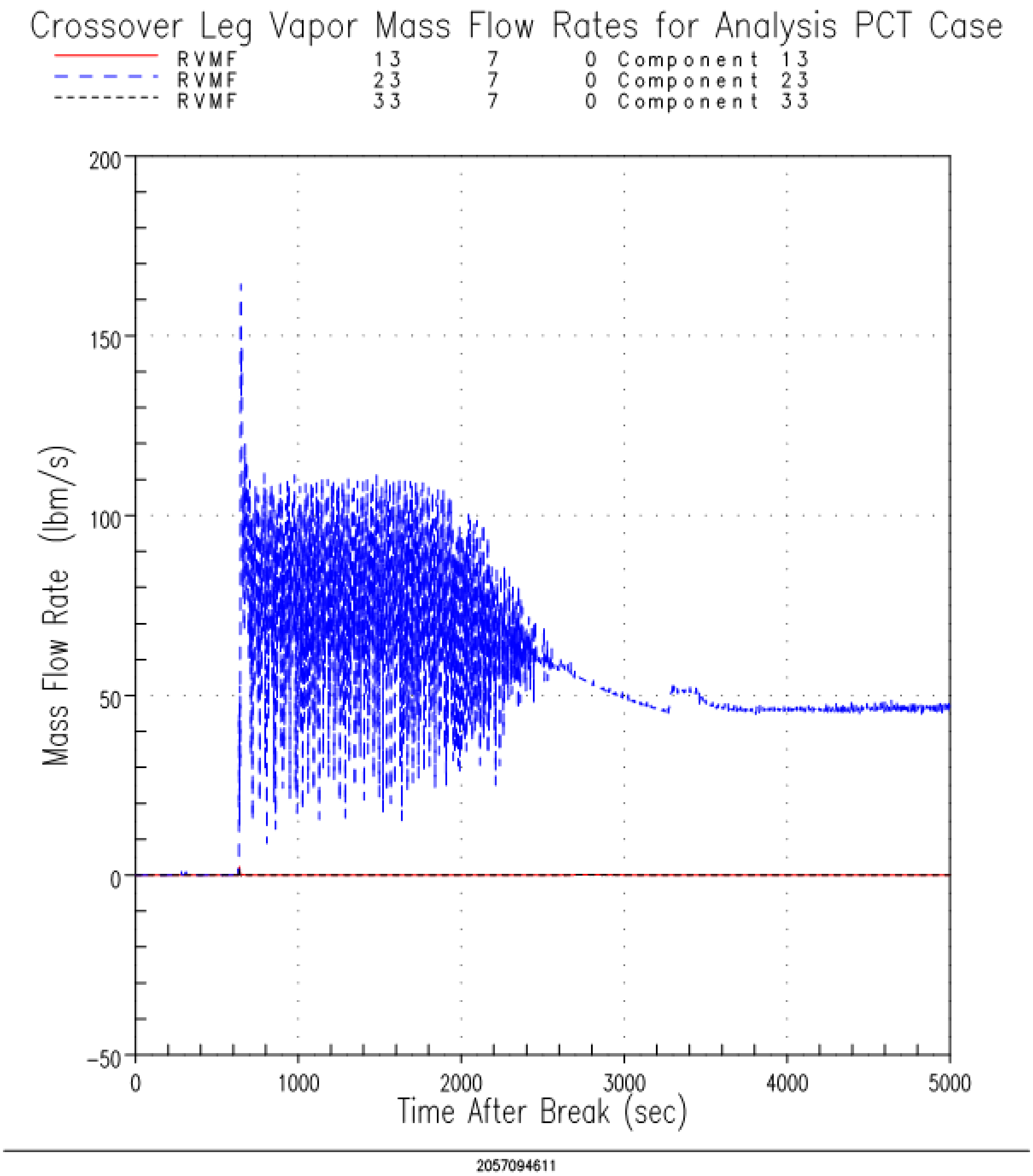

Figure 6b: Beaver Valley Unit 2 Vapor Mass Flow Rate through the Crossover Legs for the Region I Analysis PCT Case

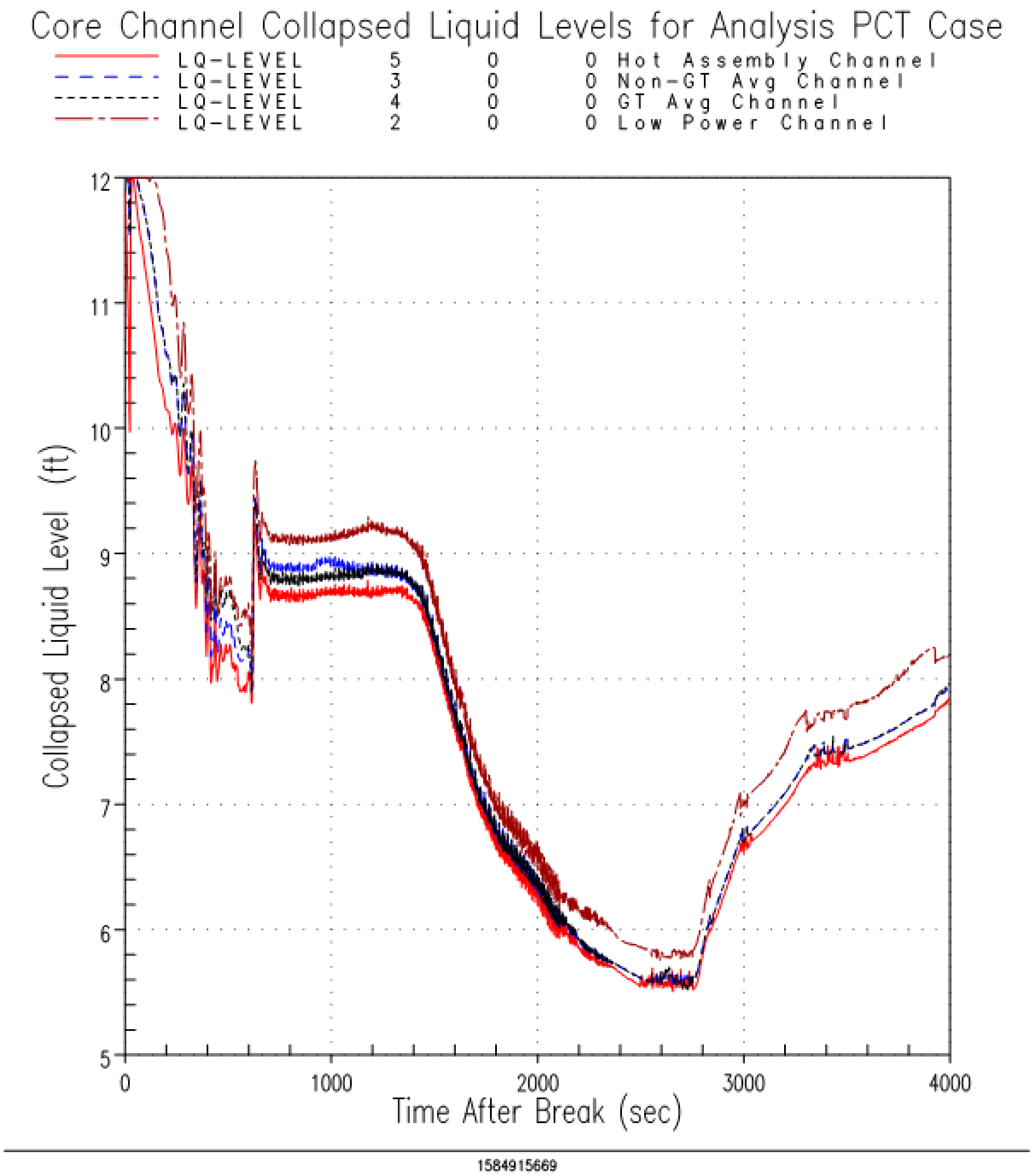

Figure 7a: Beaver Valley Unit 1 Core Collapsed Liquid Levels (Relative to Bottom of Active Fuel) for the Region I Analysis PCT Case

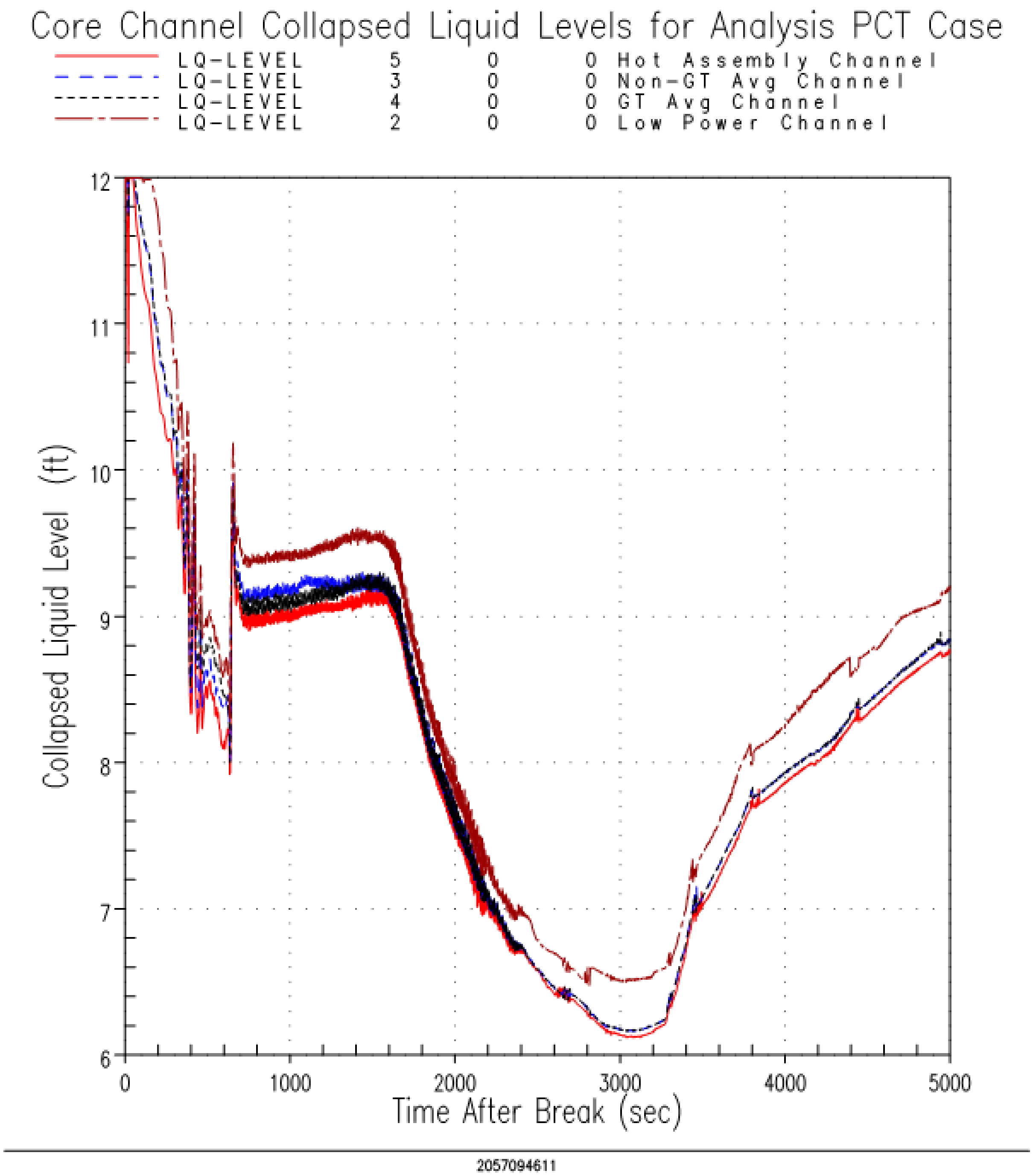

Figure 7b: Beaver Valley Unit 2 Core Collapsed Liquid Levels (Relative to Bottom of Active Fuel) for the Region I Analysis PCT Case

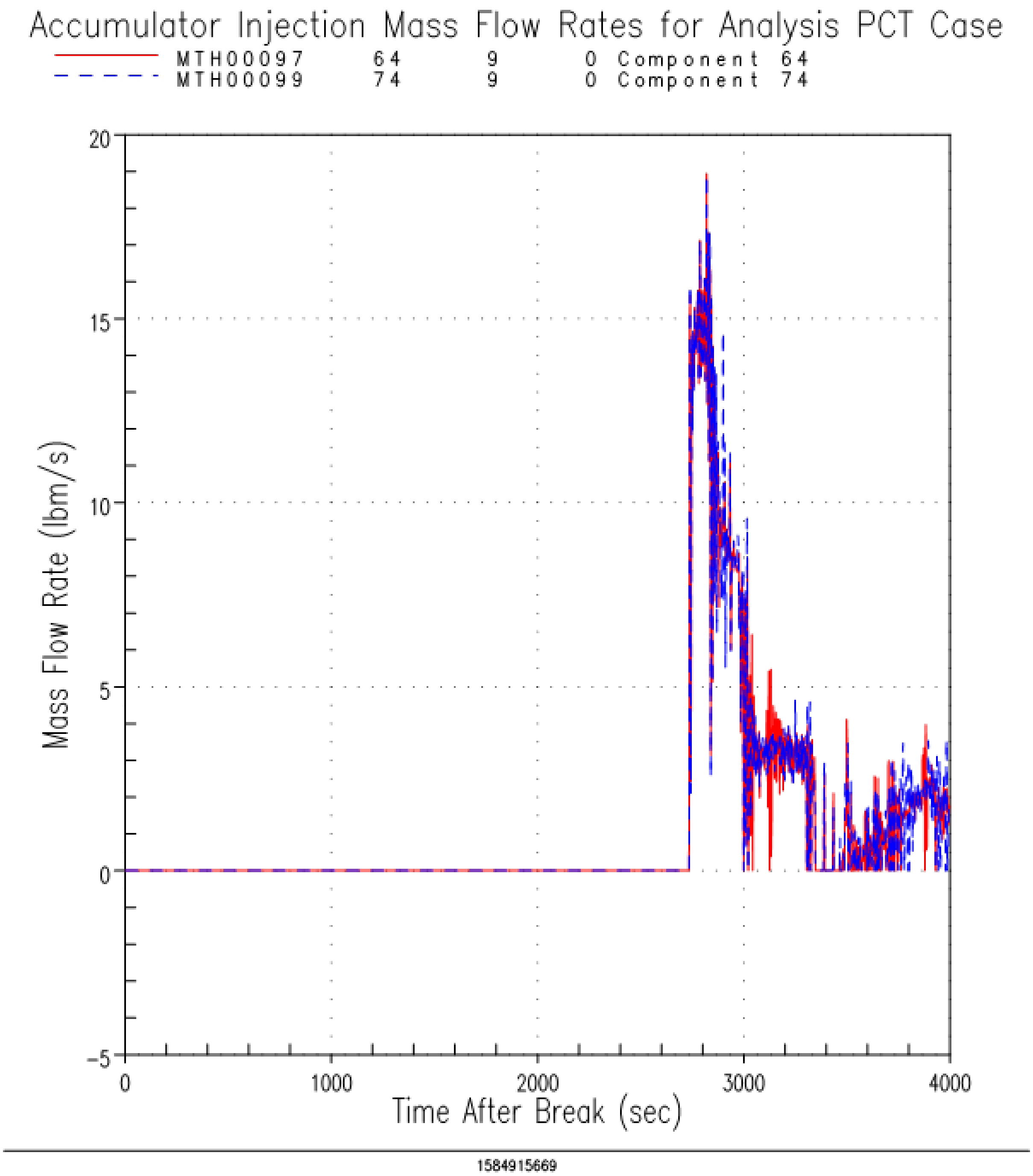

Figure 8a: Beaver Valley Unit 1 Accumulator Injection Flow for the Region I Analysis PCT Case

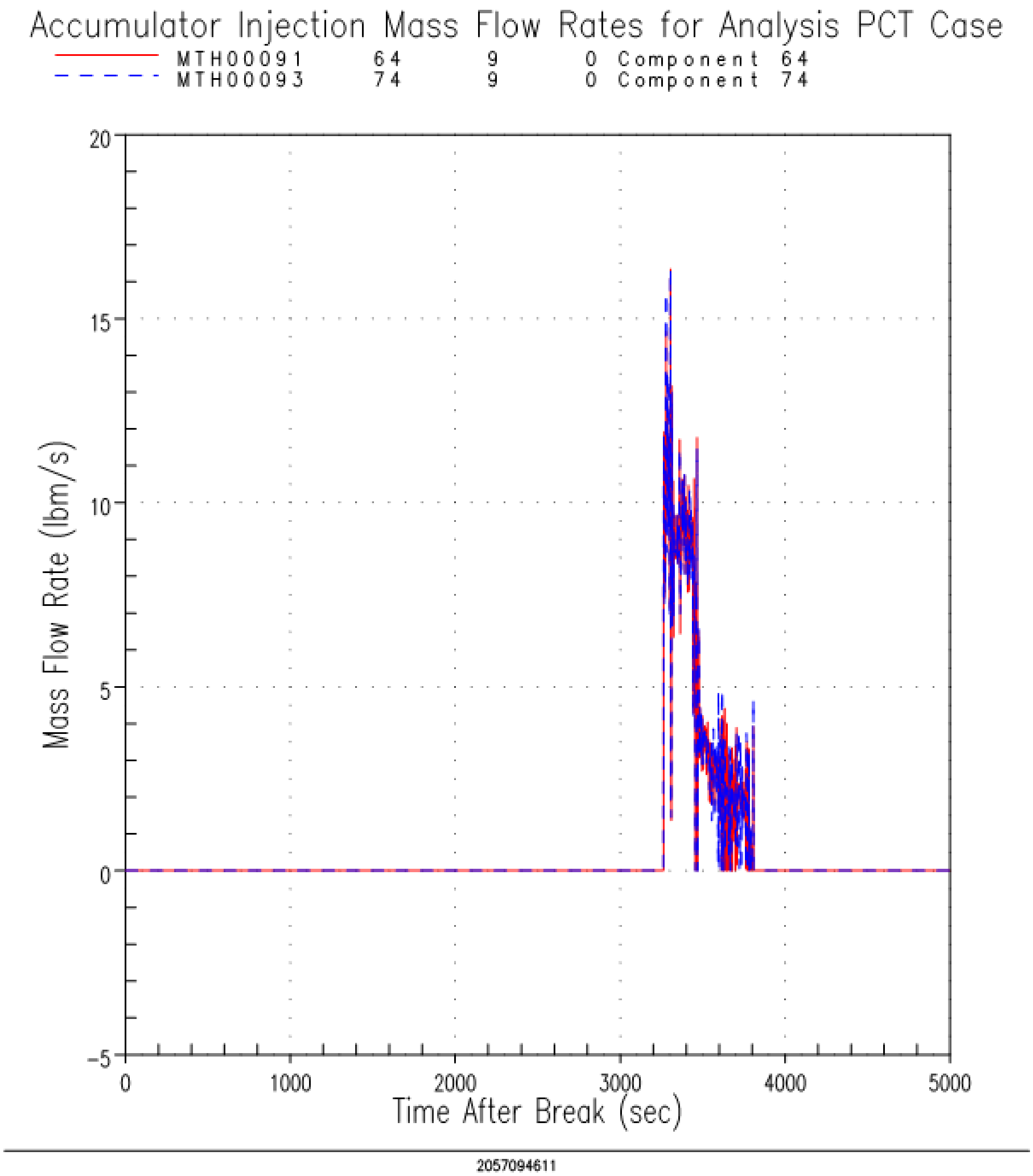

Figure 8b: Beaver Valley Unit 2 Accumulator Injection Flow for the Region I Analysis PCT Case

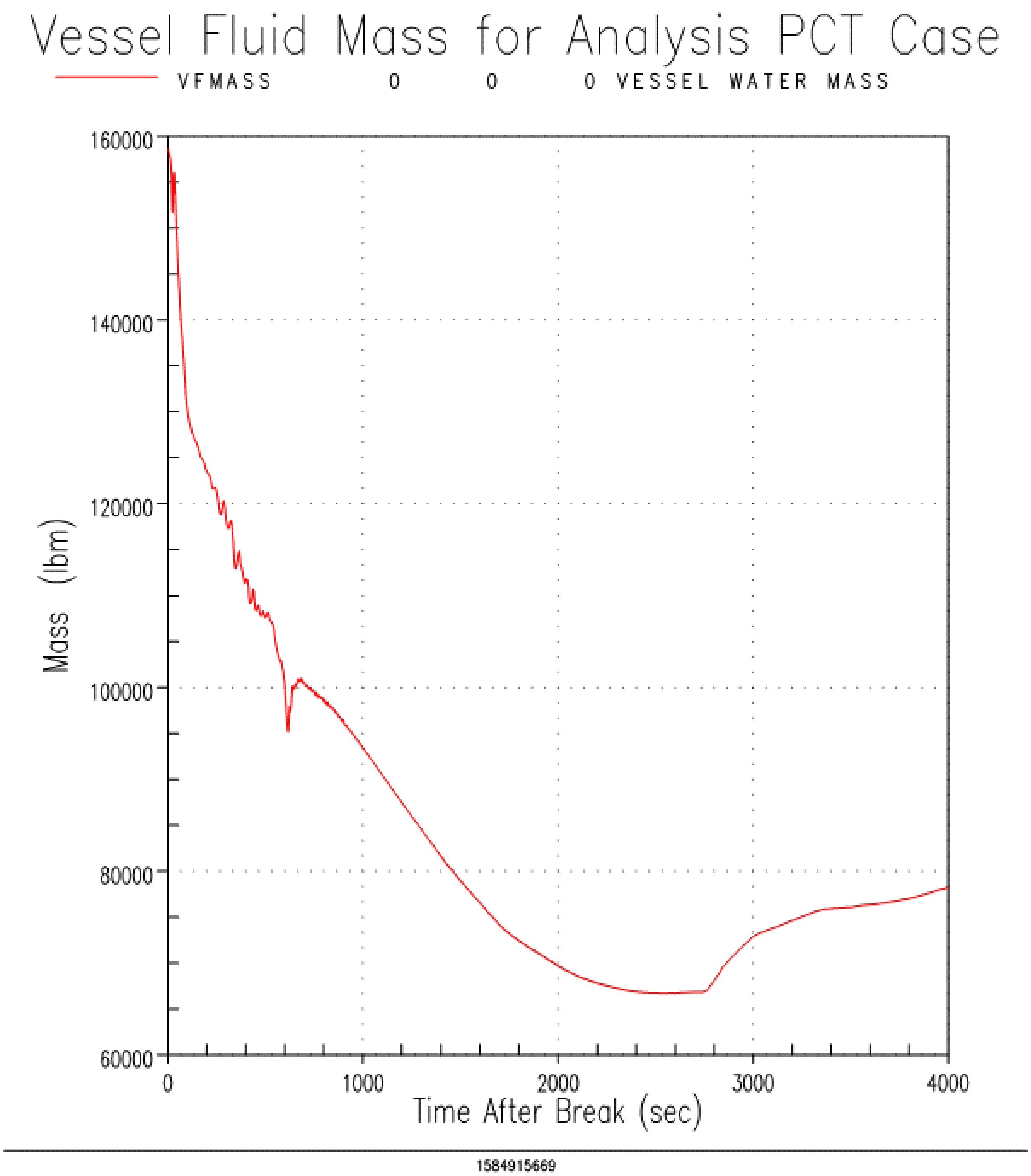

Figure 9a: Beaver Valley Unit 1 Vessel Fluid Mass for the Region I Analysis PCT Case

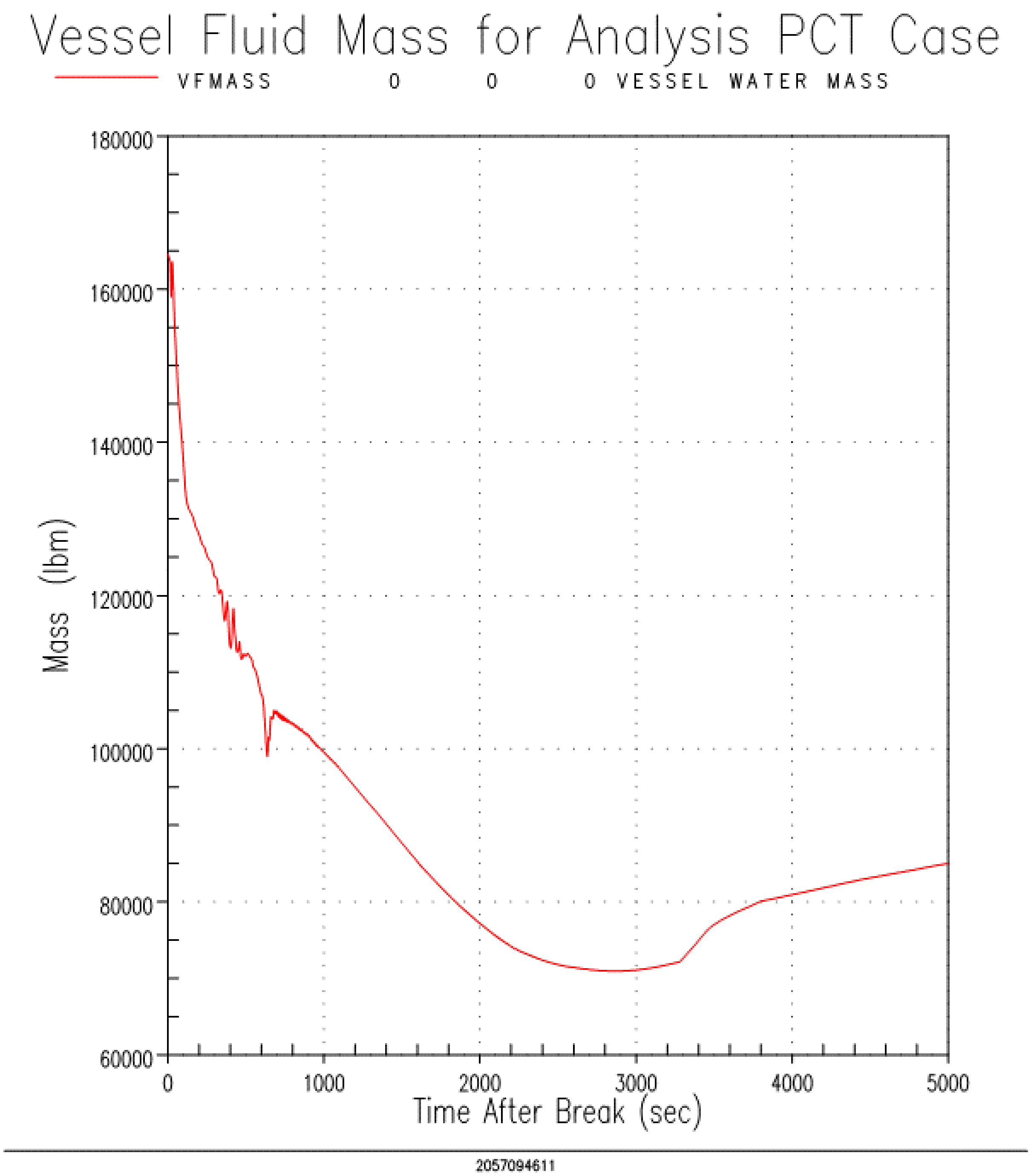

Figure 9b: Beaver Valley Unit 2 Vessel Fluid Mass for the Region I Analysis PCT Case

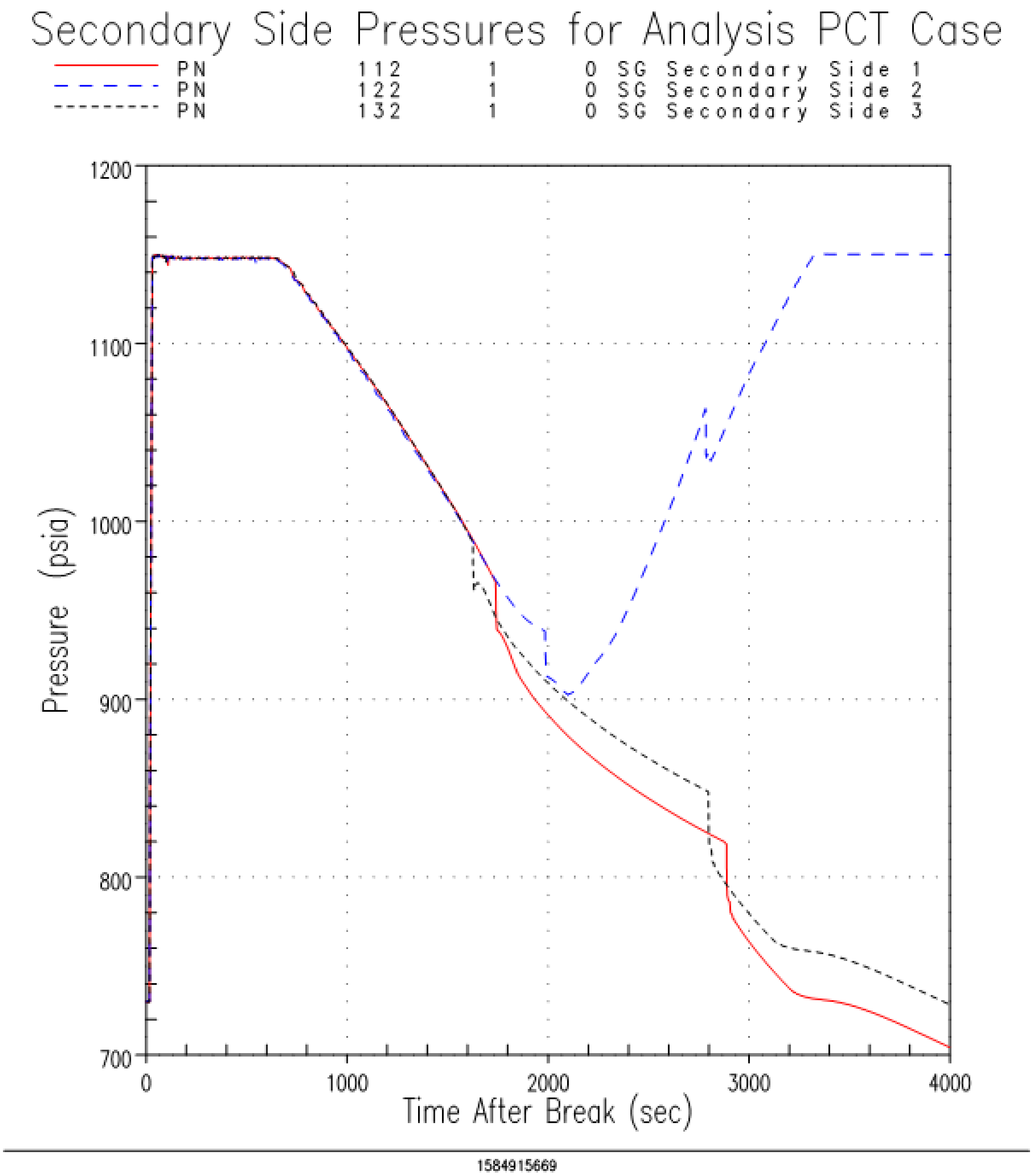

Figure 10a: Beaver Valley Unit 1 Steam Generator Secondary Side Pressure for the Region I Analysis PCT Case

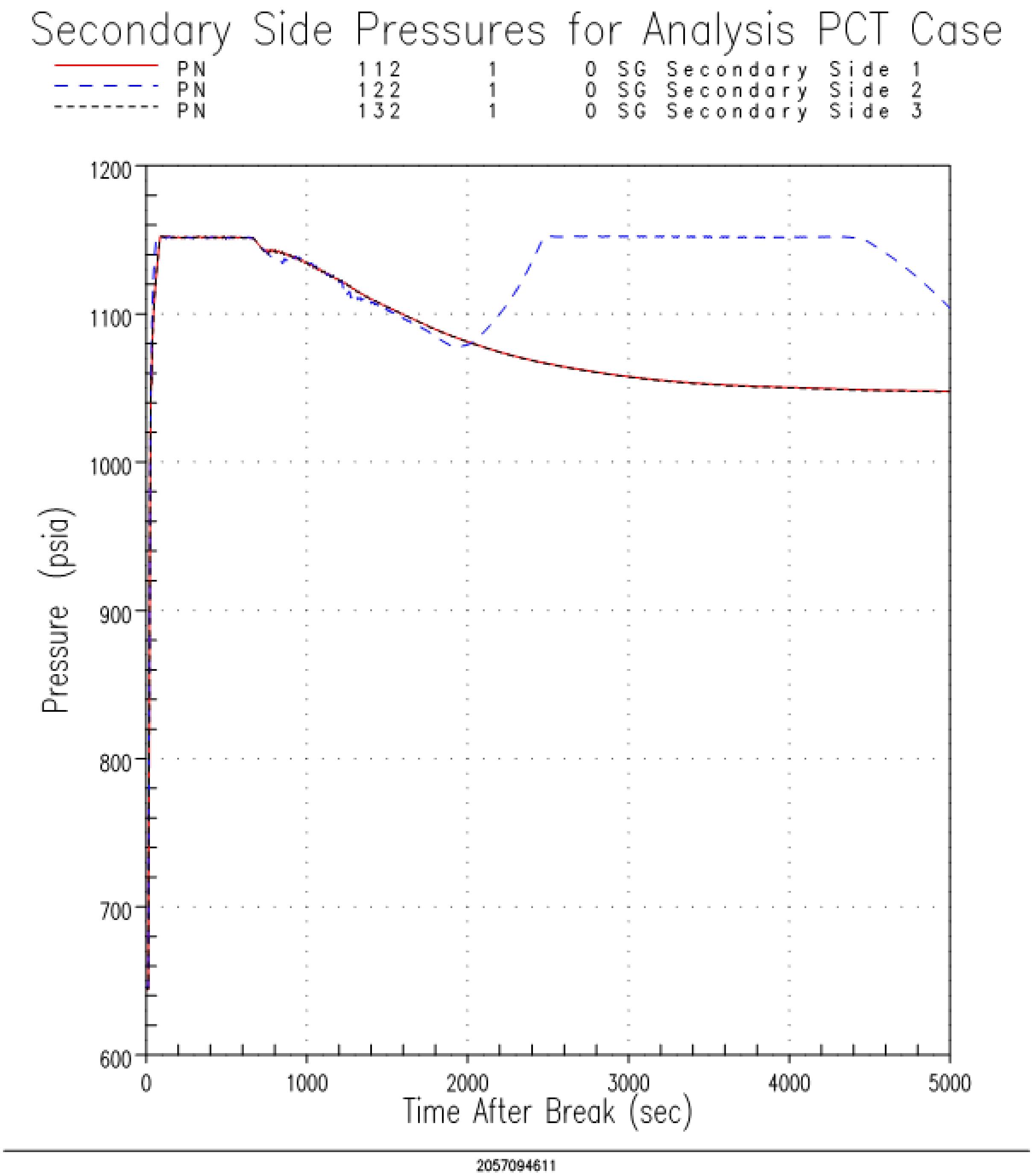

Figure 10b: Beaver Valley Unit 2 Steam Generator Secondary Side Pressure for the Region I Analysis PCT Case

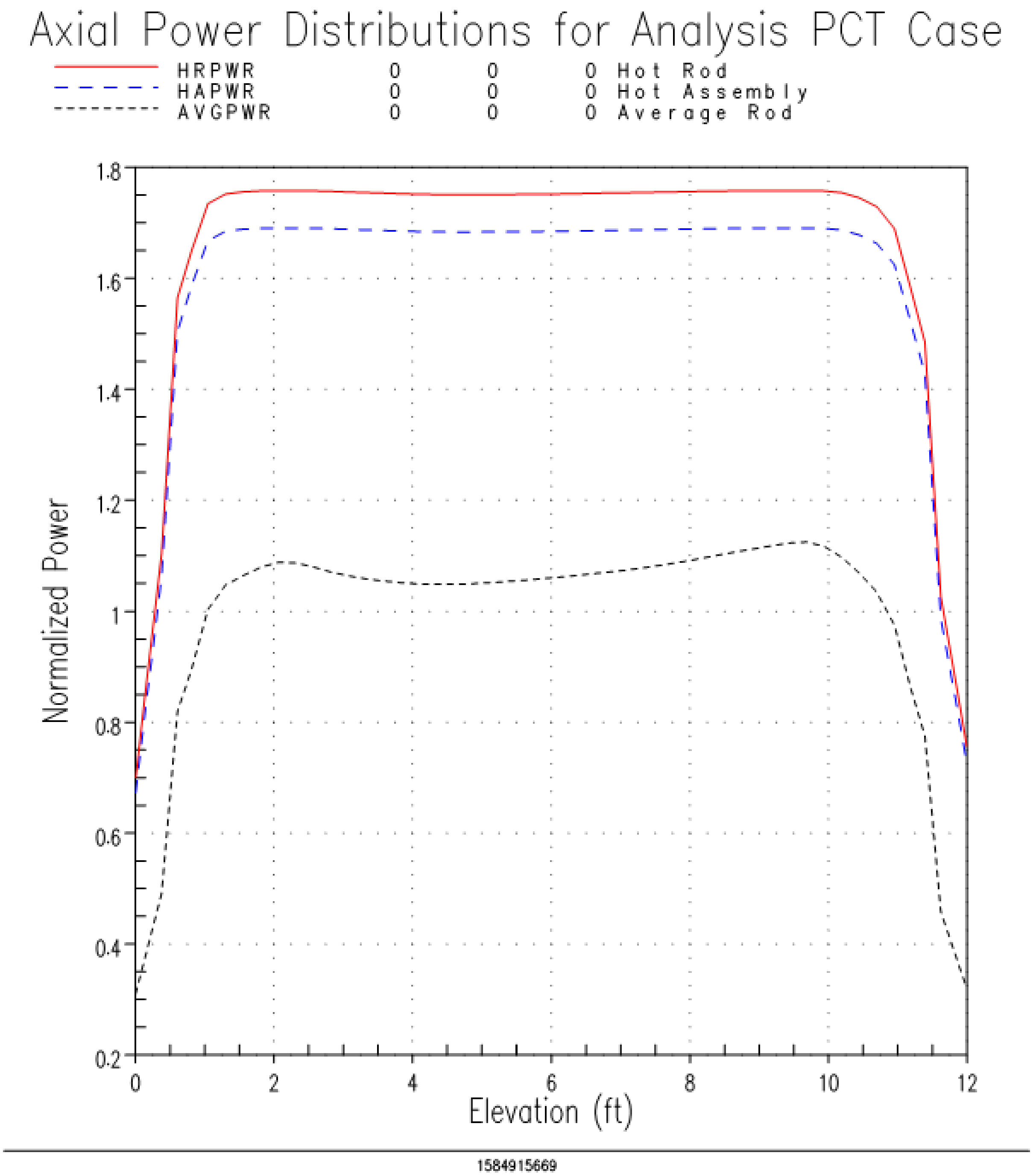

Figure 11a: Beaver Valley Unit 1 Normalized Core Power Shapes for the Region I Analysis PCT Case

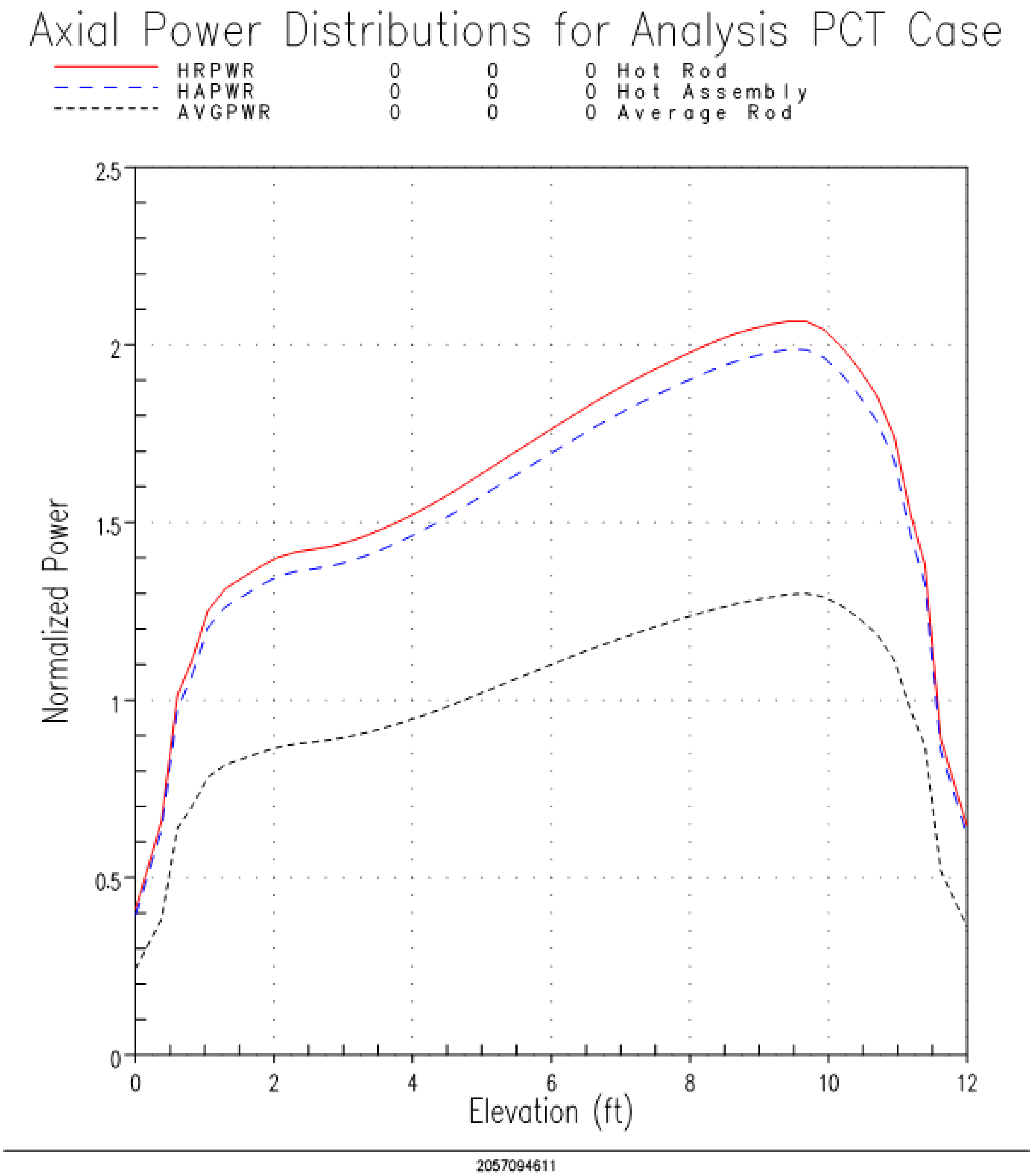

Figure 11b: Beaver Valley Unit 2 Normalized Core Power Shapes for the Region I Analysis PCT Case

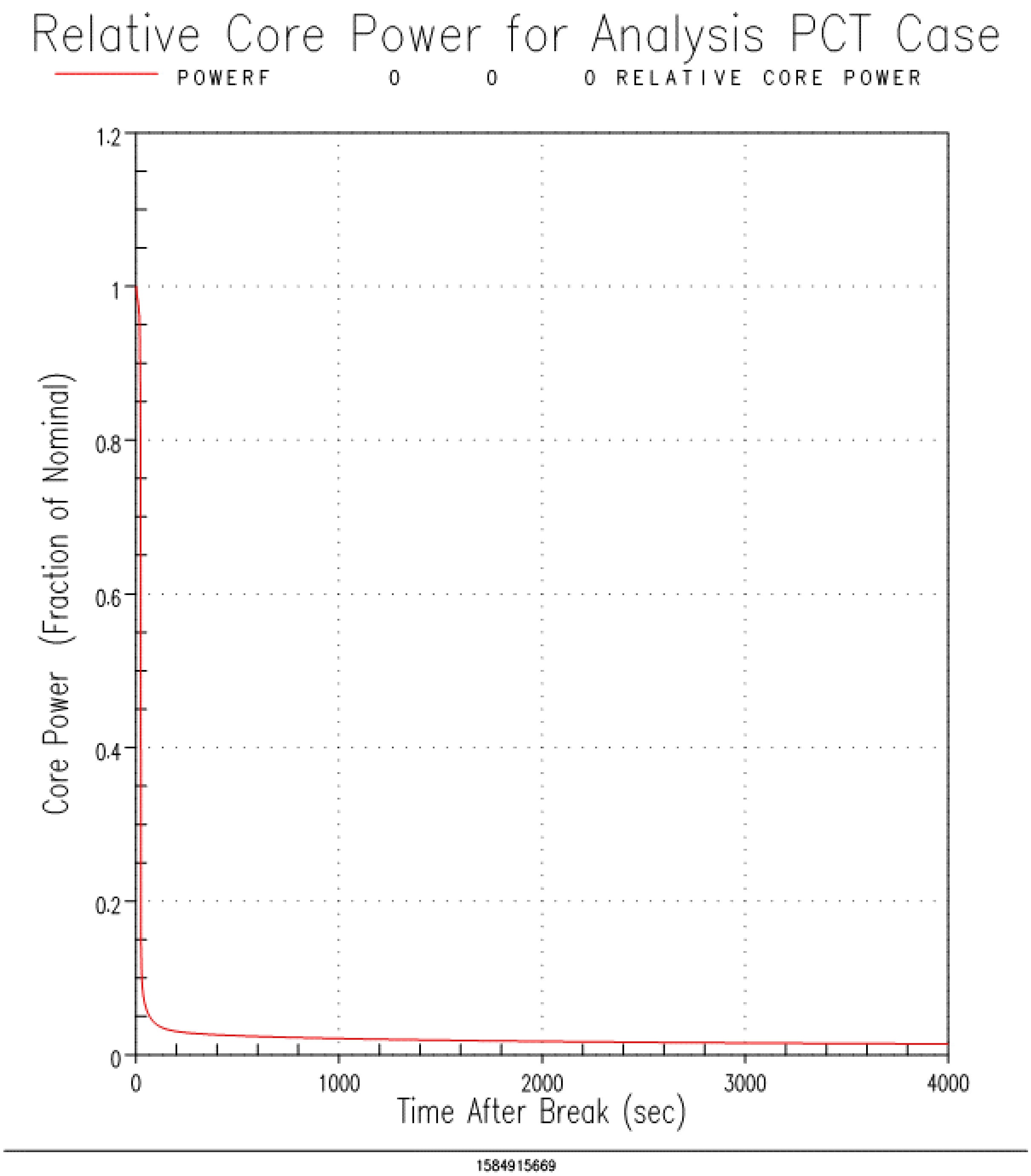

Figure 12a: Beaver Valley Unit 1 Relative Core Power for the Region I Analysis PCT Case

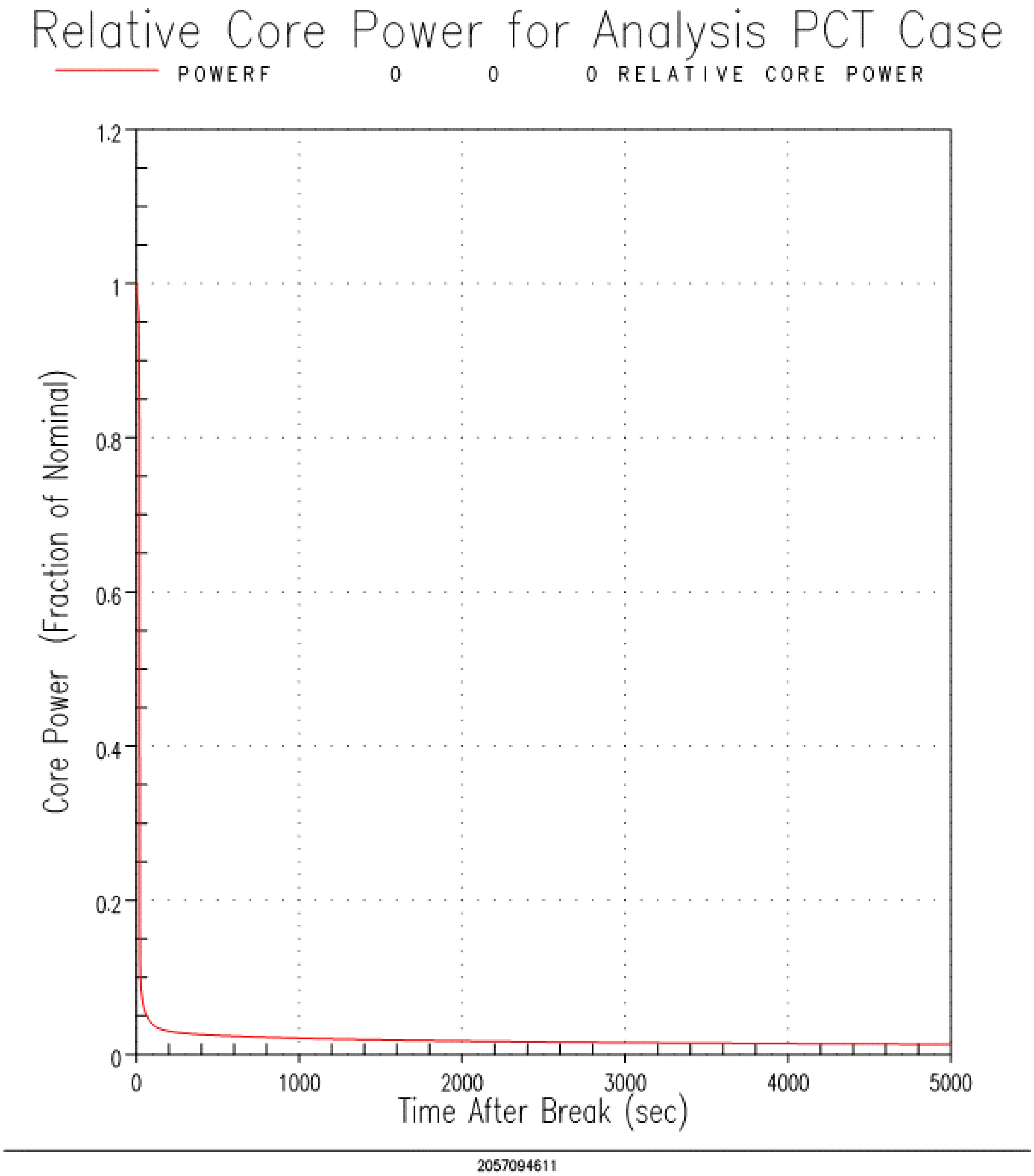

Figure 12b: Beaver Valley Unit 2 Relative Core Power for the Region I Analysis PCT Case

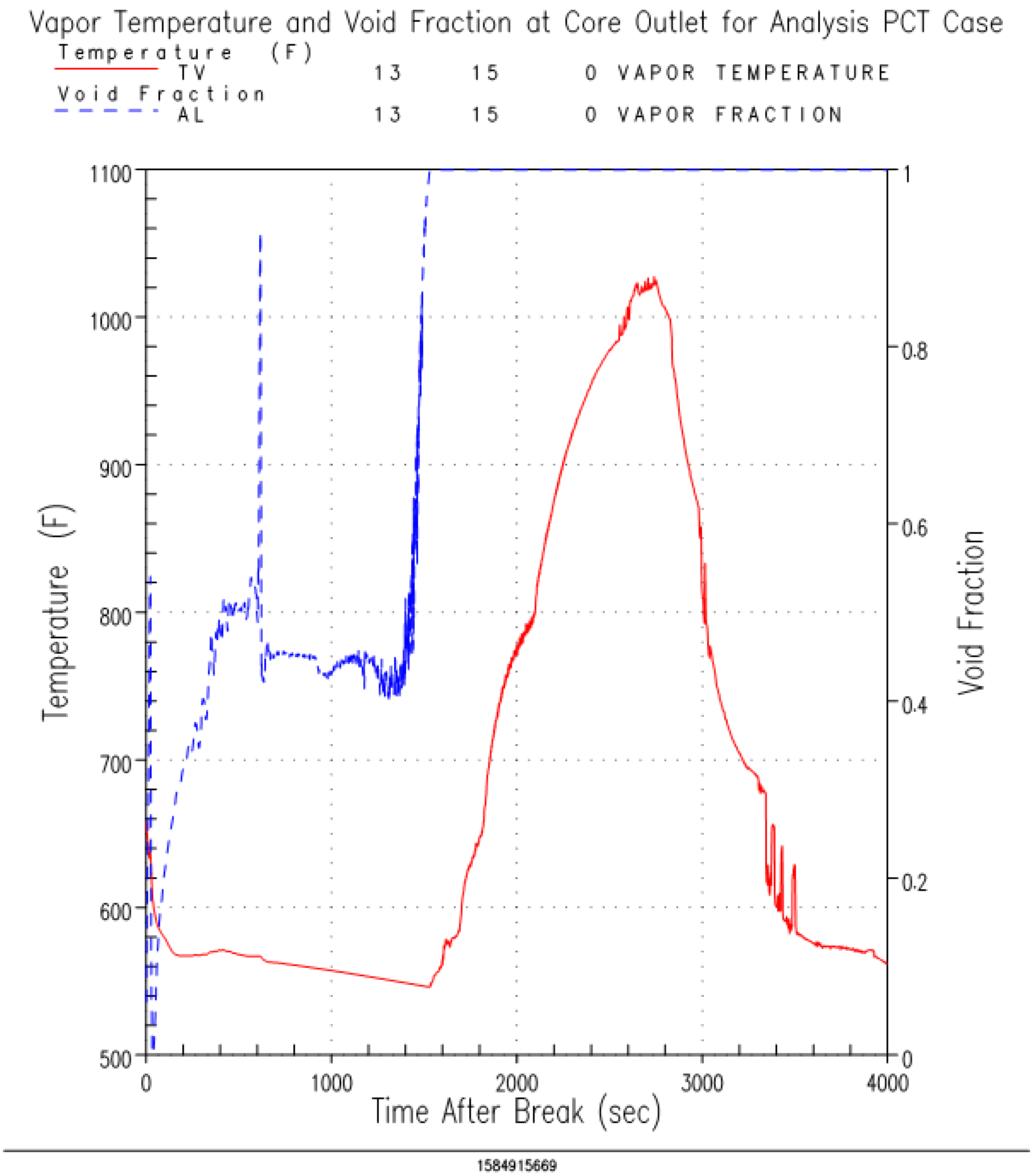

Figure 13a: Beaver Valley Unit 1 Vapor Temperature and Void Fraction at Core Outlet (Hot Assembly Channel) for the Region I Analysis PCT Case

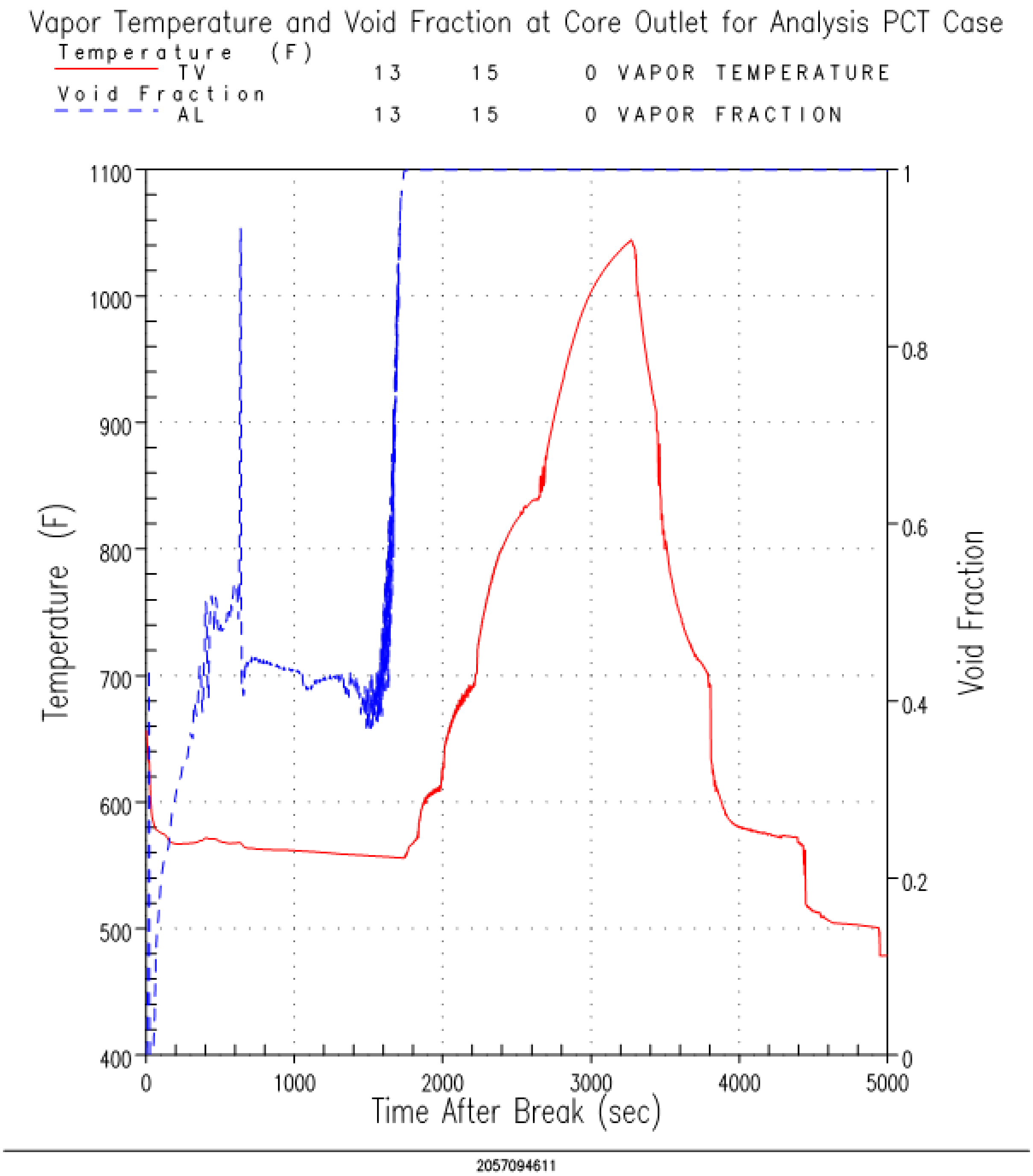

Figure 13b: Beaver Valley Unit 2 Vapor Temperature and Void Fraction at Core Outlet (Hot Assembly Channel) for the Region I Analysis PCT Case

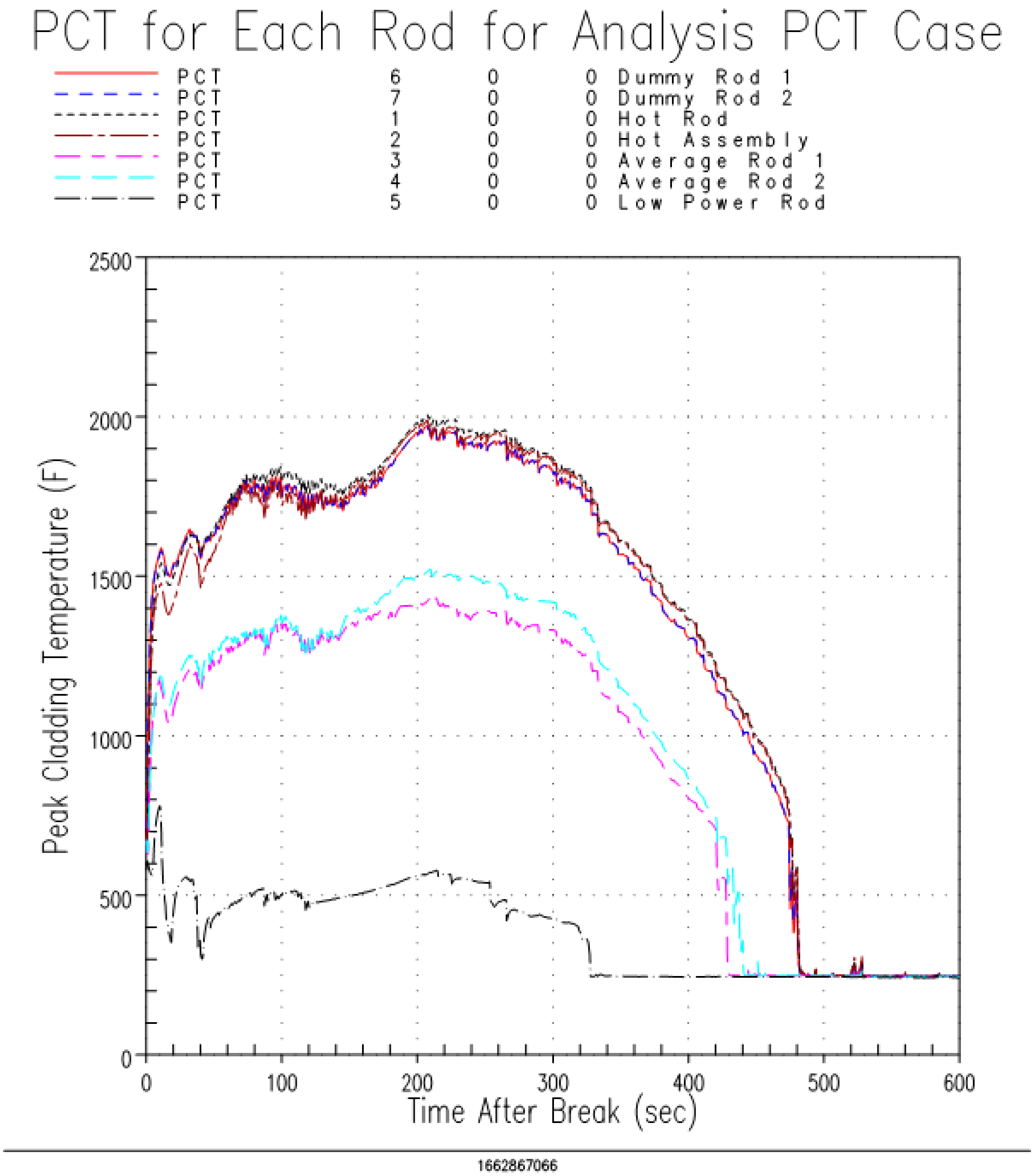

Figure 14a: Beaver Valley Unit 1 Peak Cladding Temperature for all Rods for the Region II Analysis PCT Case

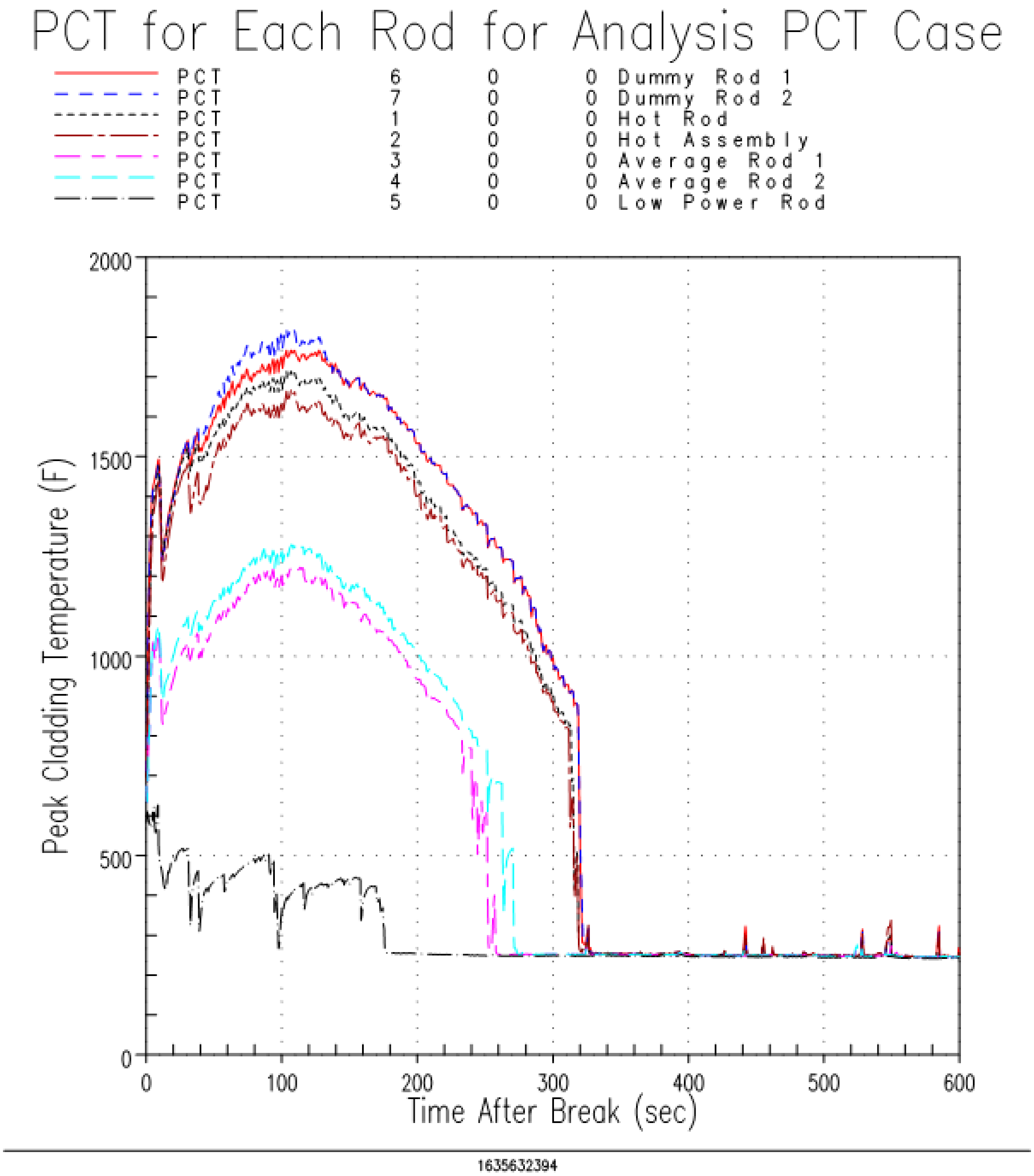

Figure 14b: Beaver Valley Unit 2 Peak Cladding Temperature for all Rods for the Region II Analysis PCT Case

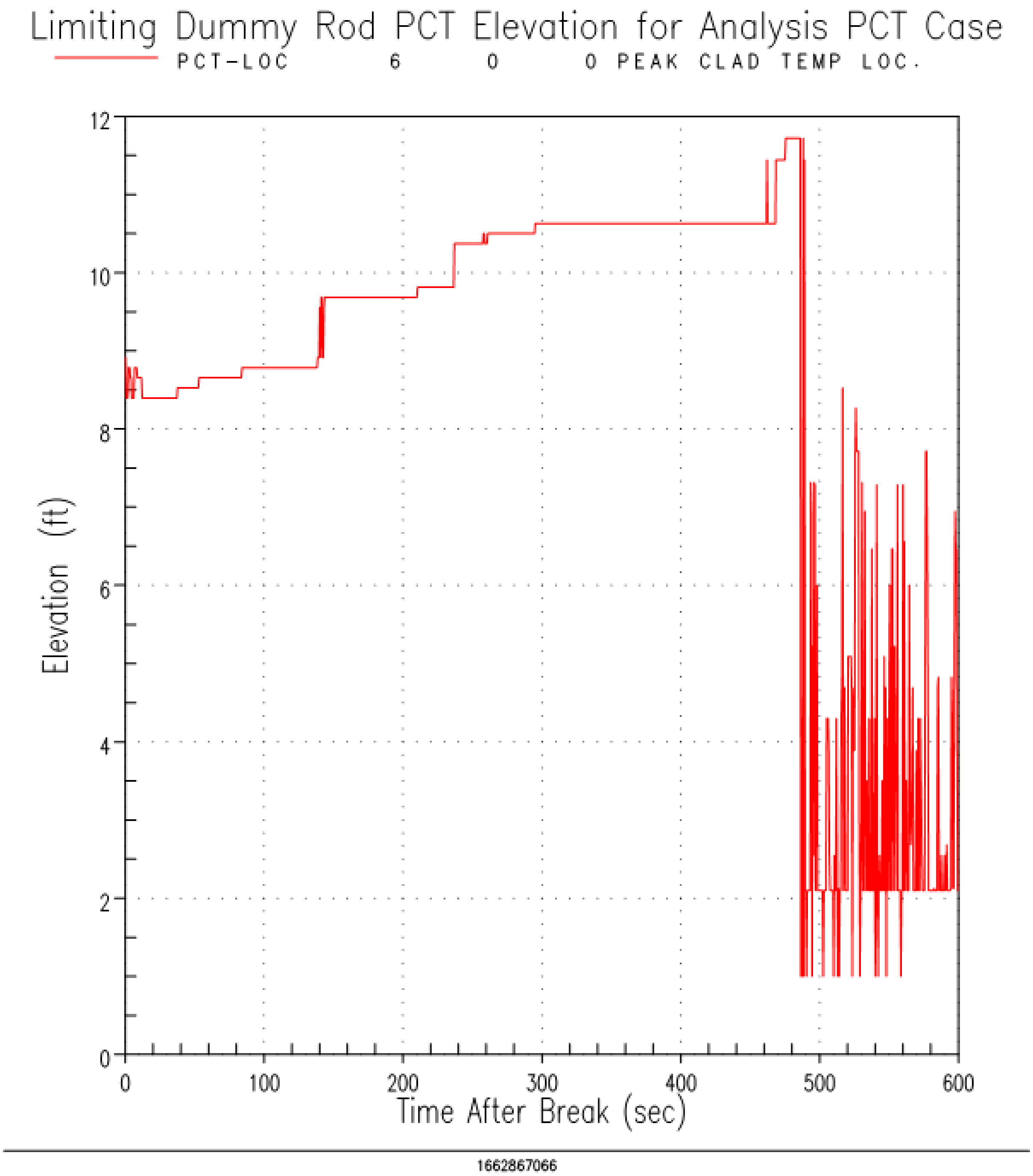

Figure 15a: Beaver Valley Unit 1 Peak Cladding Temperature Elevation (Relative to Bottom of Active Fuel) for the Region II Analysis PCT Case

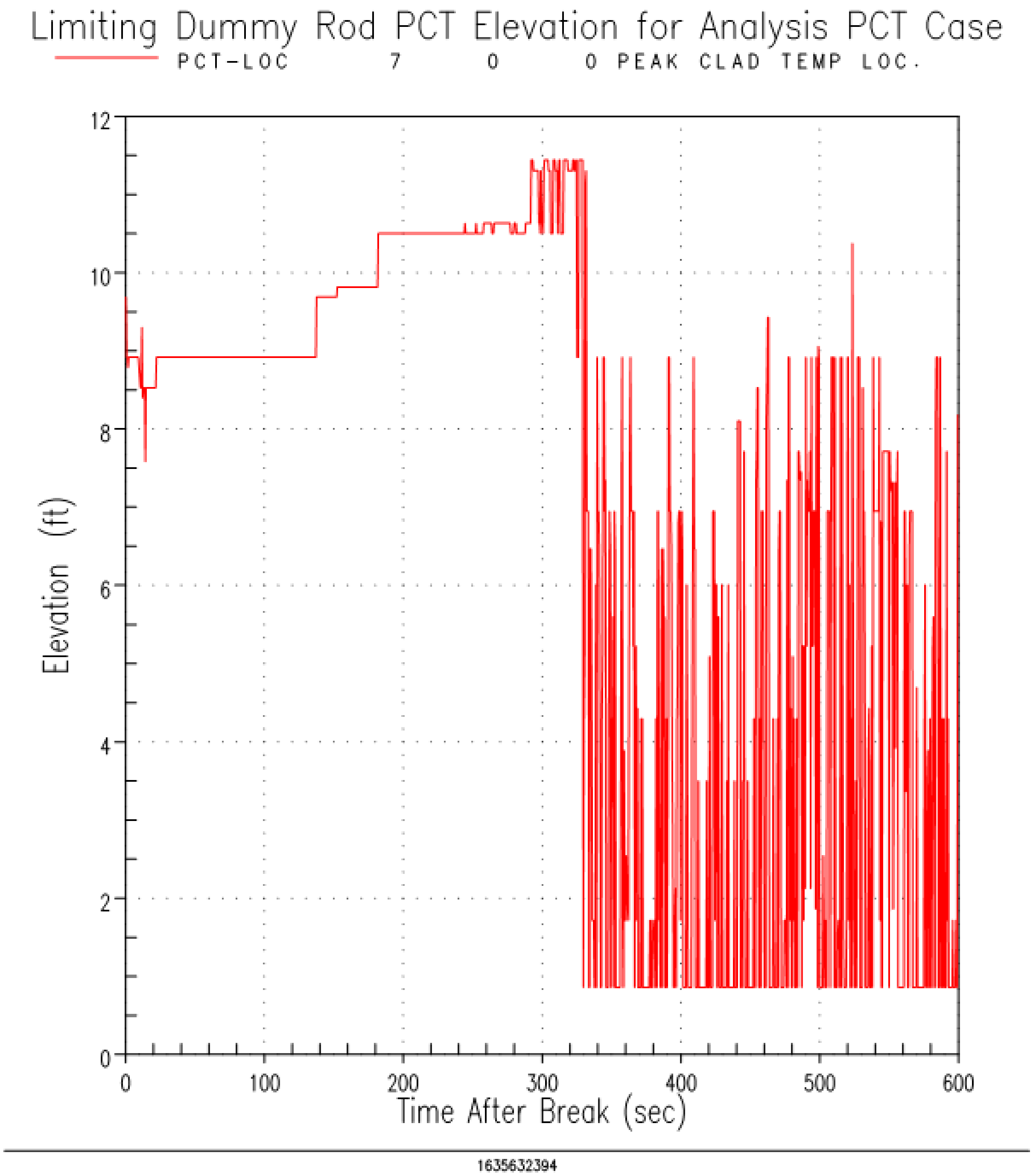

Figure 15b: Beaver Valley Unit 2 Peak Cladding Temperature Elevation (Relative to Bottom of Active Fuel) for the Region II Analysis PCT Case

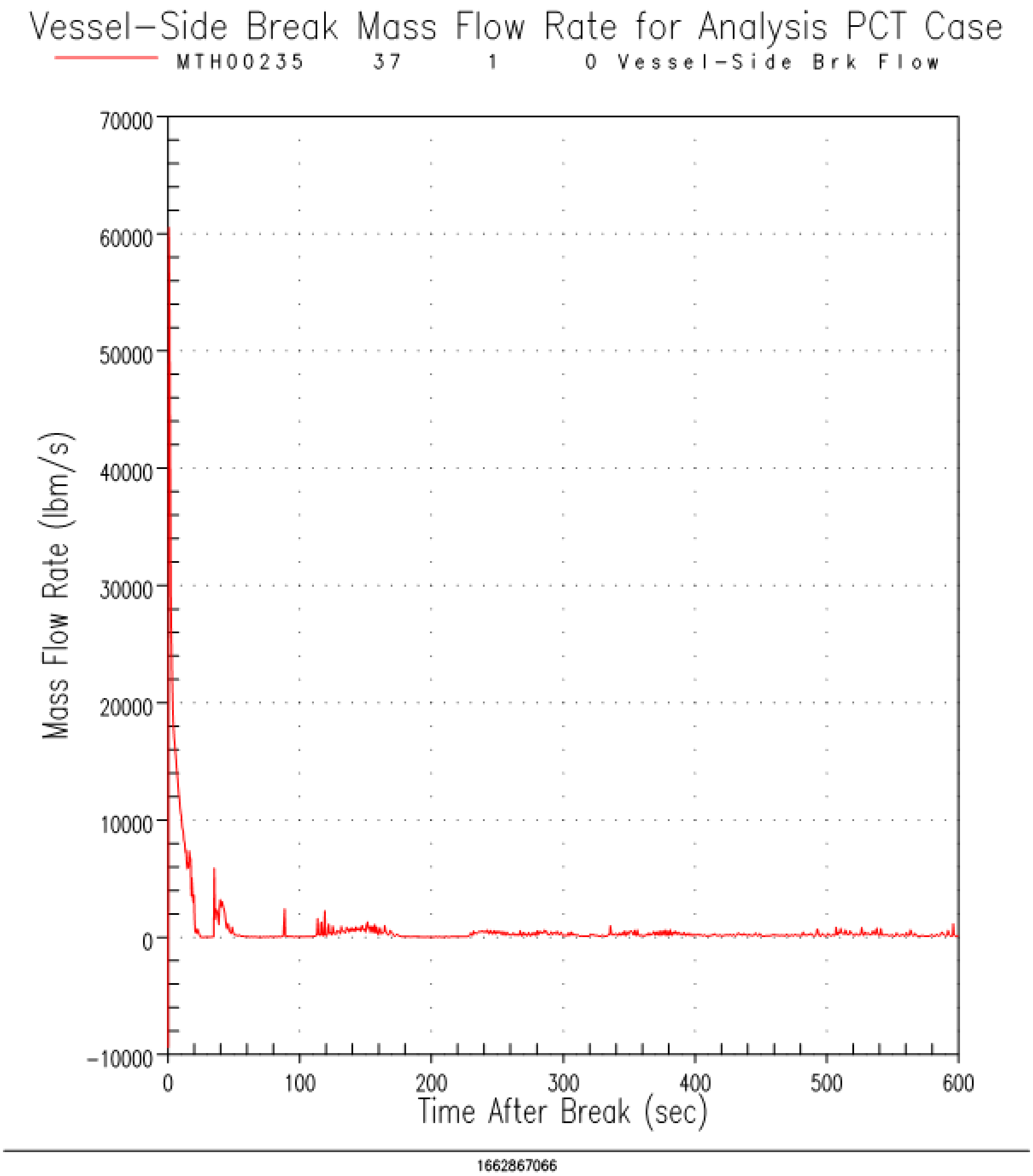

Figure 16a: Beaver Valley Unit 1 Vessel-Side Break Mass Flow Rate for the Region II Analysis PCT Case

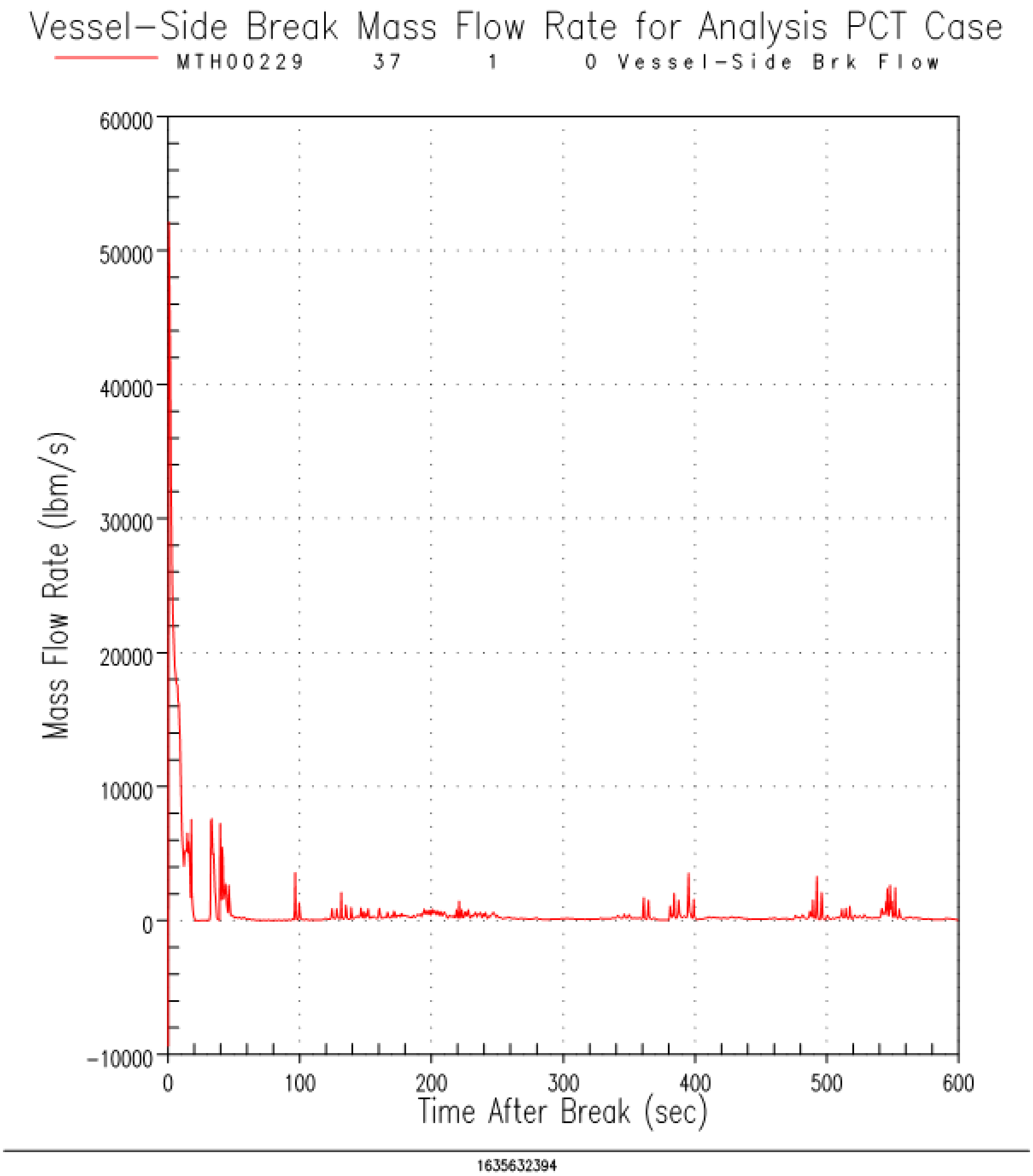

Figure 16b: Beaver Valley Unit 2 Vessel-Side Break Mass Flow Rate for the Region II Analysis PCT Case

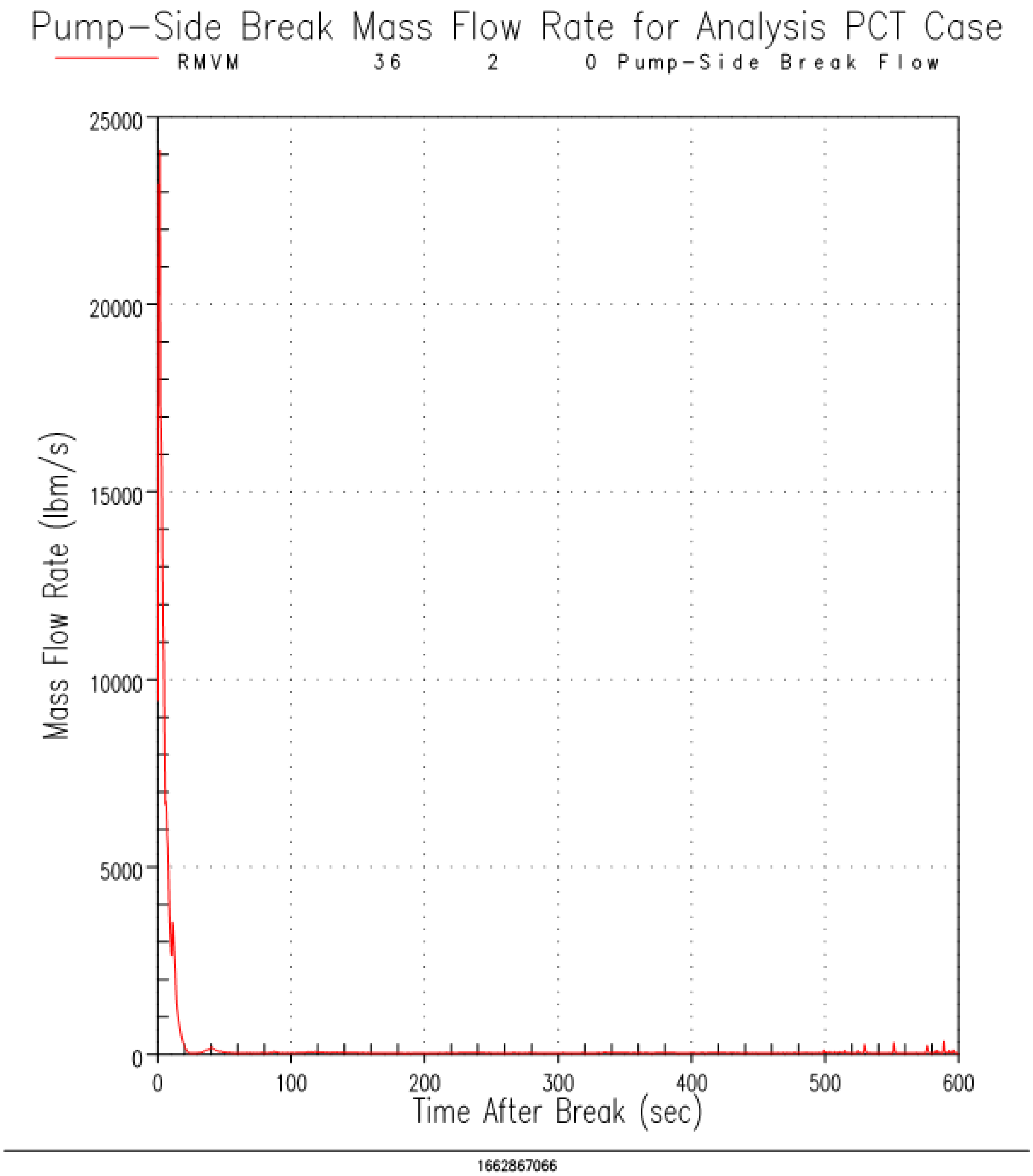

Figure 17a: Beaver Valley Unit 1 Pump-Side Break Mass Flow Rate for the Region II Analysis PCT Case

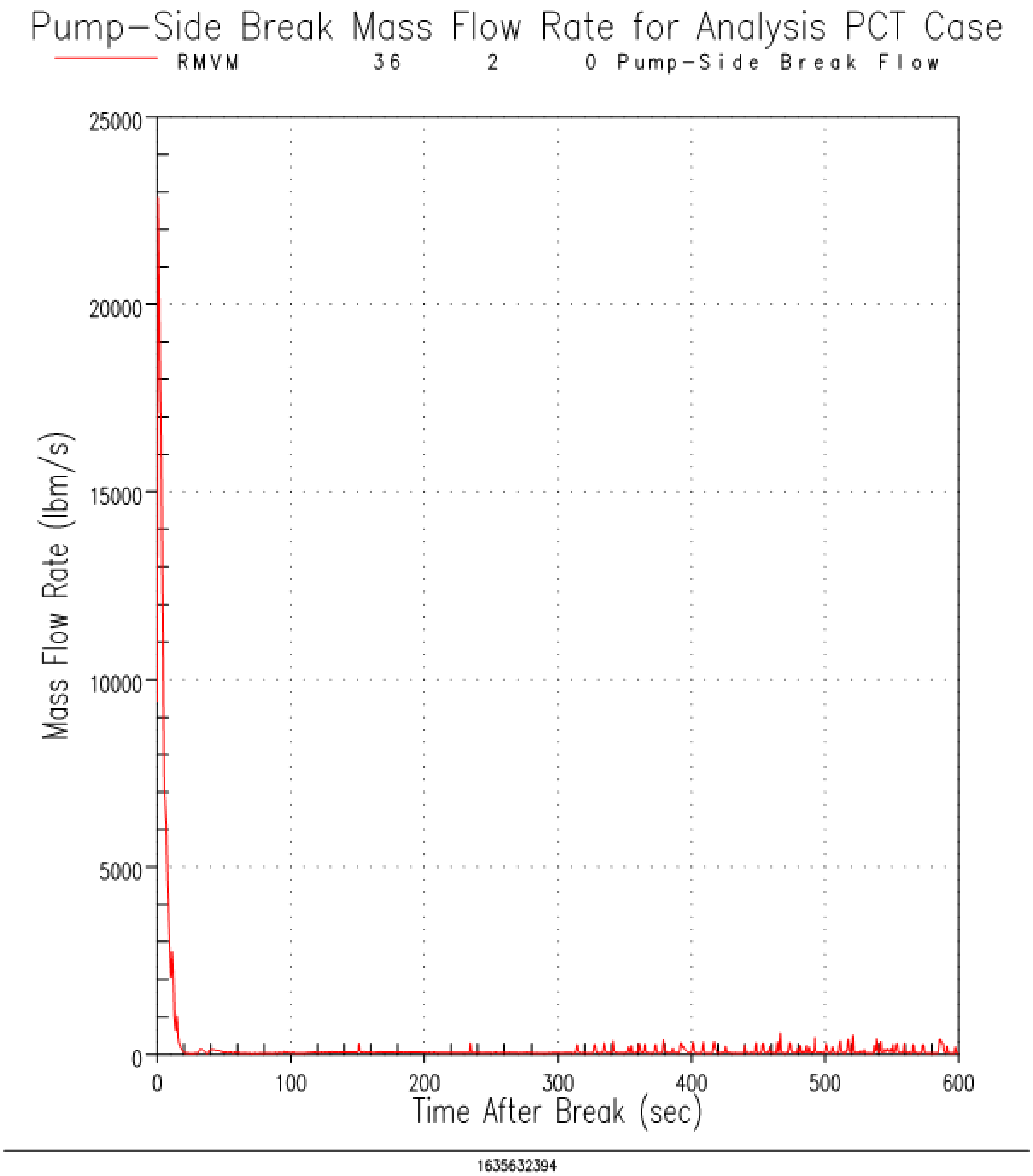

Figure 17b: Beaver Valley Unit 2 Pump-Side Break Mass Flow Rate for the Region II Analysis PCT Case

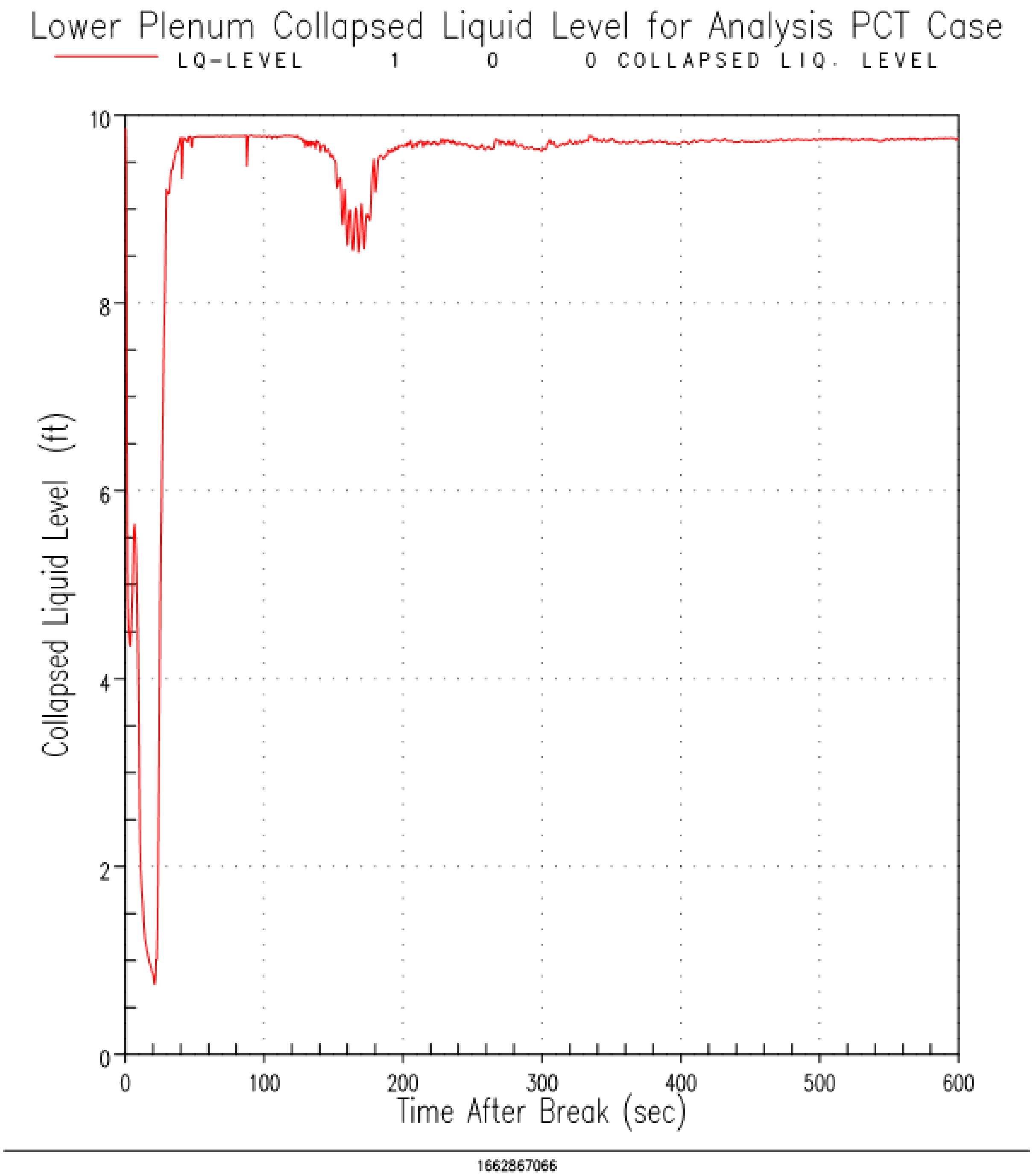

Figure 18a: Beaver Valley Unit 1 Lower Plenum Collapsed Liquid Level (Relative to Inside Bottom of Vessel) for the Region II Analysis PCT Case

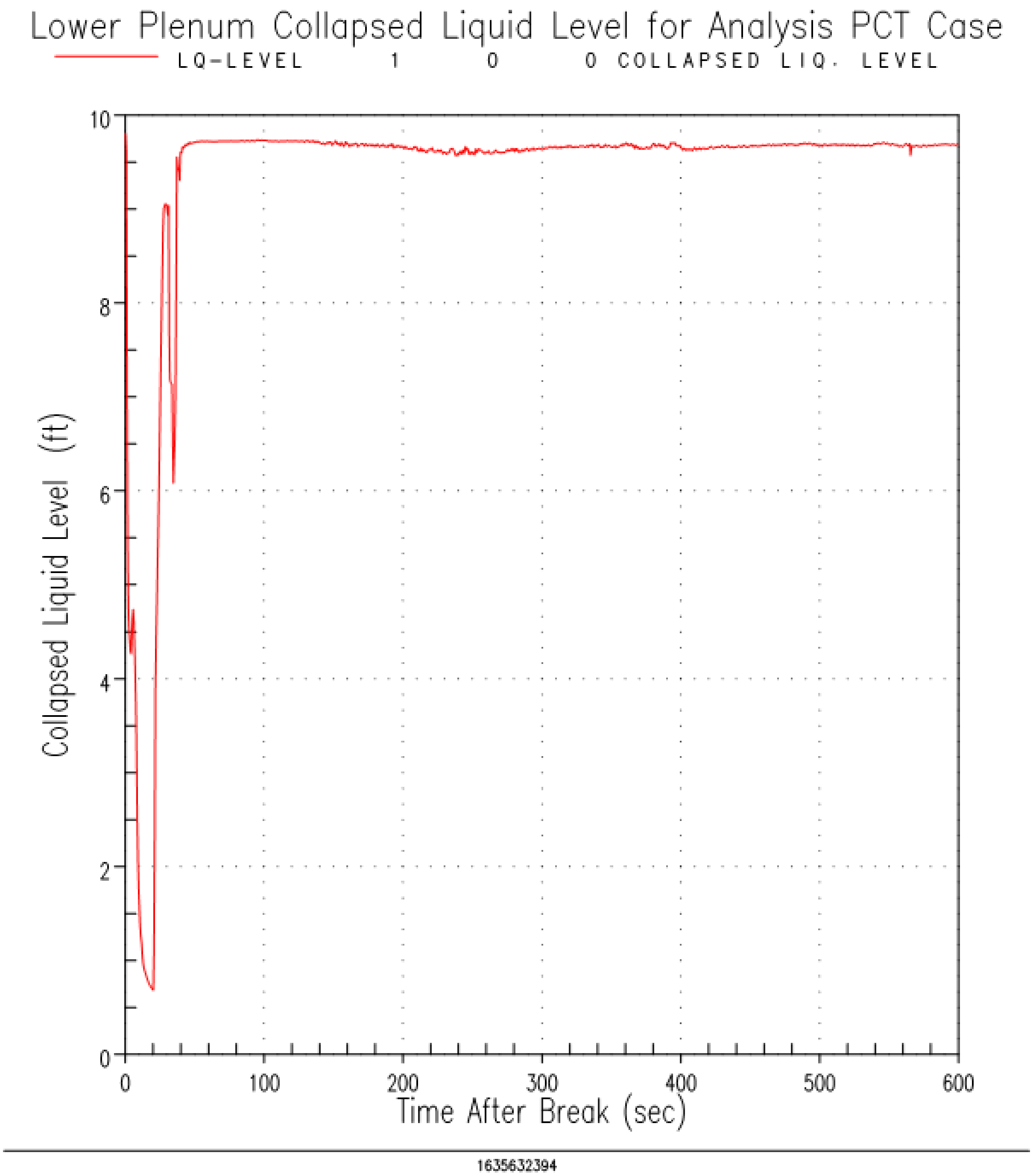

Figure 18b: Beaver Valley Unit 2 Lower Plenum Collapsed Liquid Level (Relative to Inside Bottom of Vessel) for the Region II Analysis PCT Case

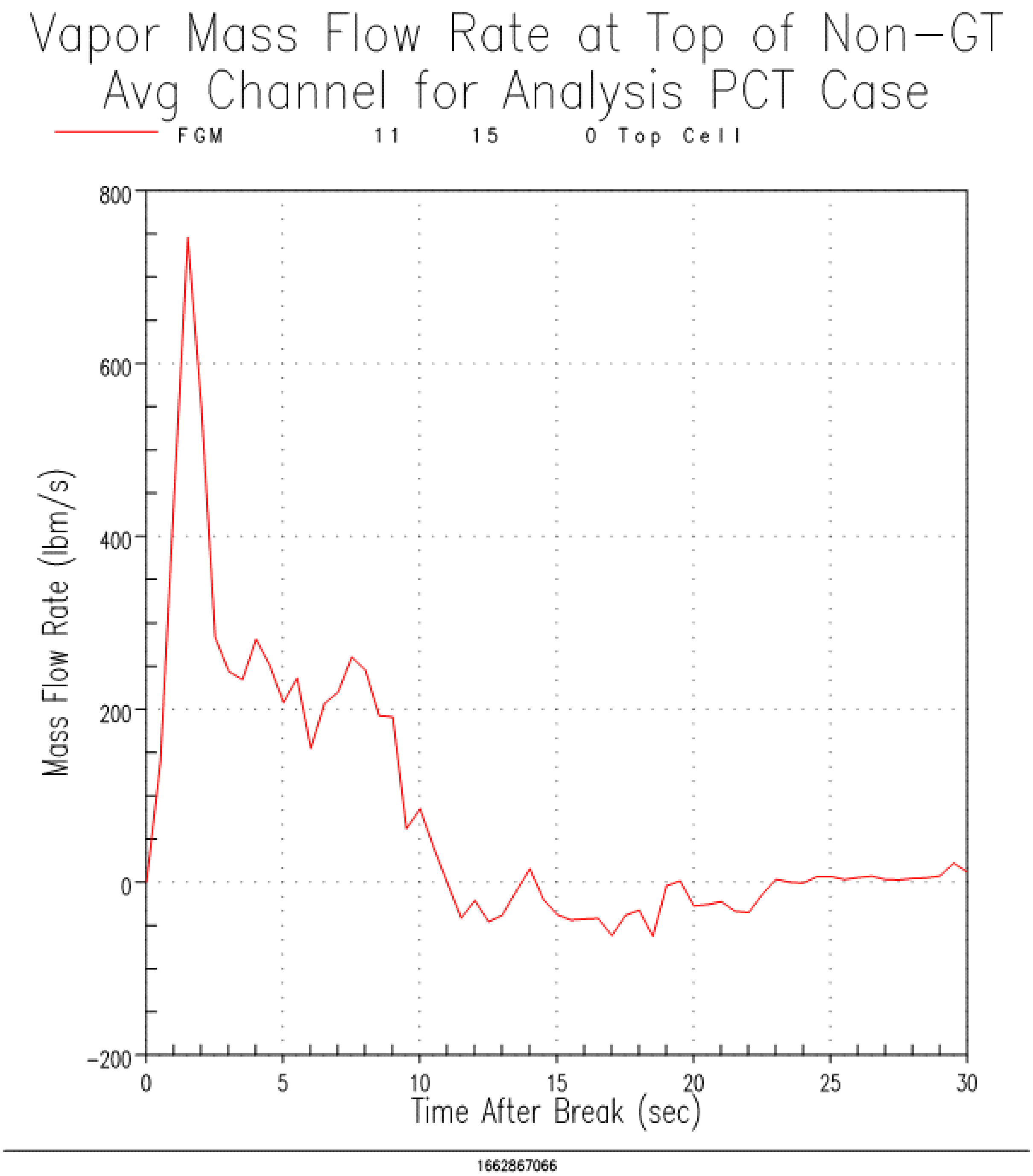

Figure 19a: Beaver Valley Unit 1 Vapor Mass Flow Rate at the Top Cell Face of the Core Average Channel not Under Guide Tubes for the Region II Analysis PCT Case

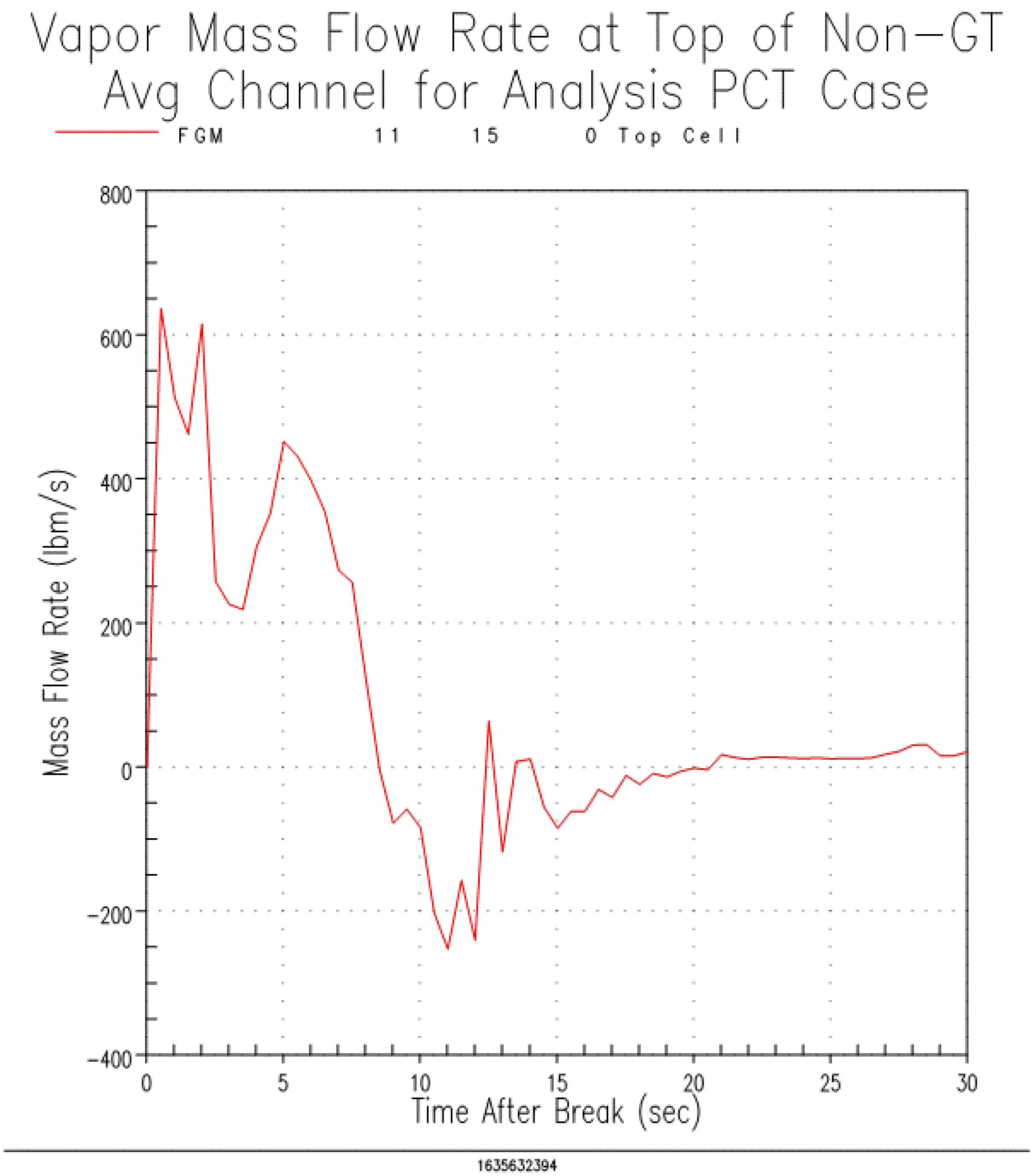

Figure 19b: Beaver Valley Unit 2 Vapor Mass Flow Rate at the Top Cell Face of the Core Average Channel not Under Guide Tubes for the Region II Analysis PCT Case

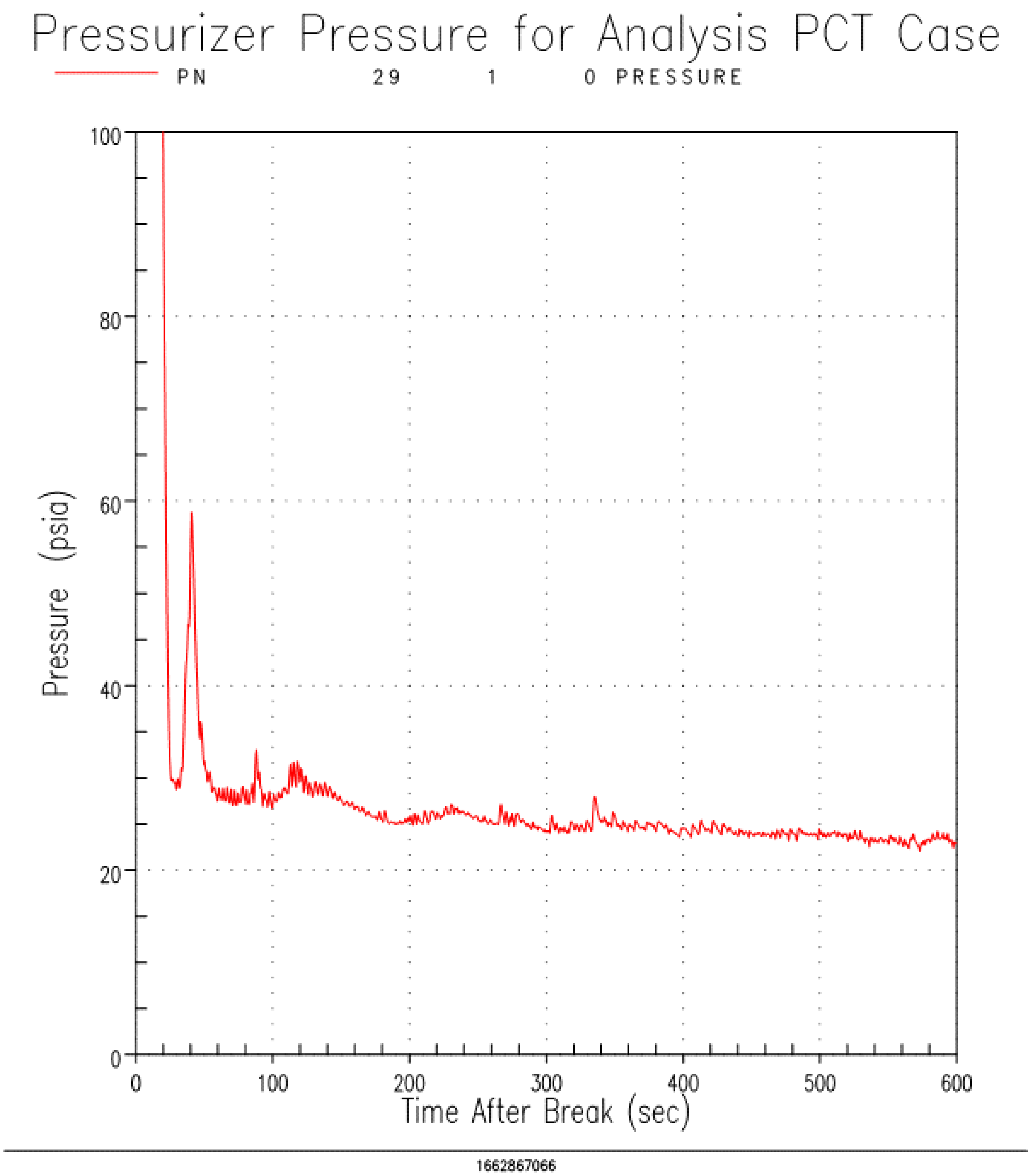

Figure 20a: Beaver Valley Unit 1 RCS Pressure for the Region II Analysis PCT Case

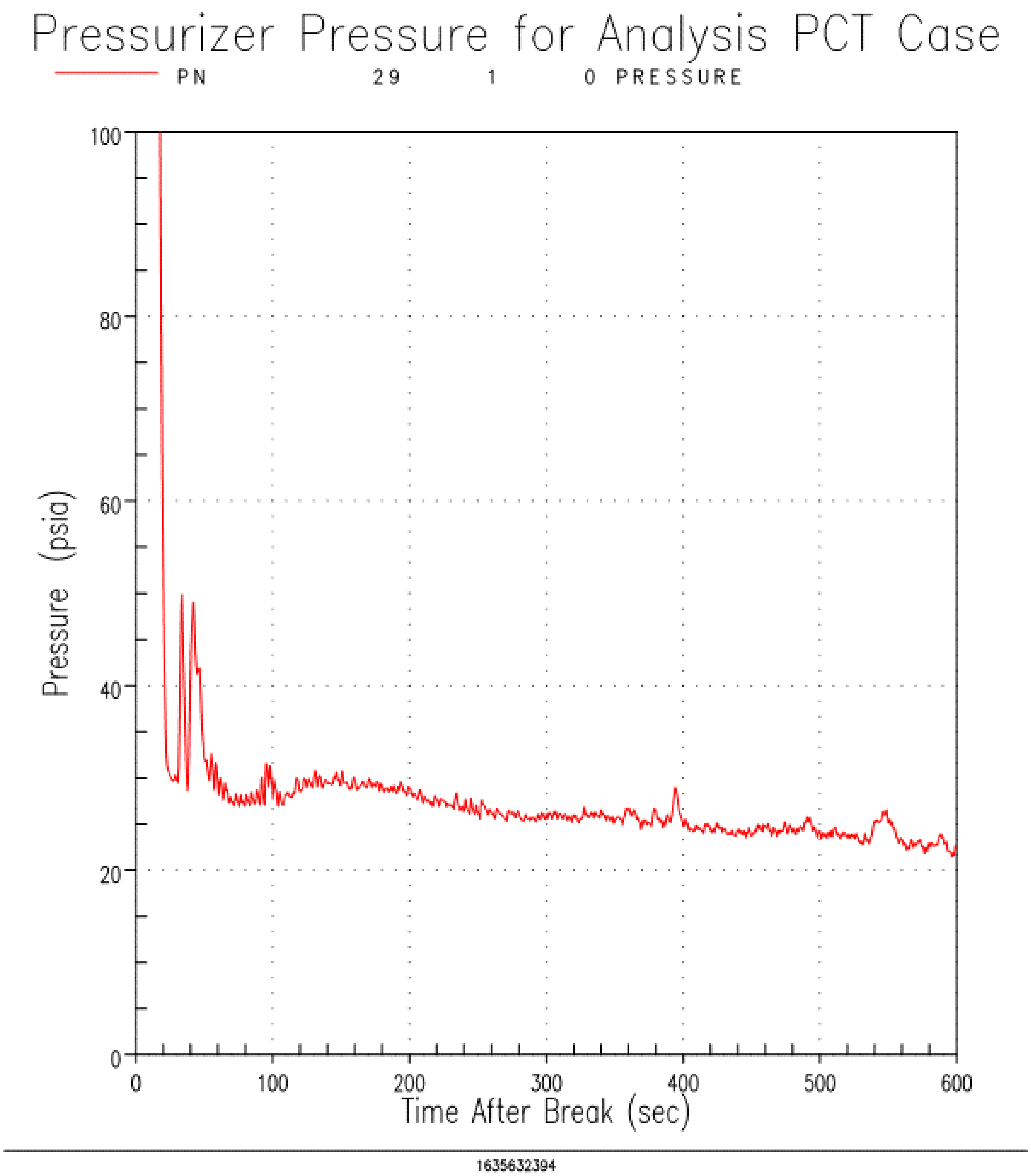

Figure 20b: Beaver Valley Unit 2 RCS Pressure for the Region II Analysis PCT Case

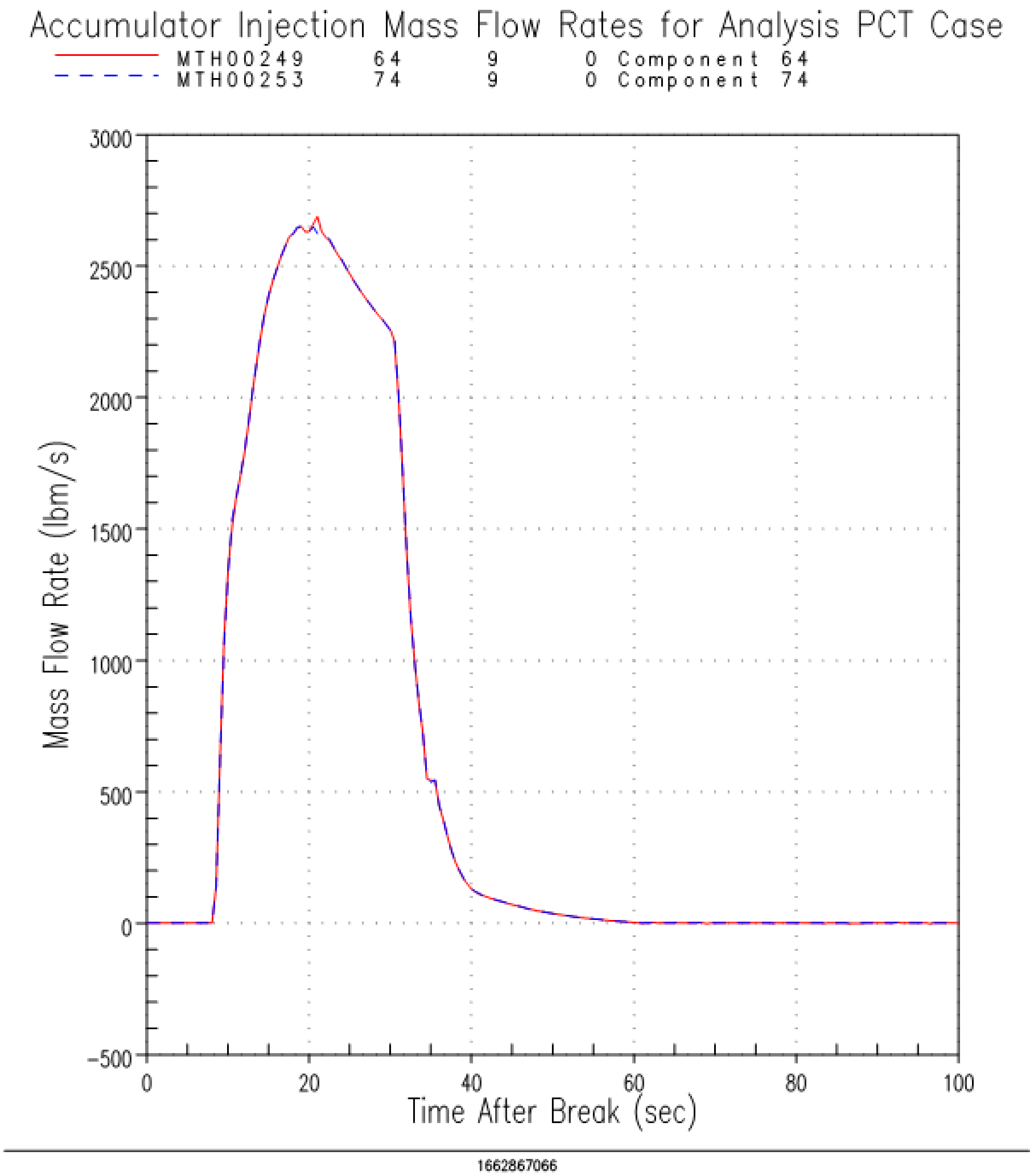

Figure 21a: Beaver Valley Unit 1 Accumulator Injection Flow per Loop for the Region II Analysis PCT Case

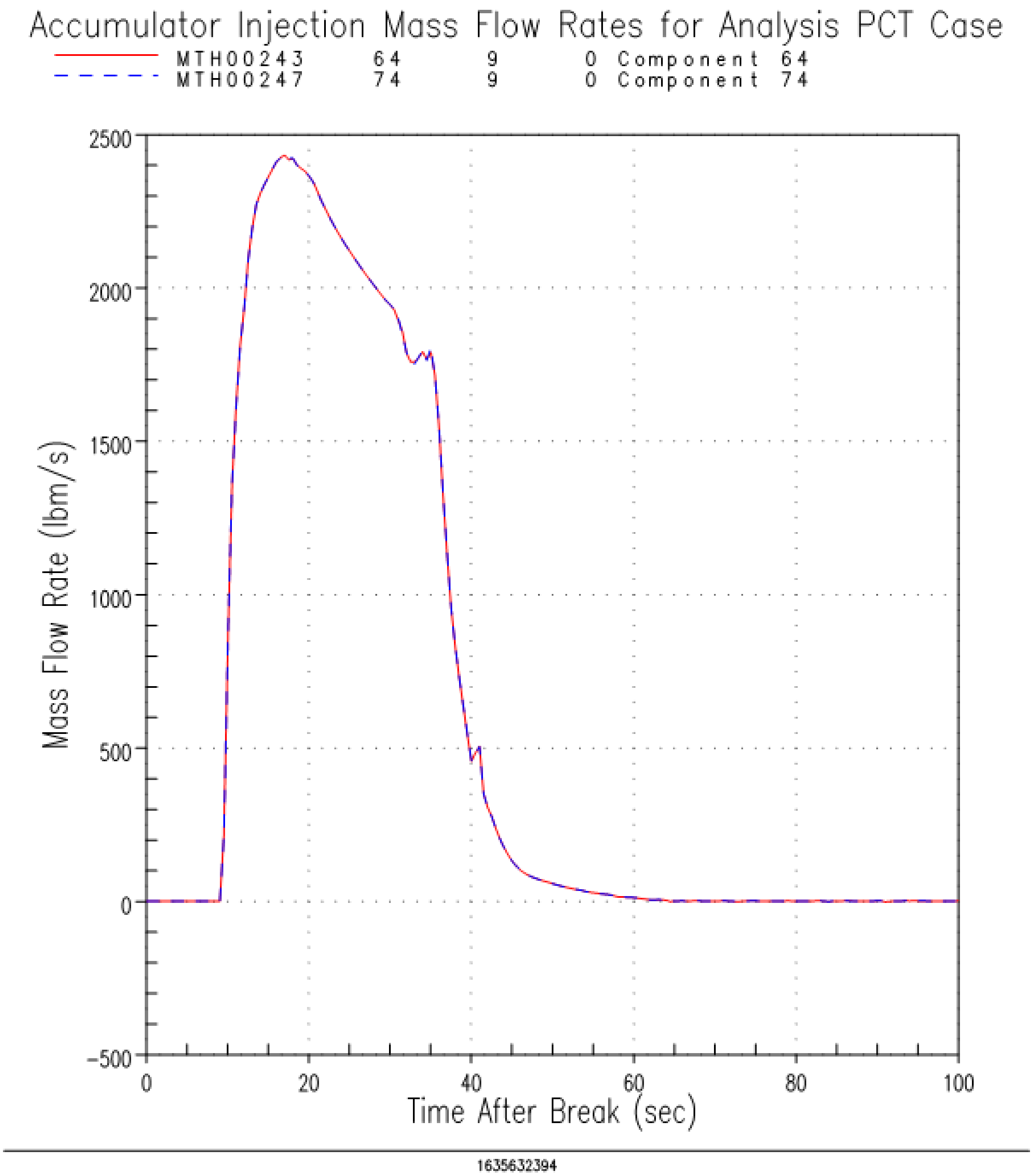

Figure 21b: Beaver Valley Unit 2 Accumulator Injection Flow per Loop for the Region II Analysis PCT Case

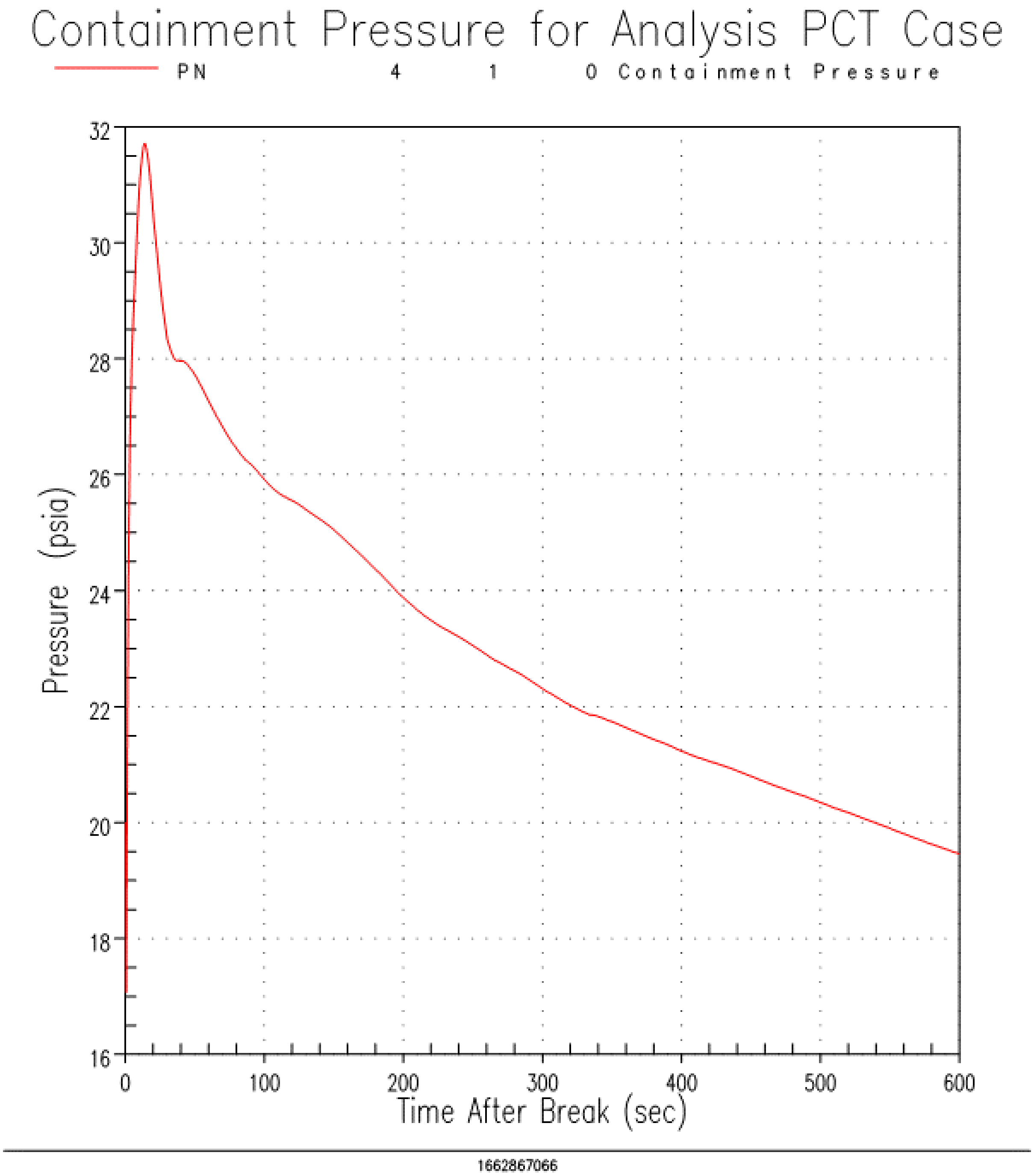

Figure 22a: Beaver Valley Unit 1 Containment Pressure for the Region II Analysis PCT Case

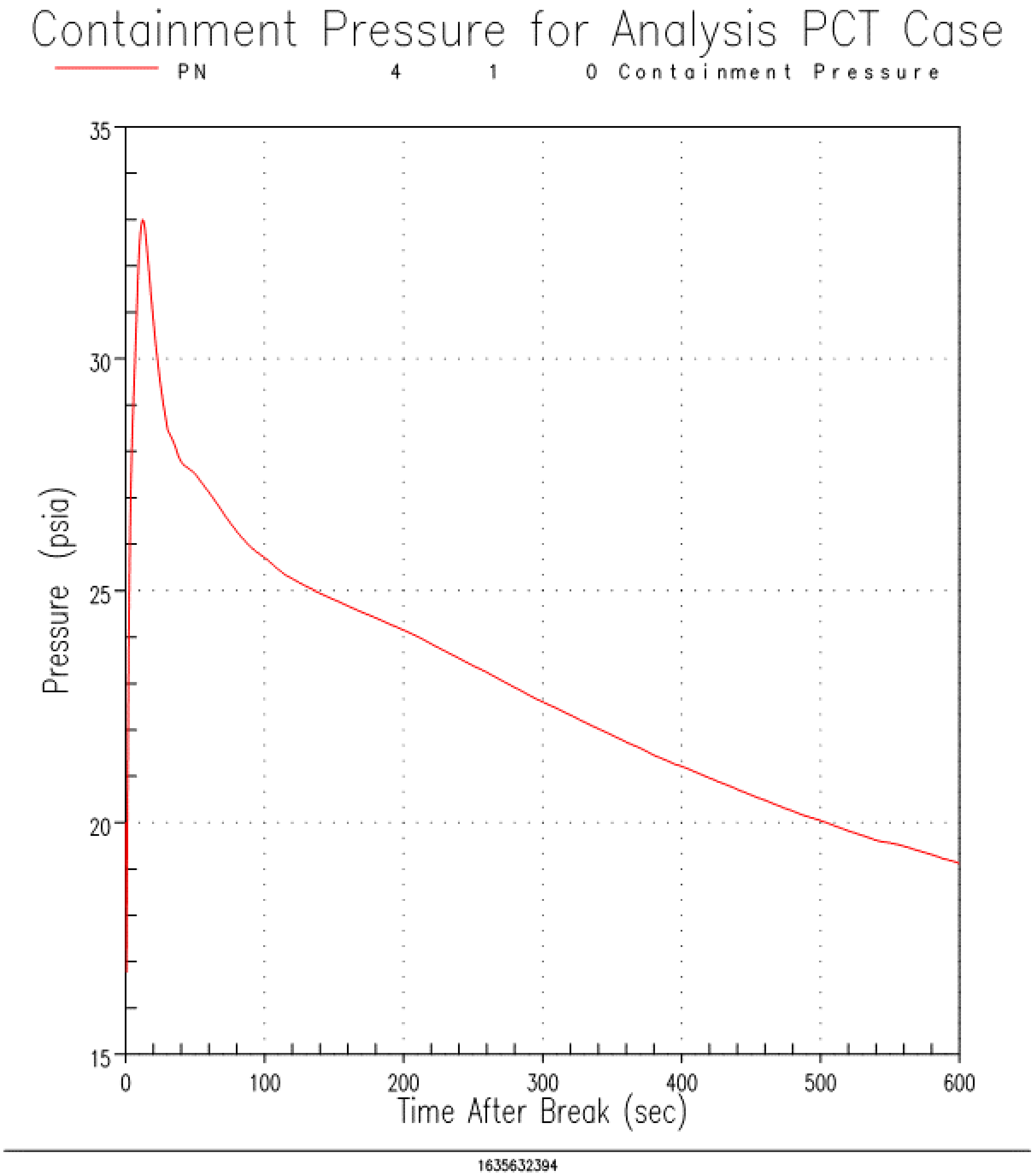

Figure 22b: Beaver Valley Unit 2 Containment Pressure for the Region II Analysis PCT Case

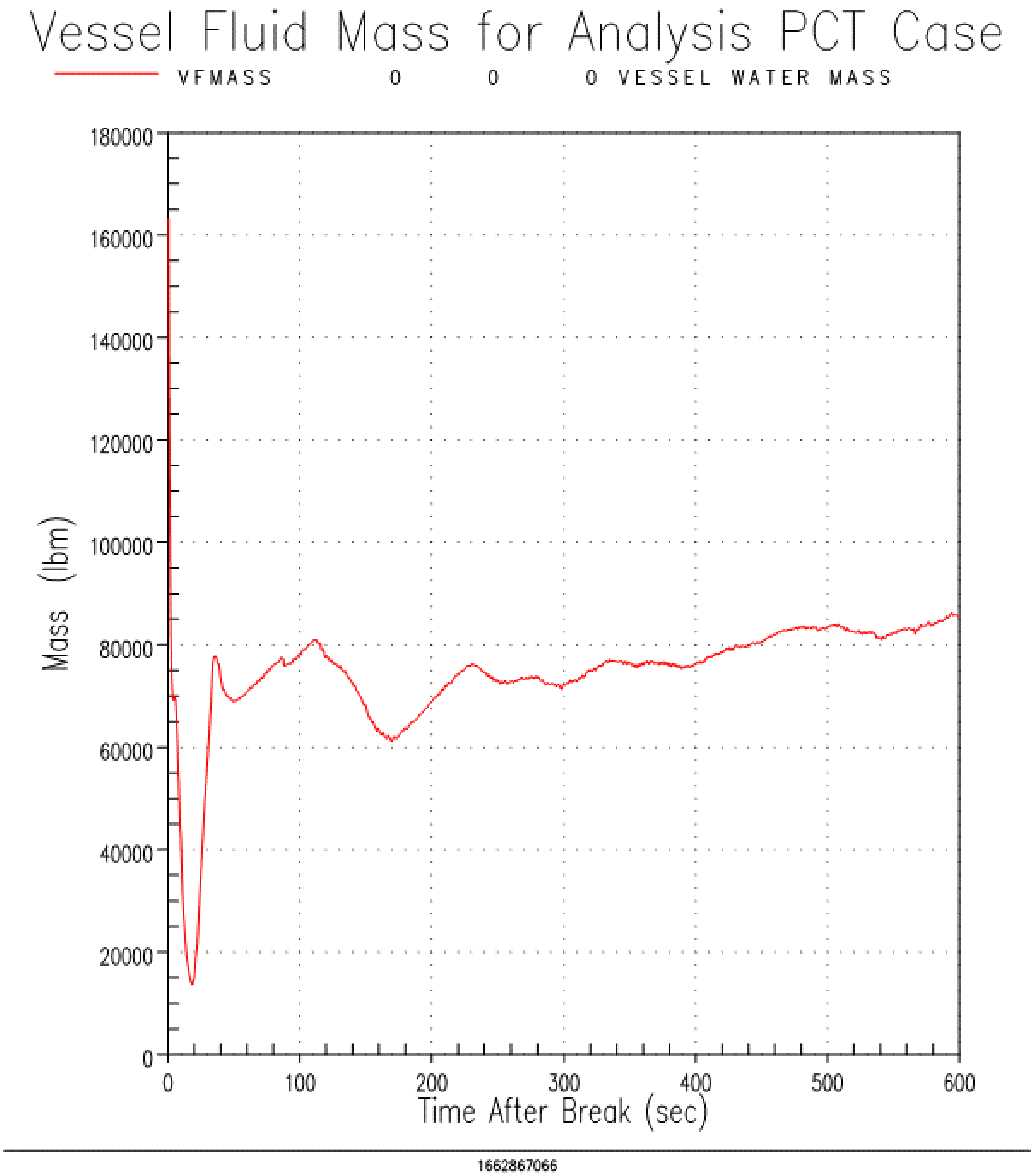

Figure 23a: Beaver Valley Unit 1 Vessel Fluid Mass for the Region II Analysis PCT Case

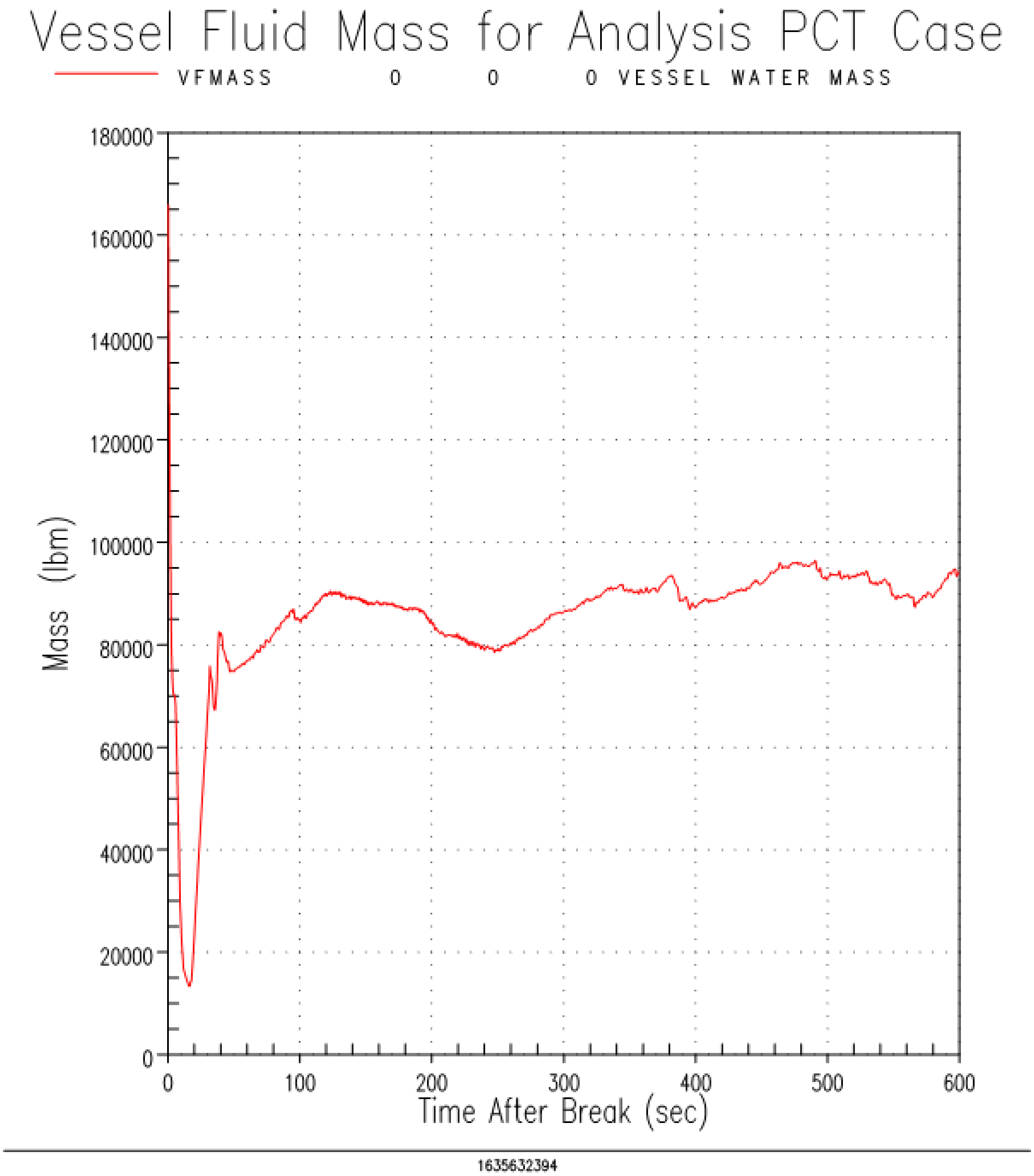

Figure 23b: Beaver Valley Unit 2 Vessel Fluid Mass for the Region II Analysis PCT Case

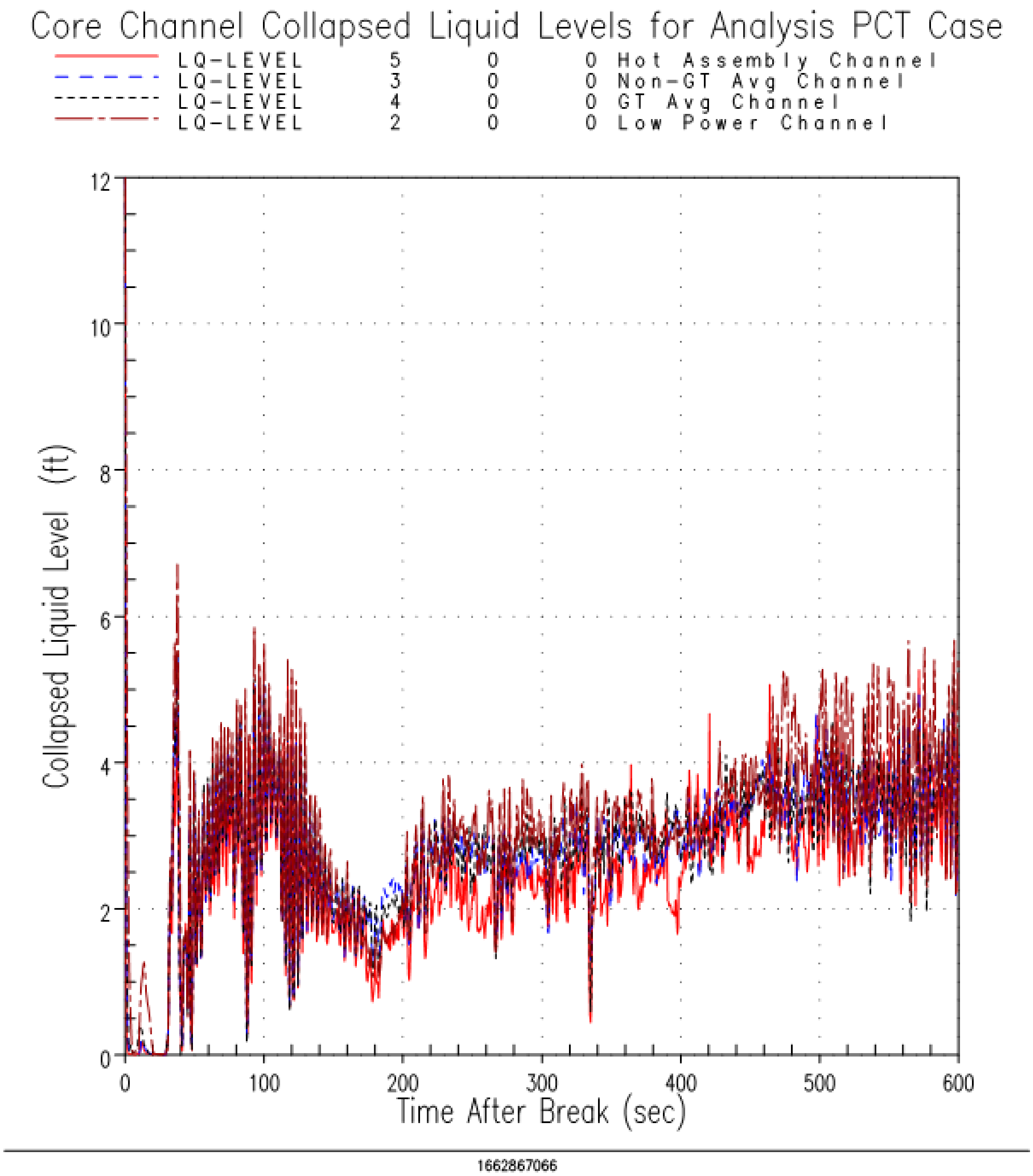

Figure 24a: Beaver Valley Unit 1 Collapsed Liquid Level for Each Core Channel (Relative to Bottom of Active Fuel) for the Region II Analysis PCT Case

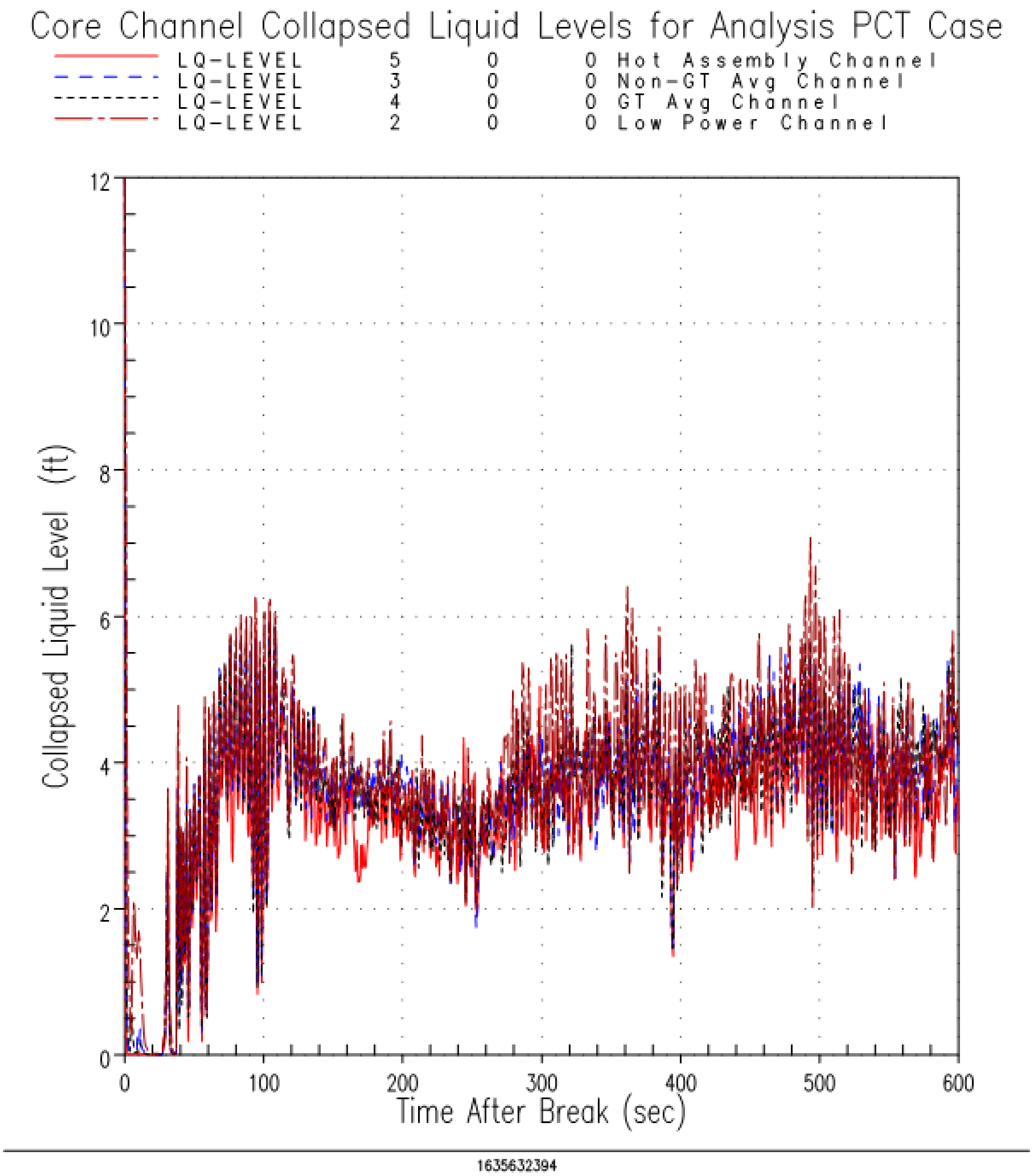

Figure 24b: Beaver Valley Unit 2 Collapsed Liquid Level for Each Core Channel (Relative to Bottom of Active Fuel) for the Region II Analysis PCT Case

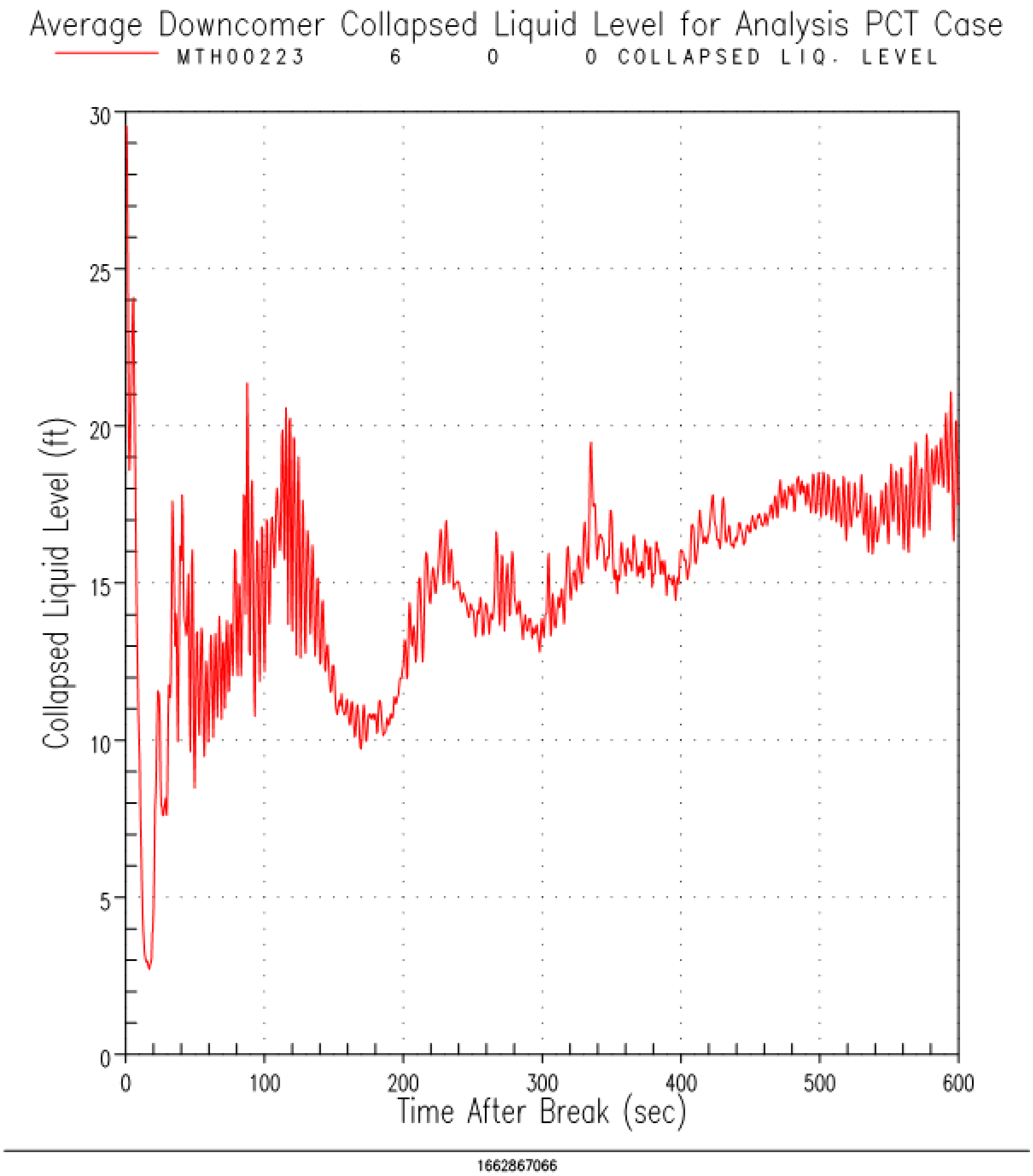

Figure 25a: Beaver Valley Unit 1 Average Downcomer Collapsed Liquid Level (Relative to Bottom of the Upper Tie Plate) for the Region II Analysis PCT Case

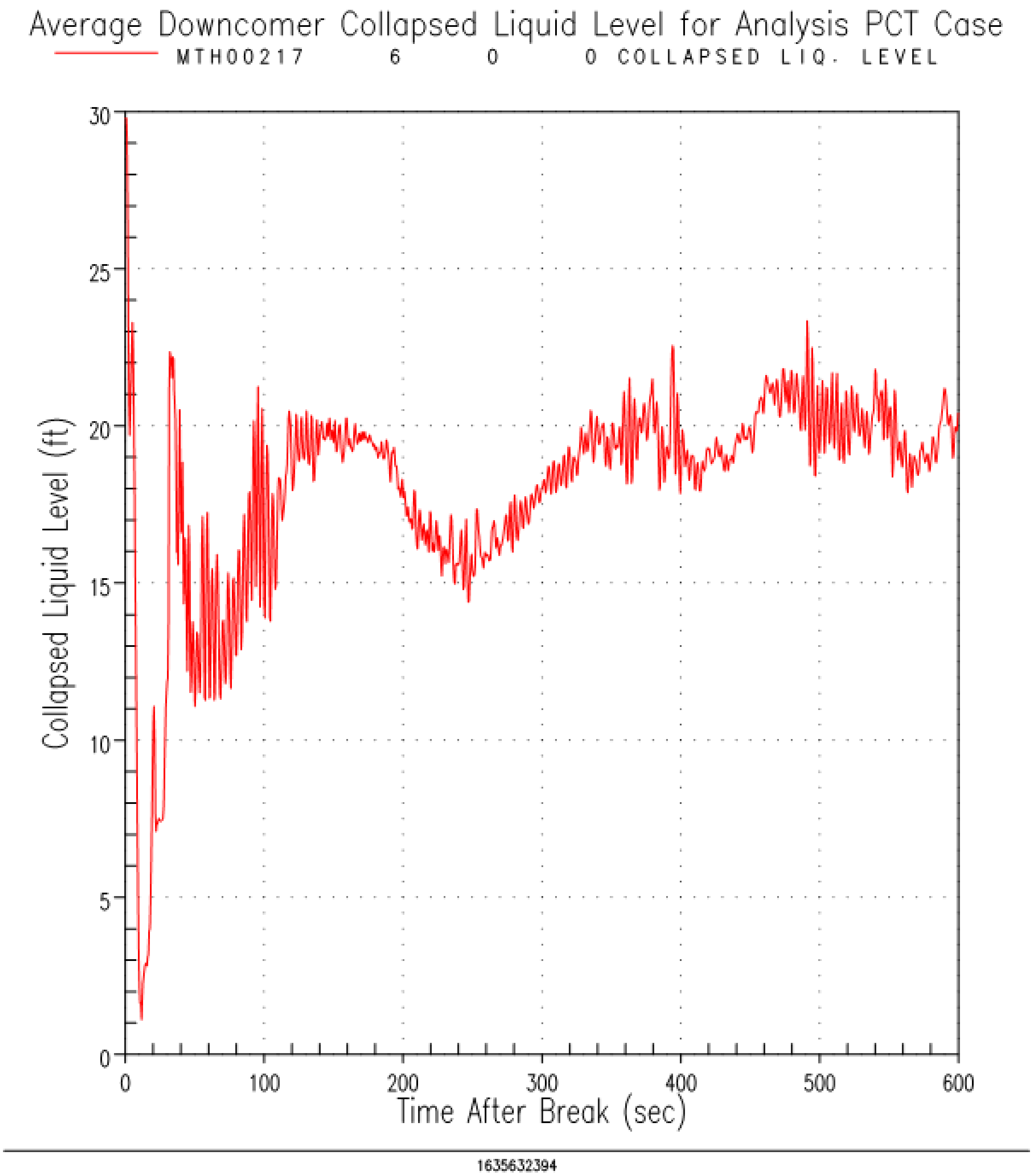

Figure 25b: Beaver Valley Unit 2 Average Downcomer Collapsed Liquid Level (Relative to Bottom of the Upper Tie Plate) for the Region II Analysis PCT Case

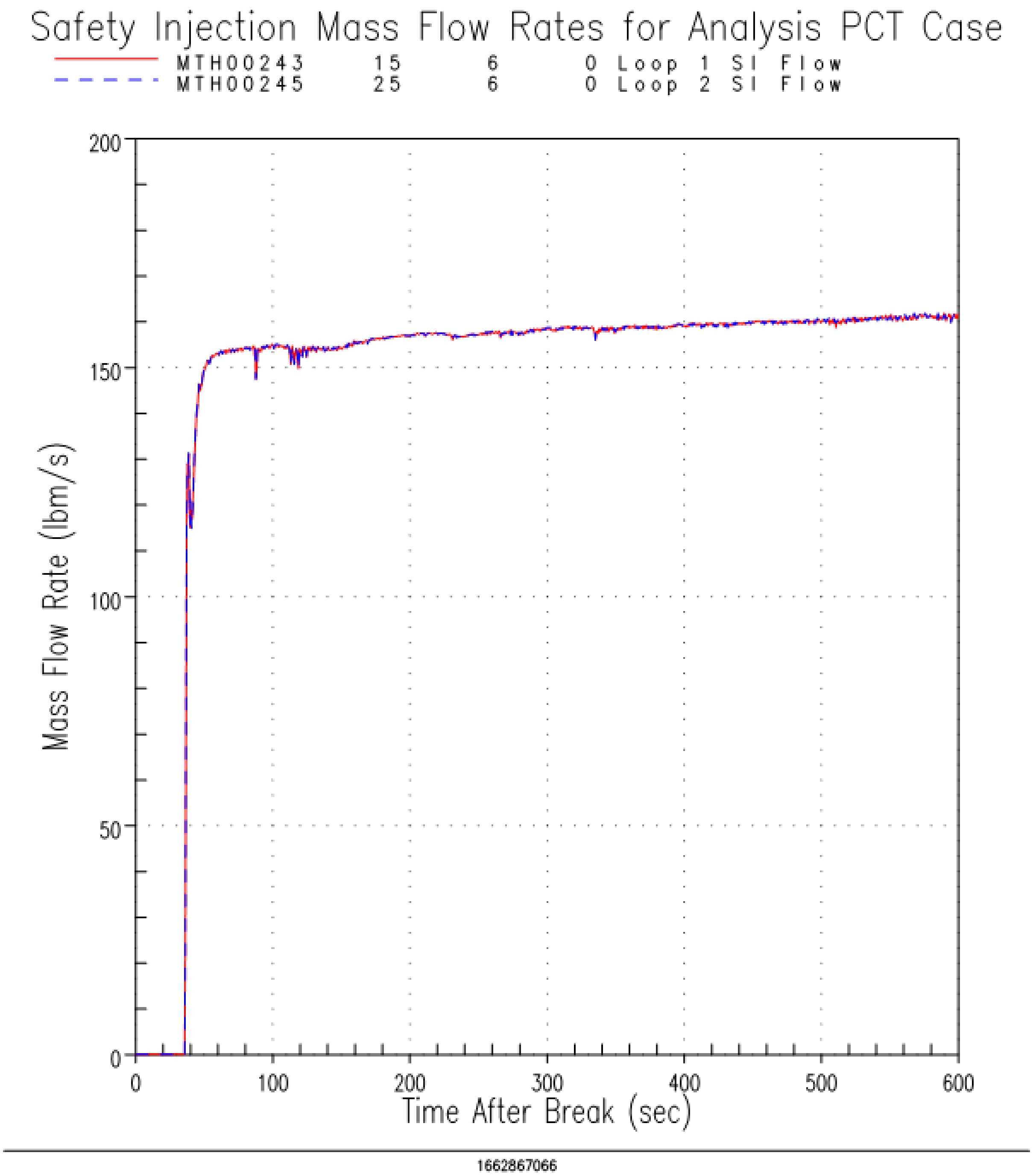

Figure 26a: Beaver Valley Unit 1 Safety Injection Flow per Loop (not including Accumulator Injection Flow) for the Region II Analysis PCT Case

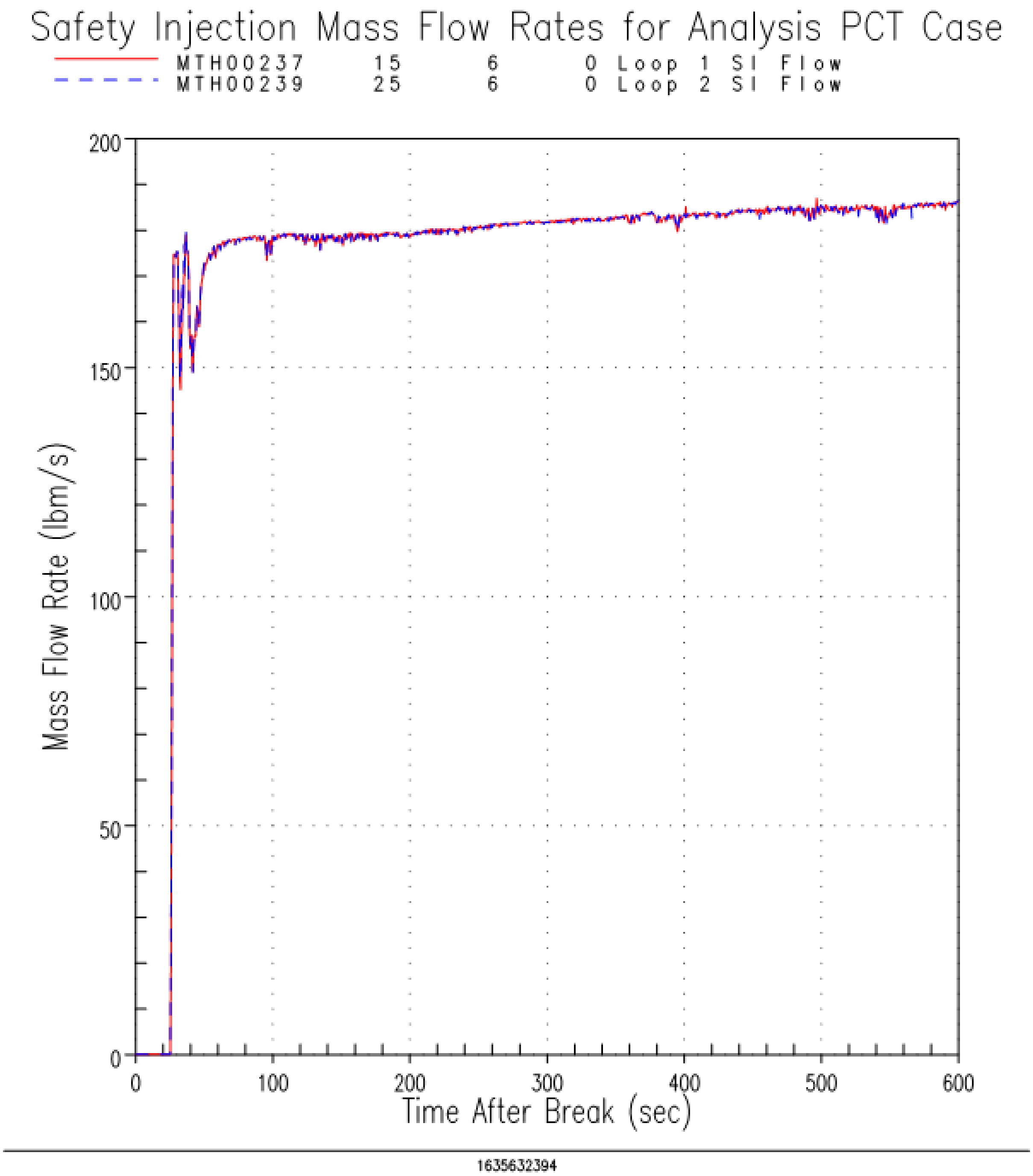

Figure 26b: Beaver Valley Unit 2 Safety Injection Flow per Loop (not including Accumulator Injection Flow) for the Region II Analysis PCT Case

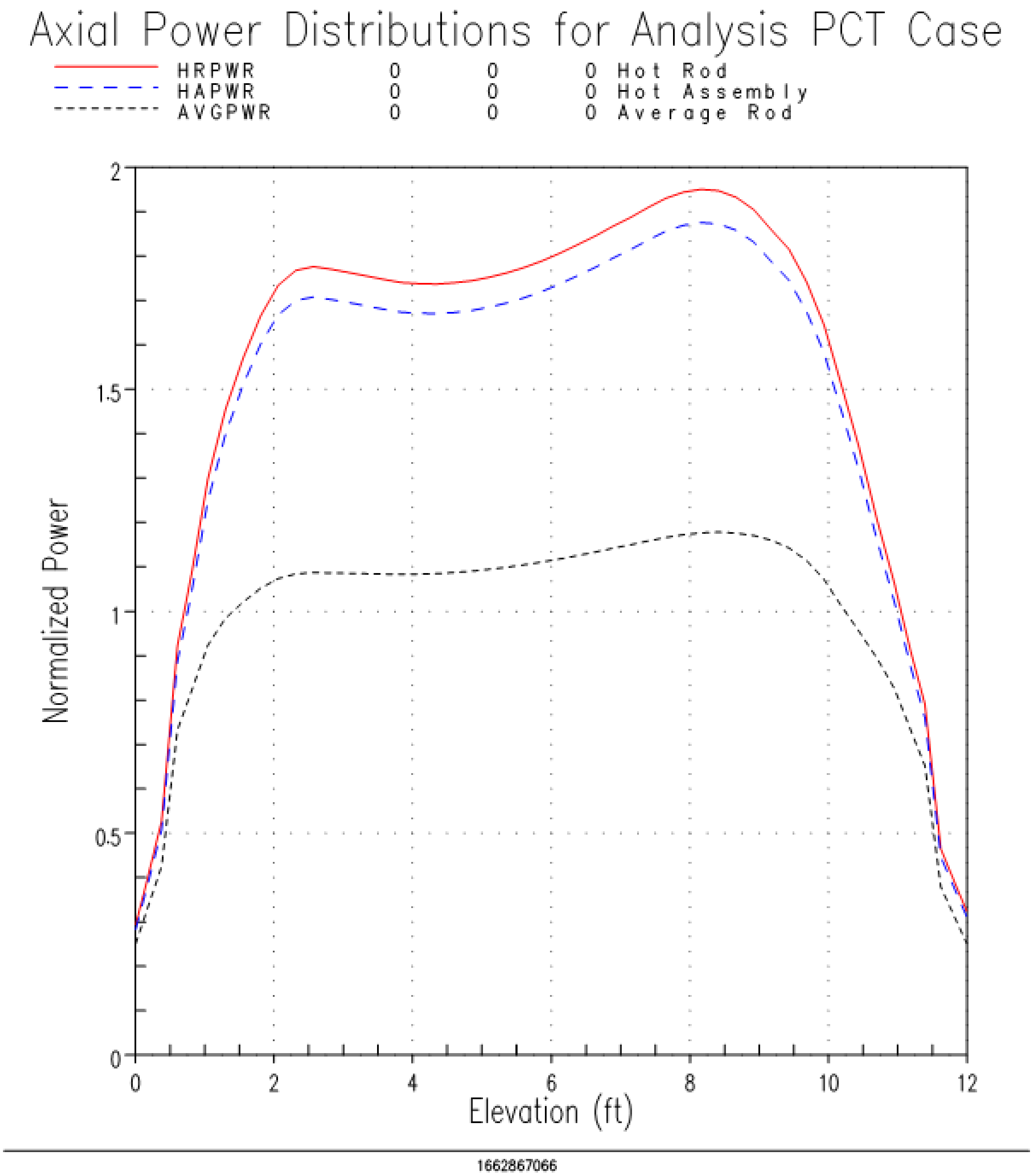

Figure 27a: Beaver Valley Unit 1 Normalized Core Power Shapes for the Region II Analysis PCT Case

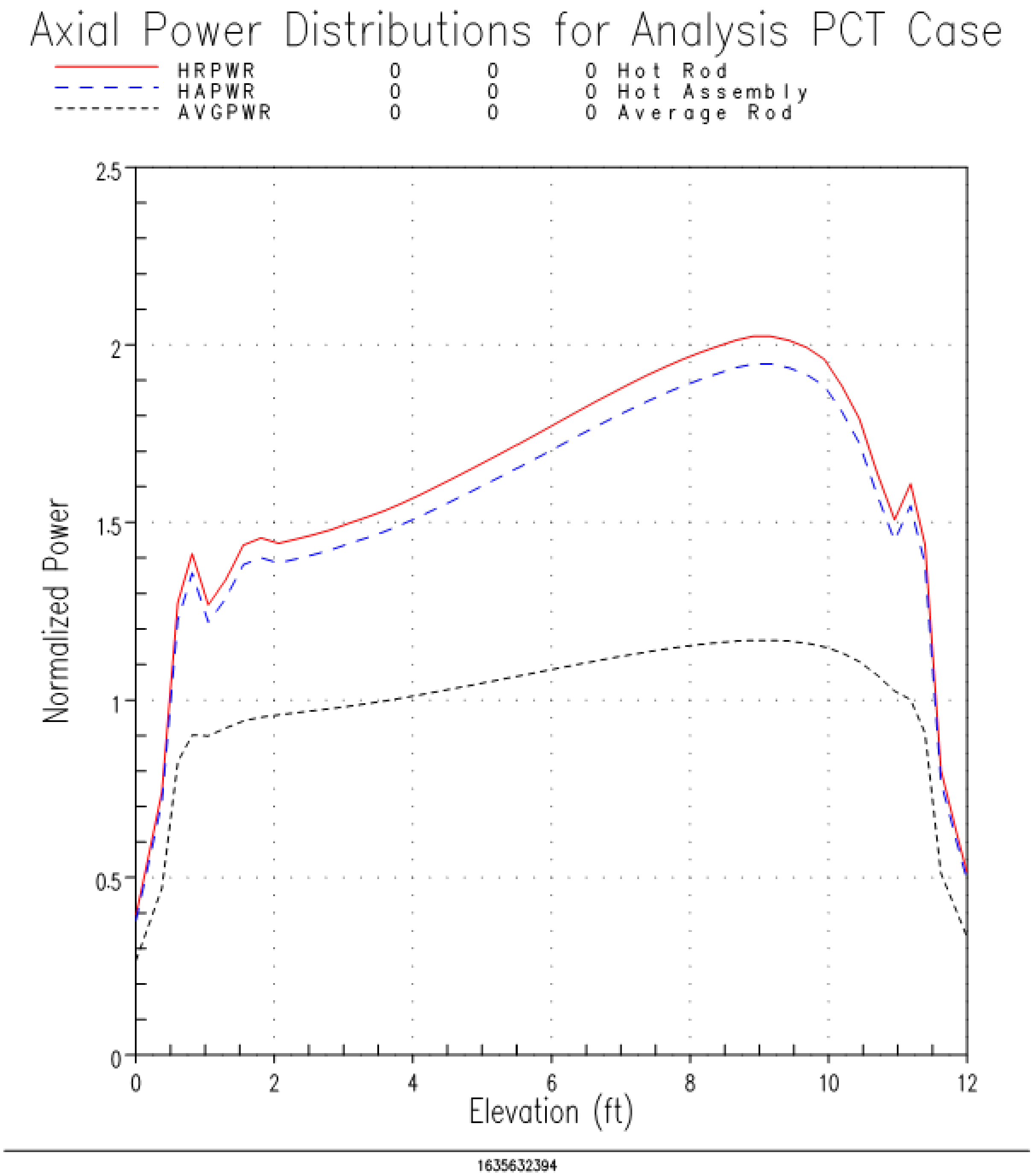

Figure 27b: Beaver Valley Unit 2 Normalized Core Power Shapes for the Region II Analysis PCT Case

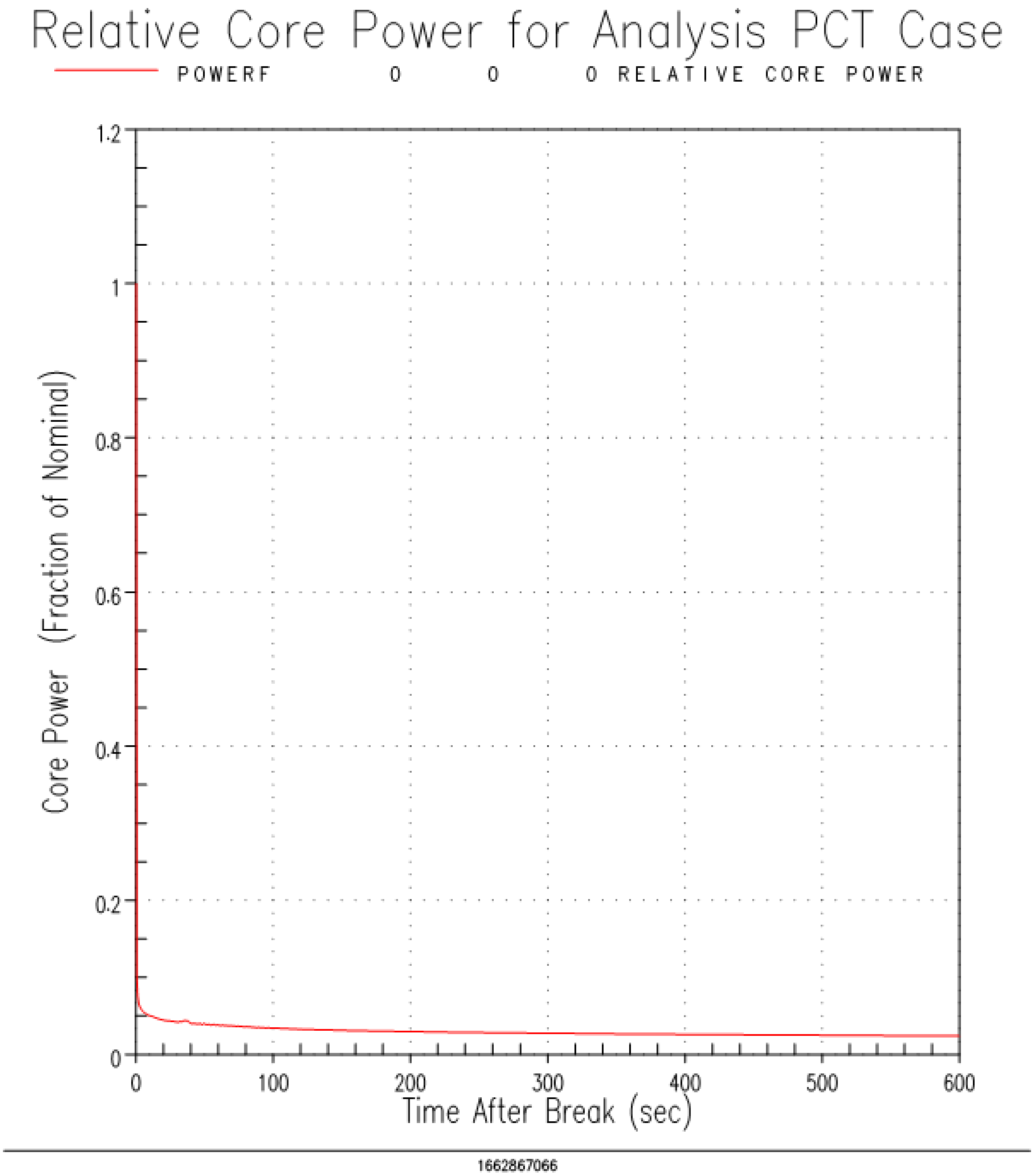

Figure 28a: Beaver Valley Unit 1 Relative Core Power for the Region II Analysis PCT Case

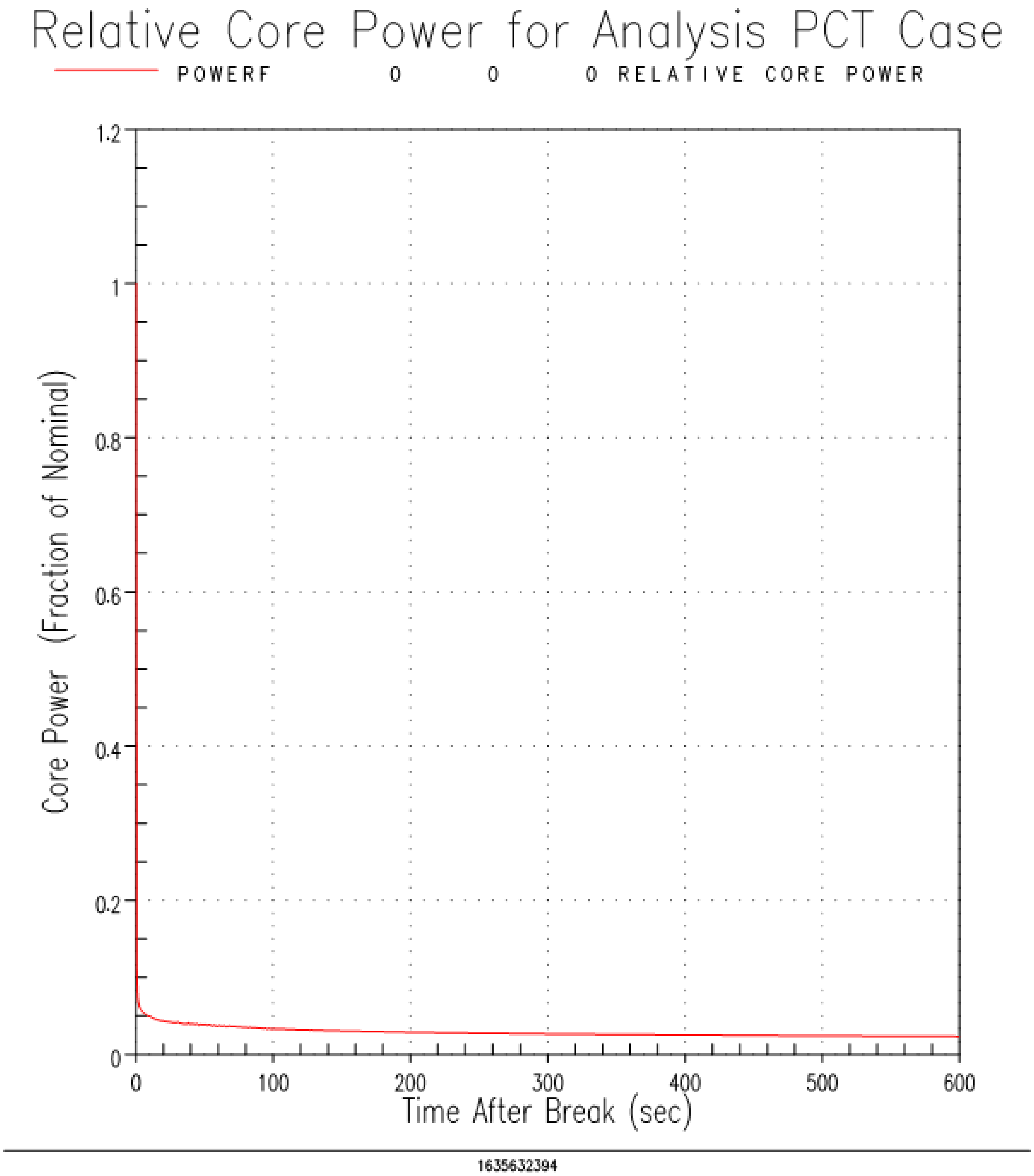

Figure 28b: Beaver Valley Unit 2 Relative Core Power for the Region II Analysis PCT Case

Enclosure E
L-21-238

Affidavit
(4 pages follow)

Attachment C: "Affidavit, CAW-22-031"

(Westinghouse Non-Proprietary Class 3 when removed from this letter)

(4 pages, including cover page)

Commonwealth of Pennsylvania:
County of Butler:

(1) I, Camille Zozula, Manager, Regulatory Compliance and Corporate Licensing, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of EH-22-013, Revision 0 be withheld from public disclosure under 10 CFR 2.390.

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

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

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

(ii) The information sought to be withheld is being transmitted to the Commission in confidence and, to Westinghouse's knowledge, is not available in public sources.

(iii) Westinghouse notes that a showing of substantial harm is no longer an applicable criterion for analyzing whether a document should be withheld from public disclosure. Nevertheless, public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

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

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

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

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

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

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

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

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

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

Executed on: 7/18/2022

Signed electronically by
Camille Zozula

Enclosure F
L-21-238

Attachment 1 of DLWM-LOCA-TM-LR-000001-NP, Revision 0, Responses to Potential RAIs on the Beaver Valley Units 1 and 2 Analysis with the Full Spectrum LOCA (FSLOCA) Methodology
(Non-Proprietary)
(17 pages follow)

Attachment A of EH-22-013

Attachment 1 to DLWM-LOCA-TM-LR-000001-NP, Revision 0, Responses to Potential RAIs on the Beaver Valley Units 1 and 2 Analyses with the FULL SPECTRUM LOCA (FSLOCA) Methodology

(17 pages, including cover page)

Attachment 1:

Responses to Potential RAIs on the Beaver Valley Units 1 and 2 Analyses with the FULL SPECTRUM LOCA (FSLOCA) Methodology

The following are the specific potential requests for additional information (RAIs) and responses:

1. Confirm that the analyses were performed with the updated code removing the errors discussed in LTR-NRC-18-30 and LTR-NRC-19-6. If not updated, provide the reasons and justify quantitatively the impact on the PCT, MLO, CWO, and containment backpressure results.

Response

The Beaver Valley Units 1 and 2 analyses with the **FULL SPECTRUM**™ Loss-of-Coolant Accident (**FSLOCA**™) Evaluation Model (EM) utilized a version of the WCOBRA/TRAC-TF2 code which incorporated the changes and error corrections described in LTR-NRC-18-30 but not those described in LTR-NRC-19-6. As described in the Limitation and Condition Number 2 discussion in Section 2.3 of the submitted License Amendment Request (LAR) "Application of Westinghouse FULL SPECTRUM LOCA Evaluation Model to the Beaver Valley Power Station Unit 1 and Unit 2," the analyses were performed with an updated code which removed the errors applicable to the FSLOCA EM that were reported in LTR-NRC-18-30. The errors applicable to the FSLOCA EM that were reported in LTR-NRC-19-6 are also discussed in the Limitation and Condition Number 2 discussion in Section 2.3 of the LAR. These errors were evaluated after completion of the analyses and found to have a negligible impact on the analysis results with the FSLOCA EM.

2. The Region II analyses do not provide the results for the break spectrum. Provide the break spectrum results (PCTs vs. break areas, MLO, CWO, etc.).

Response

Figures 1 and 2 show the peak cladding temperature (PCT) versus effective break area for the Beaver Valley Unit 1 analysis assuming loss-of-offsite power (LOOP) and offsite power available (OPA), respectively. Figures 7 and 8 show the PCT versus effective break area for the Beaver Valley Unit 2 analysis assuming LOOP and OPA, respectively. These figures reflect the combined effect of the break size and break flow model uncertainties.

Figures 3 and 4 show the transient maximum local oxidation (MLO) (or transient equivalent cladding reacted (ECR)) versus PCT for the Beaver Valley Unit 1 analysis assuming LOOP and OPA, respectively. Figures 9 and 10 show the transient ECR versus PCT for the Beaver Valley Unit 2 analysis assuming LOOP and OPA, respectively. A strong trend of increasing ECR with increasing PCT occurs due to the temperature dependence of the oxidation kinetics.

Figures 5 and 6 show the core-wide oxidation (CWO) versus PCT for the Beaver Valley Unit 1 analysis assuming LOOP and OPA, respectively. Figures 11 and 12 show the CWO versus PCT for the Beaver Valley Unit 2 analysis assuming LOOP and OPA, respectively. A strong trend of increasing CWO with increasing PCT occurs due to the temperature dependence of the oxidation kinetics.

The uncertainty analysis methodology used in the FSLOCA EM is described in Section 30 of WCAP-16996-P-A, Revision 1. A Monte Carlo sampling of all uncertainty contributors leads to the generation of a sample of simulated results from which upper tolerance limits are derived for the analysis figures of merit (PCT, MLO, CWO). **[**

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Figure 1: [ ]a,c

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Figure 2: [

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Figure 3: [ ]a,c

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Figure 4: [ ]a,c

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Figure 5: [ ]a,c

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Figure 8: [ ]a,c

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Figure 9: [ ]a,c

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Figure 10: [ ]a,c

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Figure 11: [

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Figure 12: [

]a,c

3. Considering both the current and the proposed LOCA analyses are best-estimate, differences were noted in some of the input parameters in the current analyses documented in the Updated Final Safety Analysis Report (UFSAR) and the proposed analyses as shown in the table below. Address the differences between the submittal and the UFSAR parameter values in the table below. Provide justification for the differences.

Unit	Parameter	FSLOCA EM LAR		UFSAR	
		Value	Table	Value	Table
Beaver Valley Unit 1	Steam generator tube plugging	≤ 15%	1a	≤ 22%	14.3.2-1
	Safety injection delay	≤ 22 s (OPA) ≤ 32 s (LOOP)	1a	≤ 17 s (OPA) ≤ 27 s (LOOP)	14.3.2-1
Beaver Valley Unit 2	Steam generator tube plugging	≤ 20%	1b	≤ 22%	15.6-8e
	Safety injection delay	≤ 22 s (OPA) ≤ 32 s (LOOP)	1b	≤ 17 s (OPA) ≤ 27 s (LOOP)	15.6-8e
	Safety injection temperature (T_{SI})	45°F ≤ T_{SI} ≤ 65°F	1b	45°F ≤ T_{SI} ≤ 105°F	15.6-8e
	Minimum initial containment pressure at full power operation	12.8 psia	2b	14.3 psia	15.6-8a
	Accumulator boron concentration	≥ 2300 ppm	1b	≥ 1900 ppm	15.6-8e
	Core power uncertainty	0.6%	1b	2.0%	15.6-8e

Response

The FSLOCA EM is a new best-estimate method that incorporates new conservatisms requiring a host of new inputs. As the question suggests, inputs were changed from the Beaver Valley Unit 1 ASTRUM analysis and the Beaver Valley Unit 2 CQD analysis. For example, the FSLOCA EM explicitly considers the effects of fuel pellet thermal conductivity degradation, incorporates updated fuel performance models and accounts for other burnup-related effects. Additionally, some inputs were modified from the values assumed in the ASTRUM and CQD analyses to improve operating margins or recover safety analysis margin. Other inputs were defined to maintain compliance with the new approved methodology. In short, even though the FSLOCA, ASTRUM, and CQD methodologies are all best-estimate methods, there are many differences between the methodologies. To accommodate these differences, the values were defined in accordance with the new FSLOCA methodology to ensure the analyses met the § 50.46(b) acceptance criteria.

The table provided comparing some of the input differences illustrates these different categories. For instance, the steam generator tube plugging input for both units was excessively high, so the operating margin was decreased to recover safety analysis margin. The minimum initial containment pressure change for Beaver Valley Unit 2 is an example of selecting inputs consistent with the approved FSLOCA methodology. Limitation and Condition 3 on the FSLOCA methodology states that for the purpose of calculating "a conservatively low, although not explicitly bounded, containment pressure, the input to the COCO model will be based on appropriate plant-specific containment design parameters and initial conditions." As such, an acceptable plant-specific initial containment pressure was provided to Westinghouse for the purpose of modeling the containment pressure response, consistent with the Limitation and Condition. A minimum value based on plant operating data was applied in the FSLOCA EM analysis. The safety injection delay times for both units were modified to address engineered safety feature

(ESF) response time margin issues associated with degraded grid voltage. The safety injection temperature range and minimum accumulator boron concentration for Beaver Valley Unit 2 were modified to match the Technical Specifications and to recover safety analysis margin. Lastly, the core power uncertainty for Beaver Valley Unit 2 was updated to reflect the high accuracy of the Leading Edge Flow Meters and to recover safety analysis margin.

4. Paragraph 50.46(b)(4) to 10 CFR on Coolable Geometry states that:

   > [c]alculated changes in core geometry shall be such that the core remains amenable to cooling.

   Section 5.0 of the submittal states:

   > Inboard grid deformation due to combined LOCA and seismic loads is not calculated to occur for Beaver Valley Unit 1 and Unit 2.

   Discuss why inboard grid deformation due to combined LOCA and seismic loads is not expected.

Response

The FSLOCA EM analysis does not affect the existing calculations that support the analysis of record related to combined LOCA and seismic loads, and the conclusion is retained from prior calculations and is credited in the current LOCA design basis analyses. That is, the previous calculations on grid deformation due to combined LOCA and seismic loads remain valid. As described in Section 14.3.2.6 of the Beaver Valley Unit 1 UFSAR, “The approved methodology (WCAP-12945-P-A) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the 44 assemblies in the low-power channel. This situation has not been calculated to occur for Beaver Valley Unit 1.” As described in Section 15.6.5.2.4 of the Beaver Valley Unit 2 UFSAR, “The BE methodology (WCAP-12945-P-A) specifies that the effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crush extends to in-board assemblies. Fuel assembly structural analyses performed for Beaver Valley Unit 2 indicate that this condition does not occur.”