VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

10CFR50.90

August 22, 2011

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Serial No. 11-299
NL&OS/ETS R1
Docket Nos. 50-338/339
License Nos. NPF-4/7

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
PROPOSED LICENSE AMENDMENT REQUEST (LAR)
ADDITION OF ANALYTICAL METHODOLOGY TO COLR
BEST-ESTIMATE LARGE BREAK LOSS OF COOLANT ACCIDENT (BE-LBLOCA)

In an October 21, 2010 letter (Serial No. 10-575), Dominion requested amendments, in the form of changes to the Technical Specifications (TS) to Facility Operating License Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. The proposed LAR requests the inclusion of the Westinghouse BE-LBLOCA analysis methodology using the Automated Statistical Treatment of Uncertainty Method (ASTRUM) for the analysis of LBLOCA to the list of methodologies approved for reference in the Core Operating Limits Report (COLR) in Technical Specification (TS), 5.6.5.b. This LAR also removes four obsolete COLR references that supported North Anna Improved Fuel (NAIF) product (i.e., Westinghouse Vantage 5H). The NAIF product is not planned to be used in future North Anna cores.

On May 18-19, 2011, the NRC audited Dominion and Westinghouse calculations related to the North Anna RFA-2 fuel transition. On May 24-26, 2011, Dominion and Westinghouse met with the NRC reviewers at the Westinghouse office in Rockville, MD to discuss their questions associated with the audit, as well as a pending request for additional information (RAI) on the BELOCA LAR. By letter dated June 21, 2011, the NRC formally requested additional information to complete their review of the proposed licensing actions. The response to the RAI is provided in Attachments 1 and 2 of this letter. The response to one additional BE-LBLOCA question from the audit is also incorporated into the RAI response.

Attachment 2 contains information proprietary to Westinghouse Electric Company LLC, and is supported by a Westinghouse Application for Withholding Proprietary Information for Public Disclosure and the accompanying Affidavit signed by Westinghouse, the owner of the information, and is provided in Attachment 4. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR 2.390. Correspondence with respect to the copyright or proprietary aspects of Attachment 2 or the

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supporting Westinghouse affidavit should reference letter CAW-11-3228 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066. A redacted (non-proprietary) version of Attachment 2 has been included as Attachment 3 for public disclosure.

The information provided in the attachments to this letter does not impact the conclusion of the significant hazards consideration determination as defined in 10 CFR 50.92 or the evaluation for eligibility for categorical exclusion as set forth in 10 CFR 51.22(c)(9).

Dominion is currently planning to use Westinghouse RFA-2 fuel in North Anna Units 1 and 2 commencing with North Anna Unit 1, Cycle 23 (Spring 2012) and North Anna Unit 2, Cycle 23 (Spring 2013). Therefore, Dominion requests approval of the proposed amendments by November 1, 2011. Dominion also requests a 60-day implementation period following NRC approval of the requested license amendments.

If you have any questions or require additional information, please contact Mr. Thomas Shaub at (804) 273-2763.

Sincerely,

J. Alan Price
Vice President – Nuclear Engineering

Attachments:
1. Response to Request for Additional Information – BELOCA License Amendment Request
2. Response to Request for Additional Information – BELOCA License Amendment Request (Proprietary)
3. Redacted Response of Attachment 2 – BELOCA License Amendment Request (Non-proprietary)
4. Westinghouse Affidavit

COMMONWEALTH OF VIRGINIA)

COUNTY OF HENRICO)

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2014

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price who is Vice President – Nuclear Engineering, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 22nd day of August, 2011.

My Commission Expires: May 31, 2014.

Vicki L. Hull
Notary Public

Commitments made in this letter: None

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ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
BELOCA LICENSE AMENDMENT REQUEST

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2

Response to Request for Additional Information
BELOCA License Amendment Request

Background

In an October 21, 2010 letter (Serial No. 10-575), Virginia Electric and Power Company (Dominion) requested amendments, in the form of changes to the Technical Specifications (TS) to Facility Operating License Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. The proposed LAR requests the inclusion of the Westinghouse BE-LBLOCA analysis methodology using the Automated Statistical Treatment of Uncertainty Method (ASTRUM) for the analysis of LBLOCA to the list of methodologies approved for reference in the Core Operating Limits Report (COLR) in Technical Specification (TS) 5.6.5.b. This LAR also removes four obsolete COLR references that supported North Anna Improved Fuel (NAIF) product (i.e., Westinghouse Vantage 5H). The NAIF product is not planned to be used in future North Anna cores.

As part of the amendment request, a BE-LBLOCA analysis using ASTRUM was completed for North Anna Units 1 and 2 and provided to the NRC for review and approval of the implementation of the Westinghouse BE-LBLOCA using ASTRUM for the North Anna analysis of a LBLOCA. The North Anna analysis was performed in compliance with the NRC conditions and limitations identified in WCAP-16009-P-A (Reference 1). The analysis employed a plant-specific adaptation of the ASTRUM evaluation model, consisting of increasing the number of circumferential noding stacks in the downcomer region from three to nine.

The Nuclear Regulatory Commission (NRC) staff has reviewed North Anna Power Station Unit Nos. 1 and 2 (NAPS) License Amendment Request regarding addition of analytical methodology to the core operating limits report for the best-estimate large break loss-of-coolant accident (LBLOCA) and has identified requests for additional information (RAI) as discussed below.

Request for Additional Information

AUDIT Question 1

Provide the results of the limiting large break LOCA analysis for the worst downcomer boiling case. Also show the results of this case with the 9 downcomer azimuthal cell nodalization. And, show the effect of time step size on PCT for the worst case downcomer boiling. (Per discussions during the audit, Mr. Len Ward would like to see the same plots that he requested for D. C. Cook Unit 2.)

Dominion Response

Response provided in Attachment 2

NRC Question 1

Provide a description and the results of the evaluation done against the conditions and limitations stated in the staff's safety evaluation (SE) on the Automated Statistical Treatment of Uncertainty Method (ASTRUM) in Westinghouse report WCAP-16009-P-A (Reference 2) with respect to the NAPS plant-specific adaptation of the ASTRUM methodology. Also, identify any deviations and their safety impact on the plant operations.

Dominion Response

The analysis is consistent with the NRC staff's ASTRUM SER Section 4.0 'Conditions and Limitations' including the cross-referenced historic SER Compliance related to the CQD approach given in Section 13.3 of the cited ASTRUM topical report. The plant specific adaptation for North Anna Units 1 and 2, as given in the final paragraph of LAR Section 4.1 'Method of Thermal Analysis for North Anna Units 1 and 2 is not in and of itself a deviation from the NRC staff's SER Section 4.0, however, it is viewed as an adaptation of the ASTRUM Methodology and is so characterized in that paragraph and the Dominion letter dated October 21, 2010 (Serial No. 10-575).

NRC Question 2

With respect to the analysis employing a plant-specific adaptation of the ASTRUM evaluation model, please provide the following:

(a) A clarification that a plant-specific adaption of the ASTRUM evaluation model is still within the approved limitations and conditions stated in the staff SE;

(b) The reason for increasing the number of circumferential noding stacks in the downcomer region from three to nine;

(c) The plant nodalization scheme for NAPS; and

(d) The results from the three circumferential node analyses.

Dominion Response

Response to Parts (a) and (c) are provided in Attachment 2.

(b) The nine circumferential node model for the North Anna Units 1 and 2 BE-LBLOCA analyses was selected to provide a technically acceptable model that had received NRC approval. Dominion's North Anna and the AEP's D. C. Cook units have similar plant characteristics (e.g., thermal shields and low containment backpressure) that make these plants susceptible to downcomer boiling effects during a postulated LBLOCA. As previously noted in the License Amendment Request dated October 21, 2010 (Reference 2-1), preliminary results for D.C. Cook Unit 1 with the as-approved ASTRUM method were observed to predict non-physical behaviors which were attributed to overly conservative aspects of the model. Consequently, an adaptation of ASTRUM was developed to better model the downcomer region by increasing the number of circumferential noding stacks by a power of three. In Reference 2-2, the NRC approved the D.C. Cook Unit 1 plant-specific application of the Westinghouse best estimate analysis using the increased downcomer noding. For North Anna Units 1 and 2, this increases the number of downcomer stacks modeled from three to nine.

(d) A three circumferential node model was not developed for North Anna. Therefore, there are no results to provide.

References

2-1 Letter from L. N. Hartz (Dominion) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed License Amendment Request (LAR), Addition of Analytical Methodology to COLR, Best-Estimate Large Break Loss of Coolant Accident (BE-LBLOCA)," October 21, 2010 (Serial No. 10-575). (ADAMS Accession No. ML102980447)

2-2 Letter from T. A. Beltz (USNRC) to M. W. Rencheck (Indiana Michigan Power Company), "Donald C. Cook Nuclear Plant Unit 1 – Issuance of Amendment to Renewed Facility Operating License Regarding Use of the Westinghouse ASTRUM Large Break Loss-of-Coolant Accident Analysis Methodology (TAC MD7556)," October 17, 2008. (ADAMS Accession Number ML082670351)

NRC Question 3

Not used

NRC Question 4

*Please describe the reason why higher peak cladding temperatures (PCTs) fall in the range of CD * AbreakIACL values between 1.0 and 2.0 on Figure 1 for Unit 1 and between 0.9 and 2.3 on Figure 16 for Unit 2. Also, clarify that the lower break size (around 0.8) for the split break case and the higher break size (around 2.2) for the double-ended guillotine break case yield a similar high PCT for Unit 2, while the high PCTs are dominated by double-ended guillotine break at an effective break size of 1.9 for Unit 1.*

Dominion Response

Response in provided in Attachment 2.

NRC Question 5

Please describe the physical meaning and cause with respect to a negative hot assembly vapor flow rate as shown in Figure 7 for Unit 1 and Figure 22 for Unit 2 between 7 and 30 seconds after the break.

Dominion Response

A notable aspect of blowdown is the core flow reversal behavior. The break is so large that the fluid quickly overcomes its initial steady state forward momentum near the cold leg break, and progressively at locations further from the break inclusive in the core starting at the bottom of the core and proceeding upward. The plant specific resistance network in the vessel (including the fuel assembly element) and the RCS loops is critical in determining the timing of the flow reversal. It is not uncommon for even the top of the core to eventually experience flow reversal. Figures 7 and 22 for Units 1 and 2, respectively, show the vapor flow rate at node 11 in the core, which corresponds to approximately 9' up the fuel assembly, and are fairly typical of flow reversal behavior in other 3-loop plants (though, as stated, the details are plant-specific). The hot assembly vapor flow rate plot is typically interesting as it often is a strong predictor of blowdown PCT trends since there is virtually no liquid present in the core shortly after the onset of blowdown, leaving vapor heat transfer as the dominant clad heat removal mechanism.

NRC Question 6

Provide the date of approval for the proposed Technical Specification (TS) 5.6.5.b.4 and identify which parameter listed in 5.6.5.a is supported by TS 5.6.5.b.4. Also, provide similar information for the rest of the methodologies listed in TS 5.6.5.b, such as its approval date and cycle-specific parameter it supported.

Dominion Response

The Westinghouse Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," WCAP-16009-P-A and WCAP-16009-NP-A (Non-Proprietary) was approved by the NRC on November 5, 2004. However, Dominion is requesting approval of a plant specific adaptation of WCAP-16009-P-A, which addresses a different nodalization scheme. This methodology will be used for TS 3.2.1 – Heat Flux Hot Channel Factor.

In a letter dated June 23, 2011 (Serial No. 11-349), Dominion provided the requested information for the methodologies listed in TS 5.3.5.b. This information was provided in response to a similar request for additional information for Dominion's license amendment request to include Appendix C of DOM-NAF-2-A, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code into Technical Specification 5.6.5.b as a referenced methodology.

NRC Question 7

Provide the results of the boric acid precipitation analysis (i.e., the analysis report) that supports the NAPS power level of 2951 megawatts thermal. The analysis should show the boric acid concentration versus time assuming no switch to simultaneous injection. The analysis should also list all of the key parameter inputs and assumptions applicable to the model used to identify the emergency operating procedure timing for switching to hot-leg injection.

Dominion Response

<u>Basis for Current Analysis</u>

The analysis of record for the post-LOCA hot leg recirculation switchover time was prepared to support Technical Specification changes to increase the boron concentration limits in the refueling water storage tank (RWST), casing cooling tank (CCT), safety injection (SI) accumulators, and spent fuel pool during refueling. The License Amendment Request (LAR) was submitted to the NRC in Reference 7-1 and approved by the NRC in Reference 7-2. Section 3.3.1 in Attachment 1 of Reference 7-1 describes the re-analysis for the post-LOCA hot leg recirculation time and a change to the cold-to-hot leg recirculation switchover time. The analysis requires the transfer to hot leg recirculation be completed within 5 hours of the design basis LOCA. The calculation explicitly includes a core power level of 2951 MWt to determine the core steaming rate (100.37% of the current Rated Thermal Power of 2940 MWt). The core power assumption of 2951 MWt was identified in the LAR for the North Anna Measurement Uncertainty Recapture (MUR) power uprate [Reference 7-3, Attachment 5, page 37], which was approved by the NRC in Reference 7-4.

During an audit of the North Anna transition to the Westinghouse RFA-2 fuel product on May 24-26, 2011, the NRC was provided the Dominion engineering calculation that was the basis for the Reference 7-1 LAR, three subsequent calculation addenda that document evaluations of minor changes to the plant that did not affect the hot leg switchover time, and the engineering evaluation that supported the Reference 7-3 LAR for the MUR power uprate. The following information is taken from the engineering calculation that was reviewed during the audit.

Calculation Inputs and Assumptions

The key inputs to the calculation are related to core power, water sources (volume and boron concentration) for the safety injection system during injection mode and containment sump recirculation, and the vessel mixing volume.

- The analysis assumes the contents of the reactor coolant system (RCS), RWST, SI accumulators, SI piping, boron injection tank (BIT), CCT, and chemical addition tank (CAT) are injected to the containment sump, mixed uniformly, and recirculated through the core. All of these sources contain boric acid solution except the CAT, which contains sodium hydroxide (NaOH) to maintain quench spray pH and containment sump pH within analyzed limits. Table 7-1 summarizes the volumes and boron concentrations for water sources that are controlled by Technical Specifications and supported by the analysis. In addition, the analysis includes 70,000 gallons of RCS liquid at 2000 ppm boron.
- The mass evaporation rate is based on a core power of 2951 MWt and the 1979 ANS decay heat standard with a 1.2 multiplier.
- An effective liquid mixing volume of 854 ft^3 is assumed. This was a standard assumption for initial analyses of Westinghouse 3-loop PWRs performed in the 1970s. North Anna analyses use this value which includes the volume from the top of the lower core plate to the upper plenum at the hot leg bottom elevation. While core void fraction and loop pressure drop are not modeled, the PWR Owners Group evaluation in Reference 7-5 documented that voiding effects and un-credited volumes (e.g., lower plenum) approximately offset each other. Another conservatism is that steam exiting the core is assumed to contain no boric acid (no credit for liquid entrainment), so all boron is deposited in the effective liquid mixing volume.

Calculation Results and Margins

The calculation determines the time after LOCA when the boric acid solution concentration reaches 23.5 weight percent (wt%). Figure 7-1 shows boric acid concentration versus time in the core, with 23.5 wt% reached at 5.26 hours. This figure does not display the reduction of boric acid concentration due to dilution from the sump water subsequent to hot leg switchover.

Reference 7-5 identified several margins that are not included in the North Anna analysis. Some recent licensee analyses use a higher boron solubility limit at atmospheric pressure. Using an atmospheric boric acid solution solubility limit of 29.27 wt% consistent with the Point Beach extended power uprate (Reference 7-6 and 7-7), Figure 7-1 shows 7.2 hours to reach the limit.

Also, Figure 7-1 does not credit the injection of NaOH as the containment sump pH buffer. Reference 7-5, Section B, and WCAP-17021-NP (Reference 7-8) indicate that the boron solubility limit is greater than 48 wt% and 40 wt%, respectively, for a ternary solution of water, boric acid, and NaOH undergoing boiling at saturation under atmospheric pressure conditions. Using a boron solubility limit that ignores NaOH represents a significant conservatism in the North Anna analysis basis.

Plant Procedures

The North Anna Emergency Operating Procedures (EOPs) are based on the Pressurized Water Reactors Owners Group (WOG) Emergency Response Guidelines (ERGs) for Westinghouse plants. During a large break LOCA, the EOPs direct the manual transfer of the discharge flow from the high head safety injection (HHSI) and low head safety injection (LHSI) pumps to the three RCS hot legs. The EOPs initiate this transfer 4.5 hours after LOCA initiation. This provides sufficient time to complete the manual actions within the requirements of the safety analysis, which determined 5.26 hours to reach a boric acid concentration of 23.5 wt%.

During a small break LOCA, the transfer to hot leg recirculation is entered if RCS subcooling criteria are not met, 4.5 hours have elapsed since event initiation, and SI flow is indicated.

Before 5 hours have elapsed following establishment of hot leg recirculation, the EOPs are used to transfer all SI flow from the hot legs to the cold legs. As long as the EOPs are used to respond to the event, the SI flow path is alternated between the cold legs and hot legs every 5 hours as discussed in North Anna Updated Final Safety Analysis Report Section 6.3.3.10 [Reference 7-9]. The method of using a complete transfer of SI flow from cold leg to hot leg recirculation to preclude post-LOCA boric acid precipitation was approved by the NRC during original North Anna licensing. Section 6.3.3 in NUREG-0053 [Reference 7-10] documents NRC approval of 18 hours to initiate hot leg recirculation, followed by alternating cold leg and hot leg recirculation every 27.5 hours. Currently, 5 hours is used as the alternating frequency between cold and hot leg recirculation, based on the calculation described above for the period starting at LOCA initiation, which is conservative for significantly lower decay heat later in the accident.

Evaluation of Flushing Flow

The transfer of all SI flow to the RCS hot legs provides adequate flushing flow to reduce the boron concentration in the core. The limiting case for minimum SI flow is a small break LOCA with RCS pressure above the LHSI pump shutoff head, such that flow is only delivered from the HHSI system. A single HHSI pump can deliver to the RCS hot

legs at least 55 lbm/sec at 1200 psia RCS pressure, which is an upper bound pressure for SBLOCAs that would not meet RCS subcooling criteria at hot leg switchover time. The decay heat removal requirement for 2951 MWt at 4.5 hours is 33 lbm/sec so there is at least 22 lbm/sec of flushing flow above the boil-off requirement.

Operators use the EOPs to depressurize the RCS below the LHSI pump shutoff head to increase the total injection flow and refill the RCS. As RCS pressure decreases, the HHSI injection flow increases. For LOCAs with the RCS depressurized fully, a single LHSI pump can inject at least 4000 gpm to the RCS hot legs and dilute the core boron concentration rapidly.

References

7-1 Letter from David A. Christian (Dominion) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specification Changes, Increased Boron Concentration," Serial No. 00-305, June 22, 2000. (ADAMS Accession No. ML003728735)

7-2 Letter from Stephen Monarque (NRC) to David A. Christian (Dominion), "North Anna Power Station, Units 1 and 2 - Issuance of Amendments RE: Technical Specifications Changes to Increase Boron Concentration Limits (TAC Nos. MA9362 and MA9363)," March 20, 2001. (ADAMS Accession No. ML010800310)

7-3 Letter from J. Alan Price (Dominion) to USNRC, "Virginia Electric and Power Company (Dominion), North Anna Power Station Units 1 and 2, License Amendment Request, Measurement Uncertainty Recapture Power Uprate," Serial No. 09-033, Rev. 0, March 26, 2009. (ADAMS Accession No. ML090900055)

7-4 Letter from V. Sreenivas (USNRC) to David A. Heacock (Dominion), "North Anna Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME0965 and ME0966)," October 22, 2009. (ADAMS Accession No. ML092250616)

7-5 Letter OG-06-200 from F. P. Schiffley III (PWR Owners Group) to D. S. Collins (USNRC), "Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, Post LOCA Long Term Cooling Model Due to Discovery of Non-Conservative Modeling Assumptions During Calculation Audit, PA-ASC-290," June 19, 2006. (ADAMS Accession No. ML061720175)

7-6 Letter from L. Meyer (FPL Energy Point Beach, LLC) to USNRC, "License Amendment Request 261, Extended Power Uprate Pursuant to 10 CFR 50.90," April 7, 2009. (ADAMS Accession No. ML091250564, ML091250566, and ML091250569)

7-7 Letter from T. A. Beltz (NRC) to L. Meyer (NextEra Energy Point Beach, LLC), "Point Beach Nuclear Plant (PBNP), Units 1 and 2 -Issuance of License Amendments Regarding Extended Power Uprate (TAC Nos. ME1044 AND ME1045)," May 3, 2011. (ADAMS Accession No. ML110880039 and ML110450159)

7-8 WCAP-17021-NP, "Summary of Tests to Determine the Physical Properties of Buffered and Un-buffered Boric Acid Solutions," March 2009; transmitted by letter OG-11-149 from M. L. Arey Jr. (PWR Owners Group) to USNRC, "For Information Only – WCAP-17021-NP Rev. 1, "Summary of Tests to Determine the Physical Properties of Buffered and Un-buffered Boric Acid Solutions", (PA-ASC-0264)," July 29, 2011.

7-9 North Anna Updated Final Safety Analysis Report, Revision 46.

7-10 NUREG-0053, Revision 0, "Safety Evaluation Report Related for Operation of North Anna Power Station Units 1 and 2, Virginia Electric and Power Company," June 1976.

Table 7-1

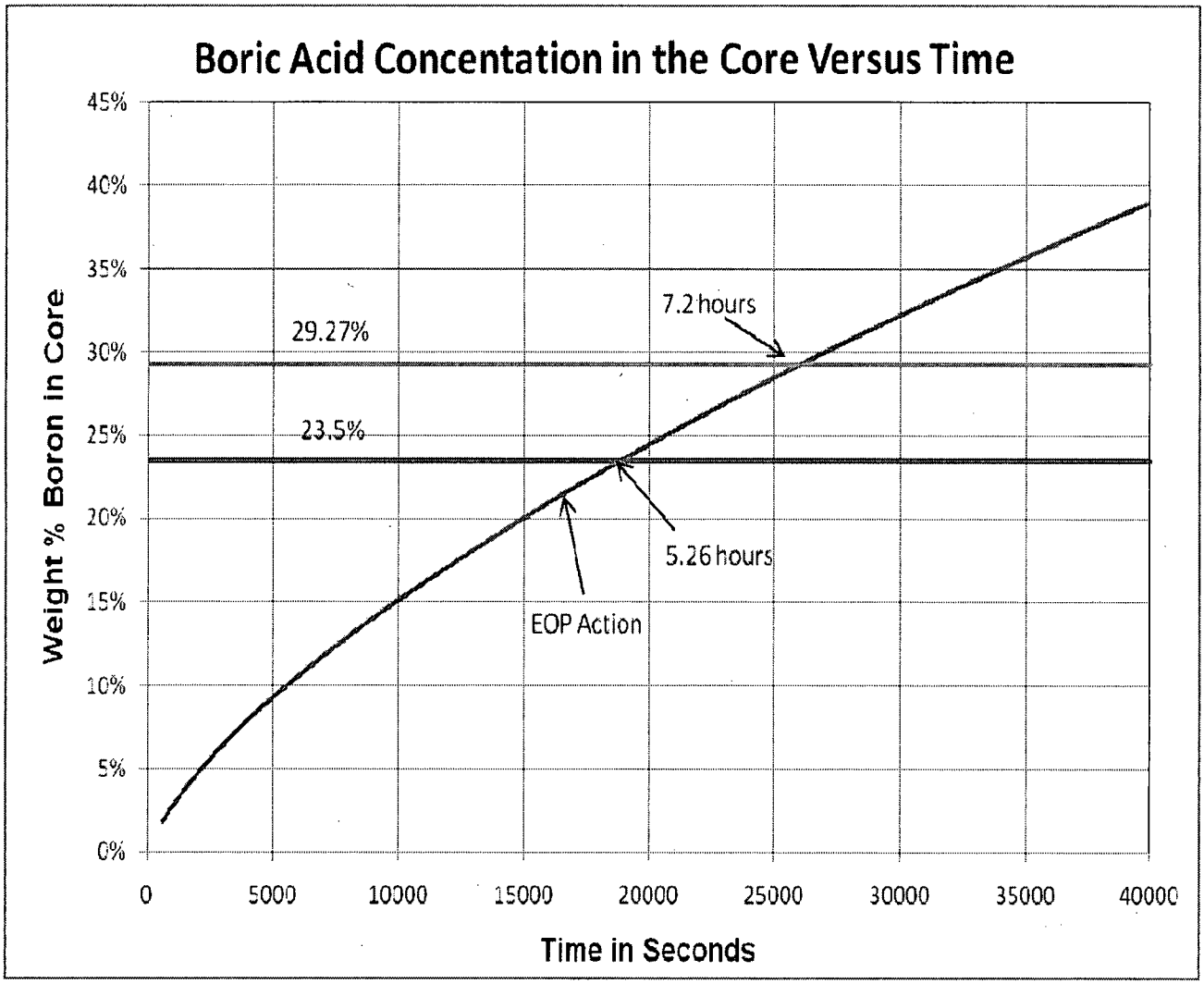
Water Sources for Post-LOCA Boric Acid Concentration Controlled by Technical Specifications

	T.S. Limits for Volume, gallons	T.S. Limits for Boron Concentration, ppm
RWST	466,200 – 487,000 (SR 3.5.4.2)	2600 – 2800 (SR 3.5.4.3)
Accumulator	7580 – 7756* (SR 3.5.1.2)	2500 – 2800 (SR 3.5.1.4)
BIT	≥ 900 (SR 3.5.6.2)	12950 – 15750 (SR 3.5.6.3)
CCT	≥ 116,500 (SR 3.6.7.2)	2600 – 2800 (SR 3.6.7.3)
CAT	4800 – 5500 (SR 3.6.8.2)	N/A

* per accumulator

Figure 7-1

Boric Acid Concentration Versus Time



NRC Question 8

Provide the following information regarding the NAPS nuclear steam supply system:

(a) Volume of the lower plenum, core and upper plenum below the bottom elevation of the hot leg, each identified separately. Also provide heights of these regions.

(b) Loop friction and geometry pressure losses from the core exit through the steam generators (SGs) to the inlet nozzle of the reactor vessel. Also, provide the locked rotor reactor coolant pump (RCP) k-factor. Provide the mass flow rates, flow areas, k-factors, and coolant temperatures for the pressure losses (upper plenum, hot legs, SGs, suction legs, RCPs, and discharge legs). Include the reduced SG flow areas due to plugged tubes. Provide the loss from each of the intact cold legs through the annulus to a single broken cold leg. Also, provide the equivalent loop resistance for the broken loop and separately for the intact loop.

(c) Capacity and boron concentration of the refueling water storage pool

(d) Capacity of the condensate storage tank

(e) Flushing flow rate at the time of the switch to simultaneous injection

(f) High pressure safety injection runout flow rate

(g) Capacities and boron concentrations for boron injection tank (BIT) storage tanks

(h) Flow rate into the reactor coolant system from the BIT

Dominion Response

Response to Parts (a) and (b) are provided in Attachment 2.

(c) The capacity and boron concentration of the refueling water storage tank (RWST) is provided in Table 7-1 in the response for RAI #7. It is noted that the temperature of the RWST is maintained between 40°F and 50°F by the Technical Specifications.

(d) Each North Anna unit has 110,000 gallons available in the Emergency Condensate Storage Tank and a 300,000 gallon capacity in the normal Condensate Storage Tank. In addition, the service water reservoir and Lake Anna can be used to supply water to the steam generators to support plant cooldown requirements. This basis is consistent with the NRC Safety Evaluation Report documented in Reference 8-1 for North Anna's commitments to Generic Letter 81-21 regarding condensate water sources available for cooldown.

(e) During the NRC Audit on May 24-26, 2011, the NRC noted the use of simultaneous injection by the high head safety injection pumps by some plants to get the core flushed during the small break LOCA. The NRC is concerned that this could lead to pump runout.

North Anna does not use simultaneous injection to the RCS cold legs and hot legs. The EOPs direct the operators to switch from cold leg injection to hot leg injection at 4.5 hours as discussed in the response to RAI #7. North Anna performs a complete swap of the safety injection flow from cold leg injection to hot leg injection. After another 5 hours, the EOPs direct the operators to switch from hot leg injection to cold leg injection. Again, North Anna performs a complete swap of the injection flow and maintains a configuration such that the HHSI pumps do not reach a runout condition.

The flushing flow rate at the time of the switch to hot leg injection at 4.5 hours is discussed in the response to RAI #7.

(f) During the NRC Audit on May 24-26, 2011, the NRC noted the use of simultaneous injection by the high head safety injection pumps by some plants to flush the core. The NRC is concerned that this lineup could lead to HHSI pump runout. Dominion stated that simultaneous injection is not employed at North Anna for boric acid precipitation control.

Testing has been performed to validate the system resistance for the HHSI system and the strength of the HHSI pump, to ensure that minimum delivered HHSI flows will be met or exceeded during all postulated LBLOCA and SBLOCA scenarios. This acceptance criteria also ensure that the maximum continuous runout flow limit of 675 gpm for each HHSI pump will not be violated when the SI system is in either injection or recirculation mode.

(g) Each North Anna unit has one BIT. The capacities and boron concentrations for the boron injection tank (BIT) storage tank are provided in Table 7-1 in the response for RAI #7.

(h) The HHSI pump discharge is directed into the RCS cold legs via the BIT. During SI injection phase, the injection fluid is a mixture of RWST water at 50°F and BIT water at 115-140°F until the 900 gallon BIT is purged.

Table 8-1 provides both the injected HHSI flows and the spilling HHSI flows (which are actually injected into the broken-loop cold leg) for the faulted loop spilling to RCS pressure. These flow rates are used for SBLOCA analysis.

Table 8-2 provides both the injected HHSI and LHSI flows and the spilling HHSI and LHSI flows (which spill into containment) for the faulted loop spilling to containment pressure (0 psig). These flow rates are used for LBLOCA analysis.

Reference

8-1 Letter from Steven A. Varga (USNRC) to W. L. Stewart (Virginia Power), October 19, 1983.

Table 8-1
HHSI Flows with the Faulted Loop Spilling to RCS Pressure

Pressure (psia)	Spilled Flow (gpm)	Injected Flow (gpm)	Spilled Flow (lbm/s)	Injected Flow (lbm/s)
14.7	174.5	323.4	24.22	44.88
64.7	172.7	320.1	23.97	44.43
114.7	170.9	316.8	23.72	43.97
264.7	171.8	318.5	23.84	44.20
514.7	162.4	300.5	22.54	41.71
764.7	152.5	281.7	21.17	39.10
1014.7	134.2	246.9	18.63	34.27
1264.7	122.7	225.3	17.03	31.27
1514.7	109.7	201.7	15.23	27.99
1764.7	96.2	176.3	13.35	24.47
2014.7	81.4	149.1	11.30	20.69
2114.7	73.7	134.9	10.23	18.72
2114.8	0.0	0.0	0.0	0.0
3000.0	0.0	0.0	0.0	0.0

Table 8-2

HHSI and LHSI Flows with the Faulted Loop Spilling to Containment Pressure (0 psig)

Pressure	HHSI				LHSI				Combined	
(psia)	Spilled Flow (gpm)	Injected Flow (gpm)	Spilled Flow (lbm/s)	Injected Flow (lbm/s)	Spilled Flow (gpm)	Injected Flow (gpm)	Spilled Flow (lbm/s)	Injected Flow (lbm/s)	Spilled Flow (lbm/s)	Injected Flow (lbm/s)
14.7	176.5	317.5	24.5	44.1	1065.0	2207.0	147.8	306.3	172.3	350.4
34.7	177.0	315.7	24.6	43.8	1195.0	1926.0	165.9	267.3	190.4	311.1
64.7	177.7	313.1	24.7	43.5	1385.0	1468.0	192.2	203.7	216.9	247.2
84.7	178.2	311.3	24.7	43.2	1514.0	1119.0	210.1	155.3	234.9	198.5
94.7	178.4	310.3	24.8	43.1	1579.0	932.0	219.1	129.4	243.9	172.4
114.7	178.9	308.5	24.8	42.8	1714.0	513.0	237.9	71.2	262.7	114.0
124.7	179.6	308.3	24.9	42.8	1785.0	268.0	247.7	37.2	272.7	80.0
124.8	179.6	308.3	24.9	42.8	1785.7	0.0	247.8	0.0	272.8	42.8
134.7	180.4	308.1	25.0	42.8	1856.0	0.0	257.6	0.0	282.6	42.8
264.7	189.9	305.3	26.4	42.4	1856.0	0.0	257.6	0.0	283.9	42.4
514.7	198.3	278.9	27.5	38.7	1856.0	0.0	257.6	0.0	285.1	38.7
764.7	206.8	251.0	28.7	34.8	1856.0	0.0	257.6	0.0	286.3	34.8
1014.7	210.1	204.4	29.2	28.4	1856.0	0.0	257.6	0.0	286.8	28.4
1264.7	219.2	170.7	30.4	23.7	1856.0	0.0	257.6	0.0	288.0	23.7
1514.7	228.7	133.5	31.7	18.5	1856.0	0.0	257.6	0.0	289.3	18.5
1764.7	243.8	90.1	33.8	12.5	1856.0	0.0	257.6	0.0	291.4	12.5
2014.7	249.2	38.1	34.6	5.3	1856.0	0.0	257.6	0.0	292.2	5.3
2114.7	254.0	12.2	35.3	1.7	1856.0	0.0	257.6	0.0	292.8	1.7
2114.8	254.0	0.0	35.3	0.0	1856.0	0.0	257.6	0.0	292.8	0.0
3000.0	254.0	0.0	35.3	0.0	1856.0	0.0	257.6	0.0	292.8	0.0

NRC Question 9

Provide the following elevation data.

(a) Bottom elevation of the suction leg horizontal leg piping and cold leg diameter

(b) Top elevation of the cold leg at the RCP discharge

(b) Top elevation of the core (also height of core)

(d) Bottom elevation of the downcomer

Dominion Response

Response provided in Attachment 2.

NRC Question 10

Provide the limiting bottom and top skewed axial power shapes.

Dominion Response

The limiting BELOCA bottom skewed power shape is shown in Figure 10-1 with an axial offset of -29.468%. The limiting BELOCA top skewed power shape is shown in Figure 10-2 with an axial offset of 24.899%.

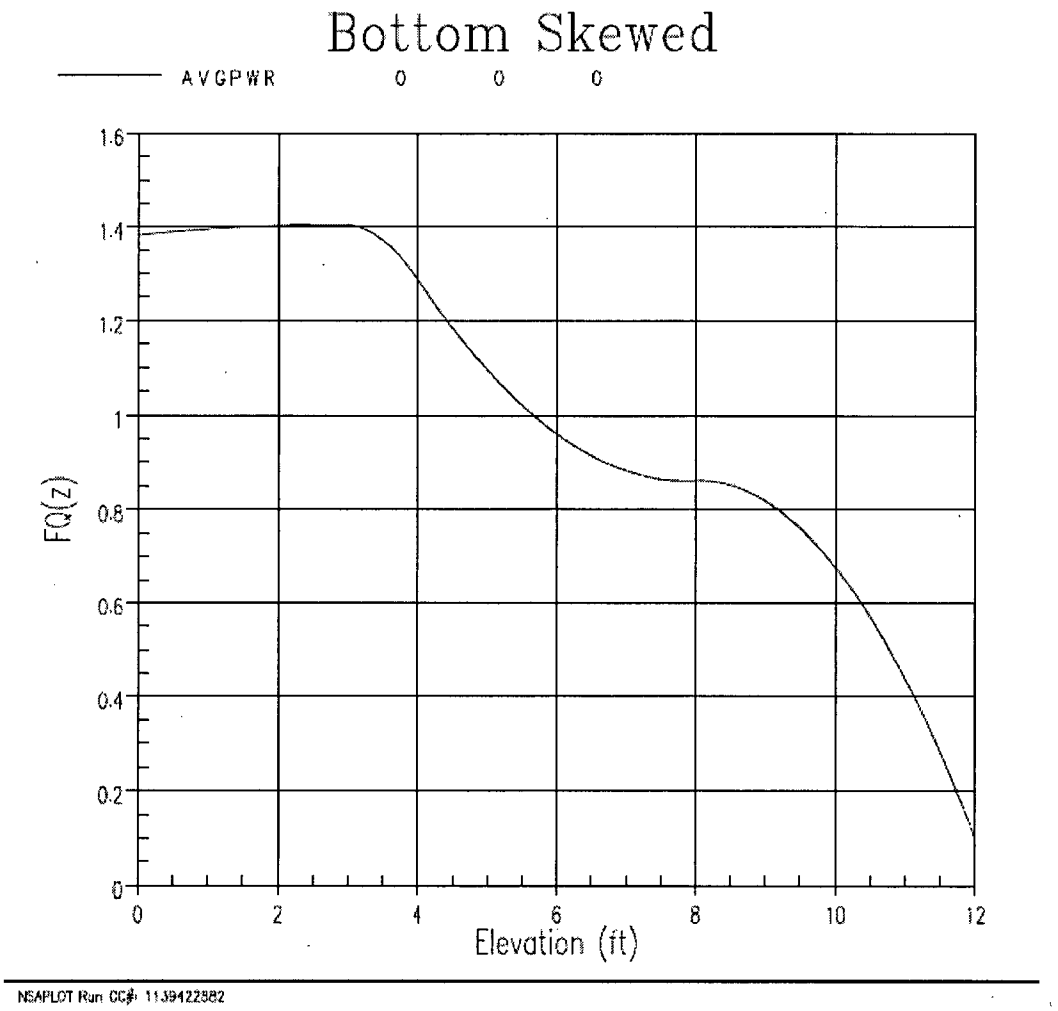


Figure 10-1 Limiting Bottom Skewed Power Shape for Average Power Rod

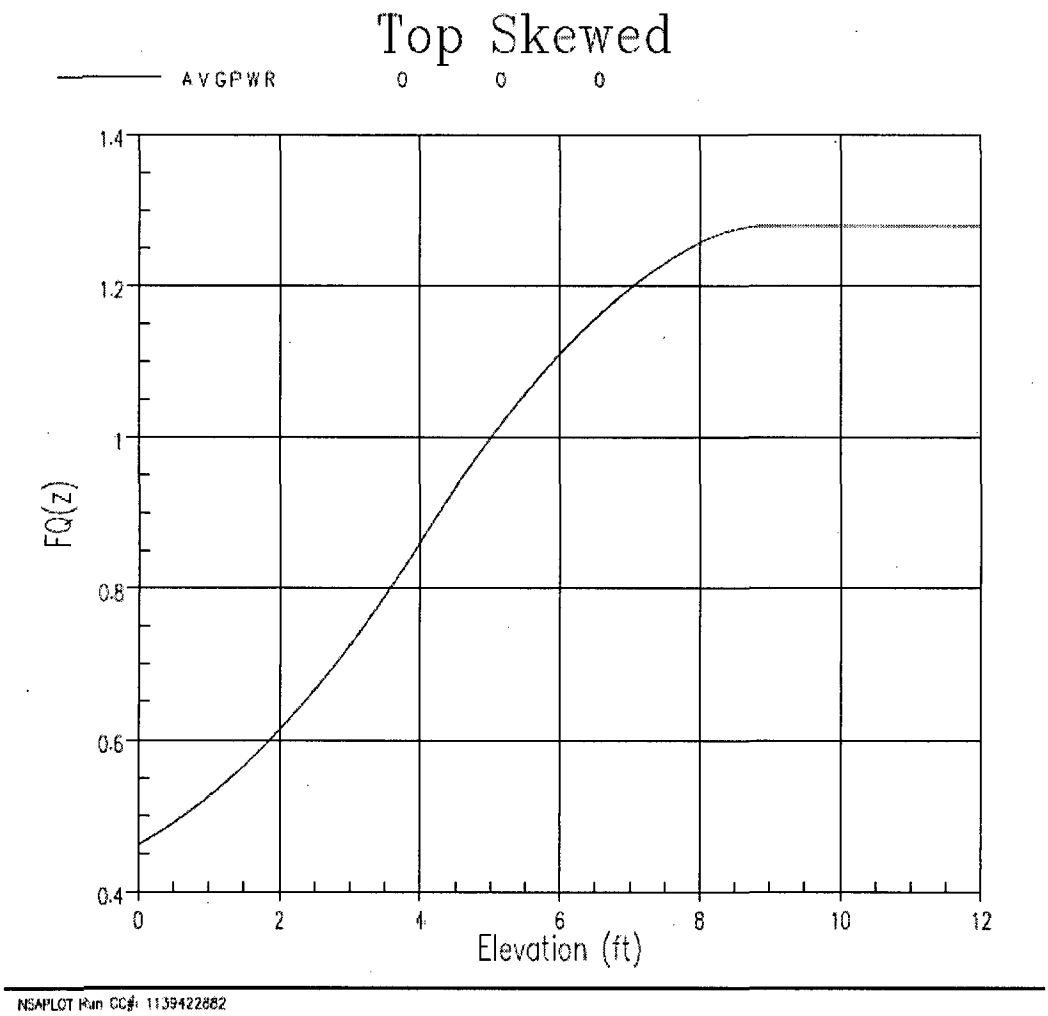


Figure 10-2 Limiting Top Skewed Power Shape for Average Power Rod

NRC Question 11

Discuss whether the Idlechik Handbook recommended expression for pressure loss coefficients along a curved channel was used. If so, explain why it was not used in the calculation for the k-factor. Also, provide the values of the lateral k-factors used for the downcomer lateral flow paths for the plant.

Dominion Response

Response provided in Attachment 2.

NRC Question 12

Provide the method used to compute the azimuthal lateral k-factors and the values used in the plant calculations. The staff notes that the "Idlechik" reference for calculating k-factors presents a method to compute k-factors in annuli of various radii. Please provide the results of a k-factor study for the lateral flow paths in the downcomer if it was performed.

Dominion Response

Response provided in Attachment 2.

NRC Question 13

Describe the azimuthal nodalization and results from the approved best estimate WCOBRA/TRAC model. Provide the results of other nodalization studies applied to the azimuthal detail in the downcomer (other than the three and nine azimuthal node studies). Also show the impact of time step on the PCT for the worst case downcomer boiling calculation.

Dominion Response

Response provided in Attachment 2.

NRC Question 14

The NRC staff completed its sensitivity study on downcomer boiling and the effect of lateral k-factor on this phenomenon. The case with zero lateral k-factor in the downcomer cross flow paths joining the azimuthal cells resulted in a 400 degrees F

reduction in PCT. This was due to the maximization of mixing between the downcomer azimuthal cells, which severely limited downcomer boiling. The cold water entering the downcomer during the long term readily mixed into the adjacent downcomer volumes and reduced boiling and the resulting core uncovery and clad temperature. Emergency core coolant bypass and liquid sweep-out that dominate the very early portion of the event (the first 100-200 seconds) does not prevail during the longer term when the downcomer fills with liquid and vapor velocities are no longer high enough to entrain and sweep out the injected liquid. Provide a detailed analysis of impact of the lateral k-factor values on PCT during downcomer boiling following an LBLOCA.

Dominion Response

Response provided in Attachment 2.

NRC Question 15

Note that the staff will review the results of the applicable small break LOCA (SBLOCA) break spectrum analysis for NAPS in a forthcoming audit activity. This will include the analysis supporting RCP trip timing, which supports the emergency operating procedure for tripping these pumps following a SBLOCA and a description of the methods and identification of the break sizes and limiting location and other pertinent assumptions supporting the RCP trip timing for North Anna.

Dominion Response

Response provided in Attachment 2.

NRC Question 16

Provide the decay heat multiplier of the limiting LBLOCA and a detailed description of how decay heat is sampled for each LBLOCA that was analyzed.

Dominion Response

Response provided in Attachment 2.

ATTACHMENT 3

NON-PROPRIETARY

REDACTED RESPONSE OF ATTACHMENT 2
BELOCA LICENSE AMENDMENT REQUEST

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2

North Anna Nuclear Power Station Units 1 and 2 – Response to the Request for Additional Information (RAI) from the Nuclear Regulatory Commission (NRC) Related to the May 2011 Audit of the Fuel Transition Project and the Best-Estimate Large Break LOCA (BE LBLOCA) License Amendment Request (LAR) No. 10-575 (TAC Nos. ME4933 and ME4934)

(Non-Proprietary)

August 2011

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NON-PROPRIETARY VERSION OF FINAL RAI RESPONSES:

The non-proprietary version of Westinghouse led Audit response 1 and RAI responses 2a, 2c, 4, 8a, 8b, 9, 11, 12, 13, 14, 15, and 16 containing proprietary information is provided in the following sections. Each section lists the RAI question and corresponding Westinghouse response:

AUDIT Question 1

Provide the results of the limiting large break LOCA analysis for the worst downcomer boiling case. Also show the results of this case with the 9 downcomer azimuthal cell nodalization. And, show the effect of time step size on PCT for the worst case downcomer boiling. (Per discussions during the audit, Mr. Len Ward would like to see the same plots that he requested for D. C. Cook Unit 2.)

RESPONSE

Four cases that showed significant downcomer boiling, as determined by a significant drop in lower-plenum collapsed liquid level after initial refill, were identified for each of the North Anna units. The Peak Cladding Temperature (PCT) and Local Maximum Oxidation (LMO) results for these cases, as compared to the PCT Limiting Case for each unit, are shown in Table 1-1 (North Anna Unit 1) and Table 1-2 (North Anna Unit 2). Plots comparing the lower plenum collapsed liquid levels and PCTs for each downcomer boiling case versus the appropriate limiting PCT case are shown in Figures 1-1 through 1-16. Additional plots are provided further showing the trends between the downcomer boiling case with the highest PCT (Unit 2 Run 080, PCT = 1410°F) and the Unit 2 PCT Limiting Case (Run 066). Figure 1-17 shows the vessel fluid mass trends for these runs. Figures 1-18 through 1-22 show the subcooling throughout various levels in the downcomer. Note that these results are from the 9-downcomer channel model calculations, since no calculations/runs were performed with the 3-downcomer channel model for North Anna Units 1 and 2.

Table 1-1: Significant Downcomer Boiling Cases for North Anna Unit 1

Run #	PCT (°F)	LMO (%)
114 (PCT Limiting Case)	1852	2.01
087	1151	0.01
055	1065	0.01
122	1288	0.08
054	1092	0.01

Table 1-2: Significant Downcomer Boiling Cases for North Anna Unit 2

Run #	PCT (°F)	LMO (%)
066 (PCT Limiting Case)	1871	3.53
114	1064	0.01
080	1410	0.20
060	1208	0.05
008	1147	0.01

The plant-specific effect of time step size on PCT for the worst-case downcomer boiling is not available, since no plant-specific time step size study was performed for the North Anna ASTRUM analyses (see

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below more detail). Note however, that the time step sizes used in the North Anna ASTRUM Units 1 and 2 analyses meet the Best-Estimate Large Break LOCA Evaluation Model requirements provided in Table 22-5-4 of Volume 4 of the CQD (WCAP-12945-P-A).

Generically, the effects of time step sizes are quantified in detail in WCAP-12945-P-A for both experimental facility and plant modeling applications as follows (Note that time step size, as referred to in this response, refers to DTMAX, the maximum allowable time step as defined by the user. The time step actually employed by the code is automatically adjusted to a value between DTMAX and DTMIN according to other convergence criteria as discussed in Section 2-7 of WCAP-12945-P-A):

- Section 2-7 and its subsections provide a detailed description of the methods for convergence criteria, time step size control, and numerical stability.

- Section 19-1-2 examines the effects of time step size on the experimental simulations used for code validation including the G-1 Blowdown Test 152, two FLECHT-SEASET forced reflood tests, CCTF Run 62, and LOFT L2-5. As stated on Page 19-1-6, [

  ]^{a,c}

- Section 22-5 presents the effects of time step size on the demonstration analyses for three PWR plants, with results presented in Table 22-5-3. Based on time step sensitivity observed during the blowdown, refill, and reflood periods, th^{c} ^{t}ime step strategy shown in Table 22-5-4 was selected.

- The response to RAI4-11 explains the procedure followed in Section 22-5 to arrive at a fixed range of allowable time step sizes (as shown in Table 22-5-4) in the interest of minimizing variability.

- The response to RAI4-50a explains the definition of convergence in terms of the magnitude of change in PCT with further reduction in time step. The scoping studies of Section 22-5 are repeated using the MOD7A version of the code for the North Anna plant as documented in Section 22-5-5. Again, sufficiently small variation (less than the uncertainty in the code bias) is observed when using time steps within the recommended range.

In summary, the "code resolution" in terms of PCT variability with time step changes has been shown to be of low significance relative to the total calculation uncertainty. This conclusion has been reached in a general sense by exercising the code for both experimental facilities and full-scale PWR simulations of multiple types (3- and 4-loop), and is therefore applicable to plant analysis applications such as North Anna Units 1 and 2.

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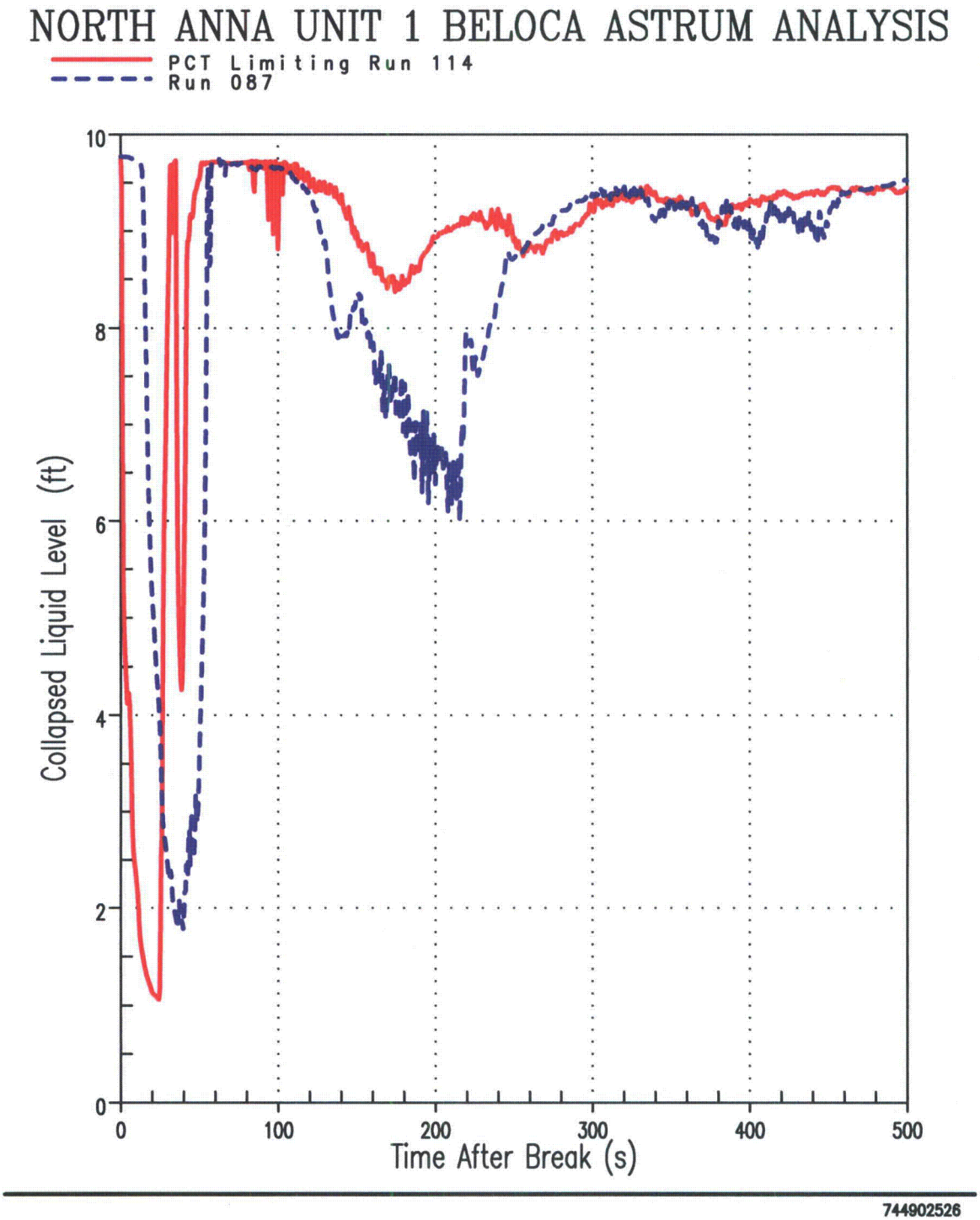

Figure 1-1: North Anna Unit 1 – Lower Plenum Collapsed Liquid Level for Limiting PCT Run 114 vs. Run 087

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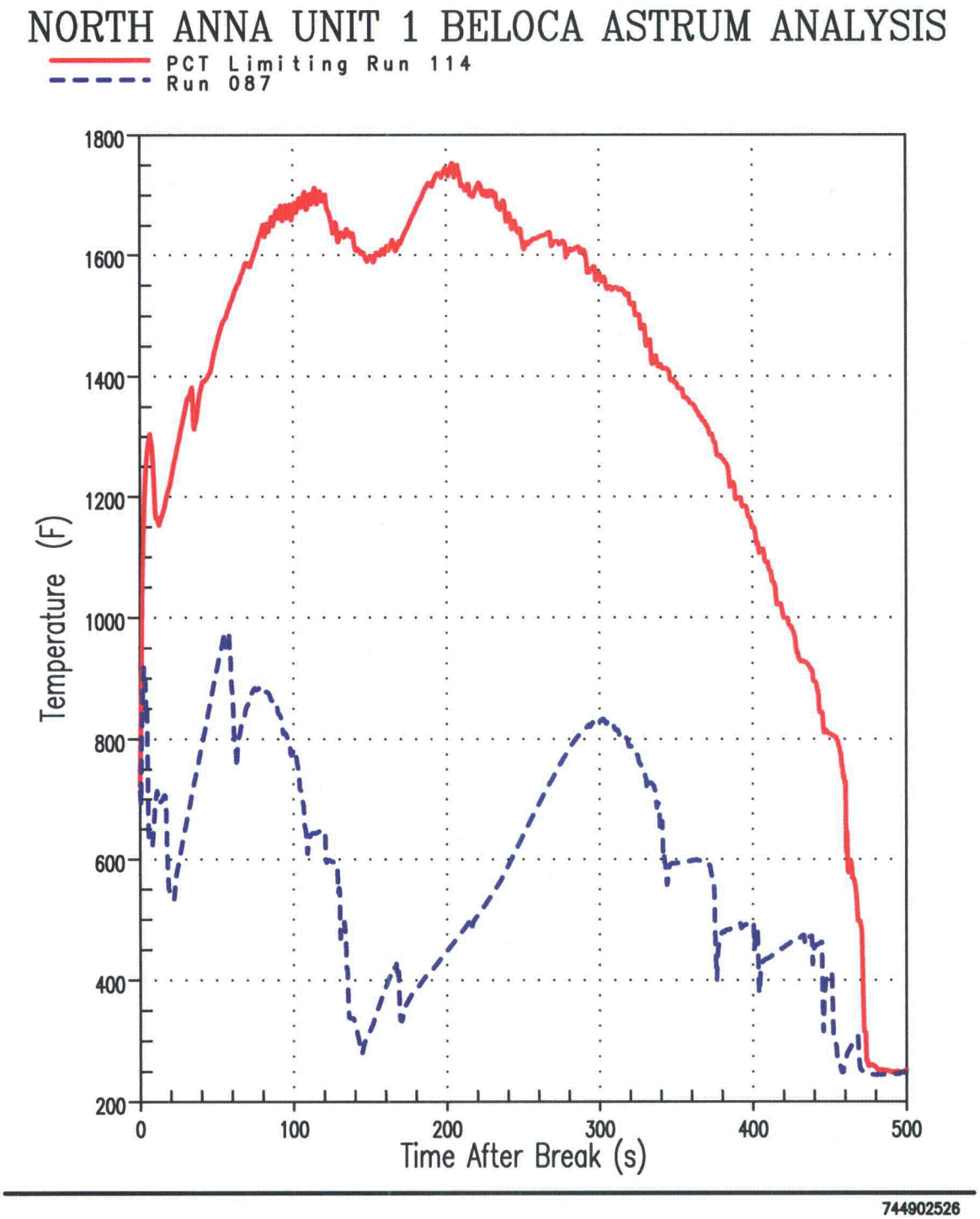

Figure 1-2: North Anna Unit 1 – Hot Rod PCT for Limiting PCT Run 114 vs. Run 087

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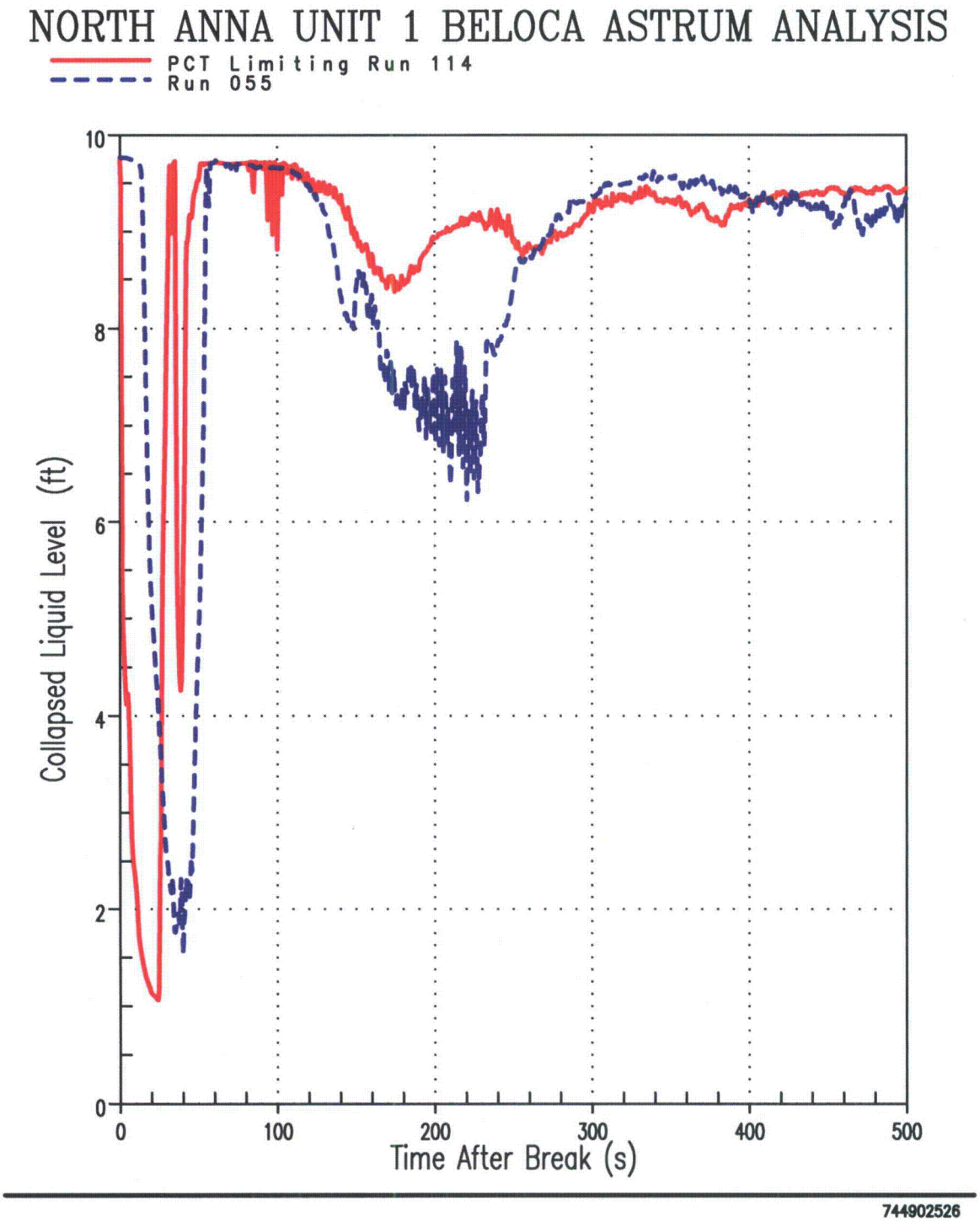

Figure 1-3: North Anna Unit 1 – Lower Plenum Collapsed Liquid Level for Limiting PCT Run 114 vs. Run 055

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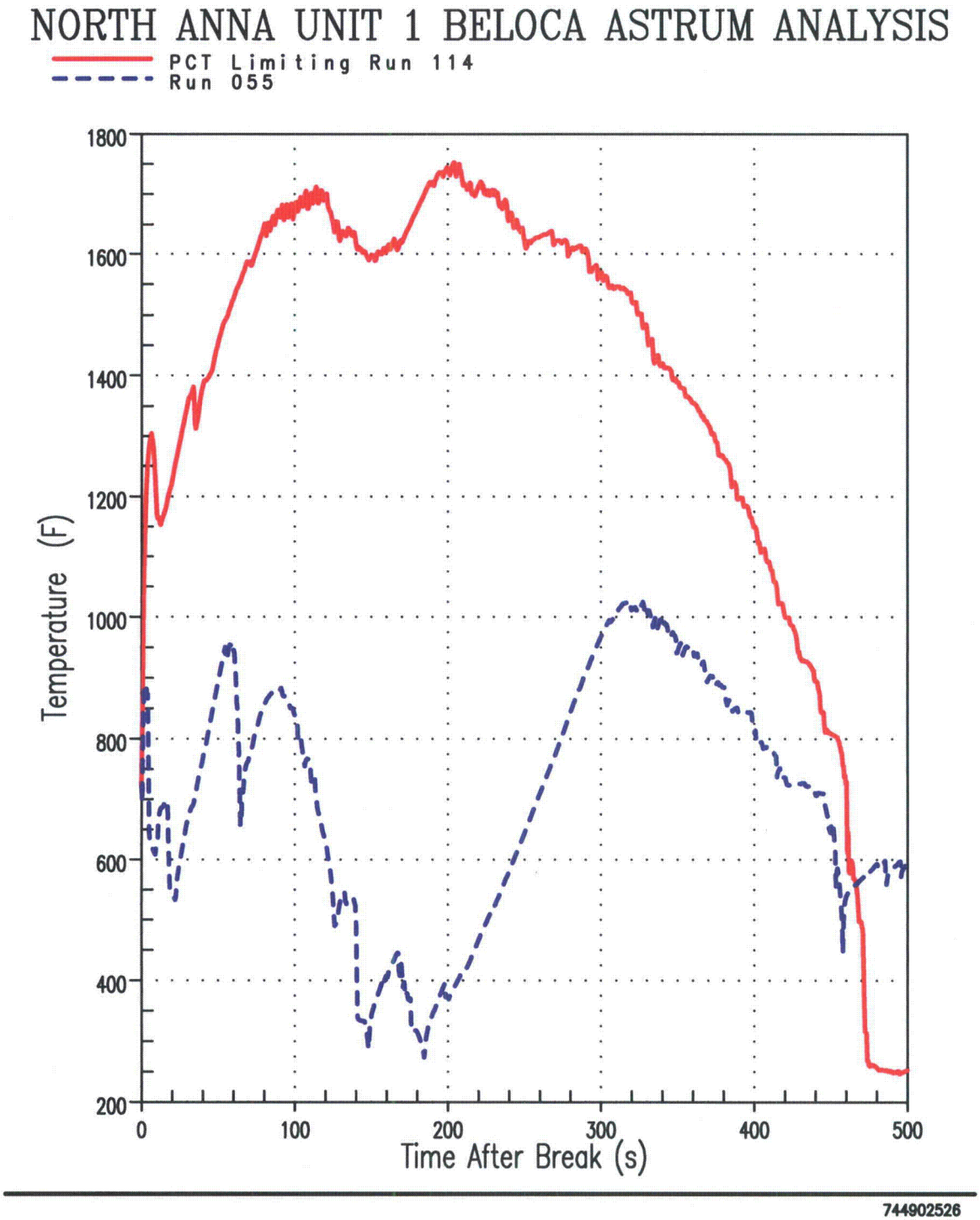

Figure 1-4: North Anna Unit 1 – Hot Rod PCT for Limiting PCT Run 114 vs. Run 055

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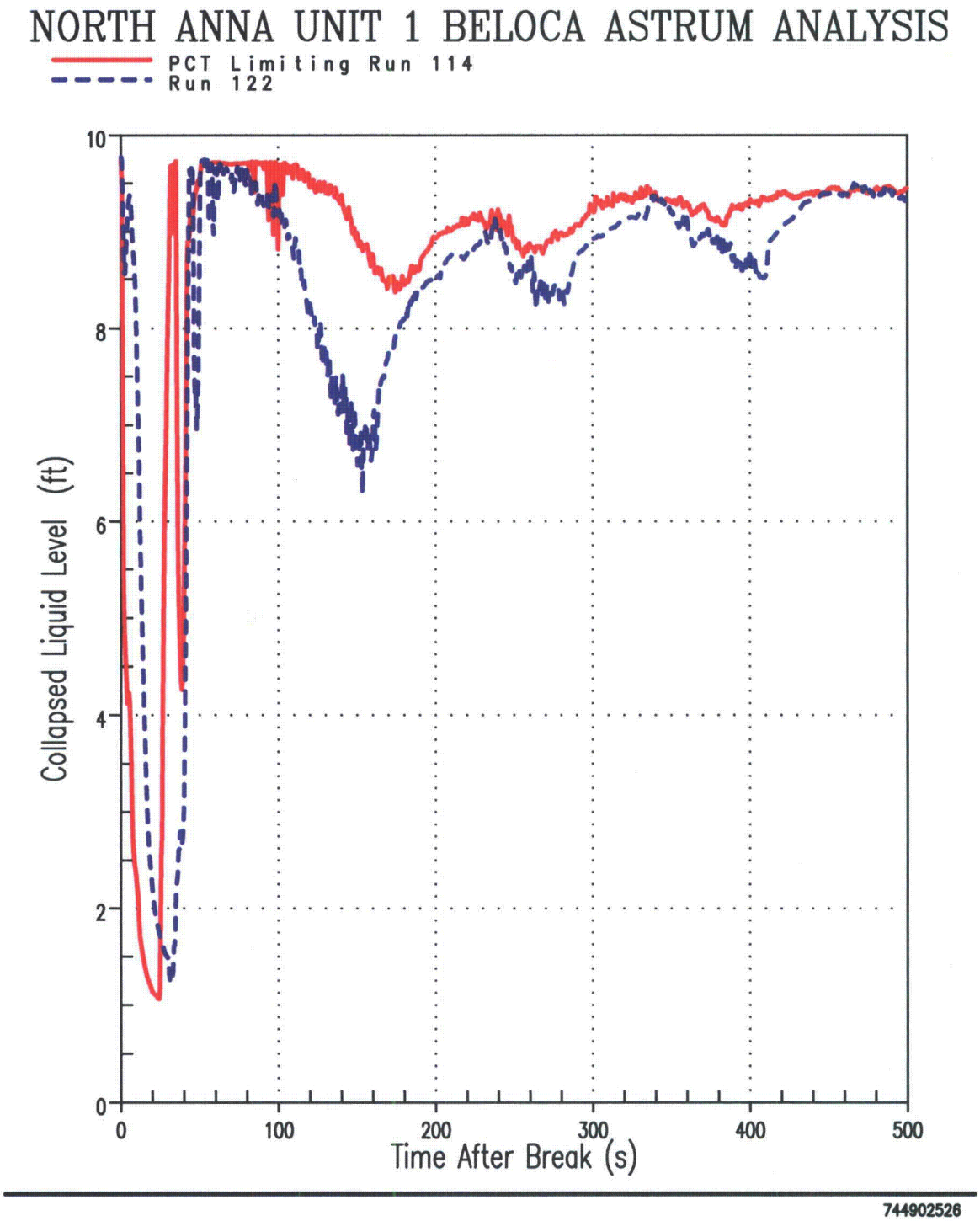

Figure 1-5: North Anna Unit 1 – Lower Plenum Collapsed Liquid Level for Limiting PCT Run 114 vs. Run 122

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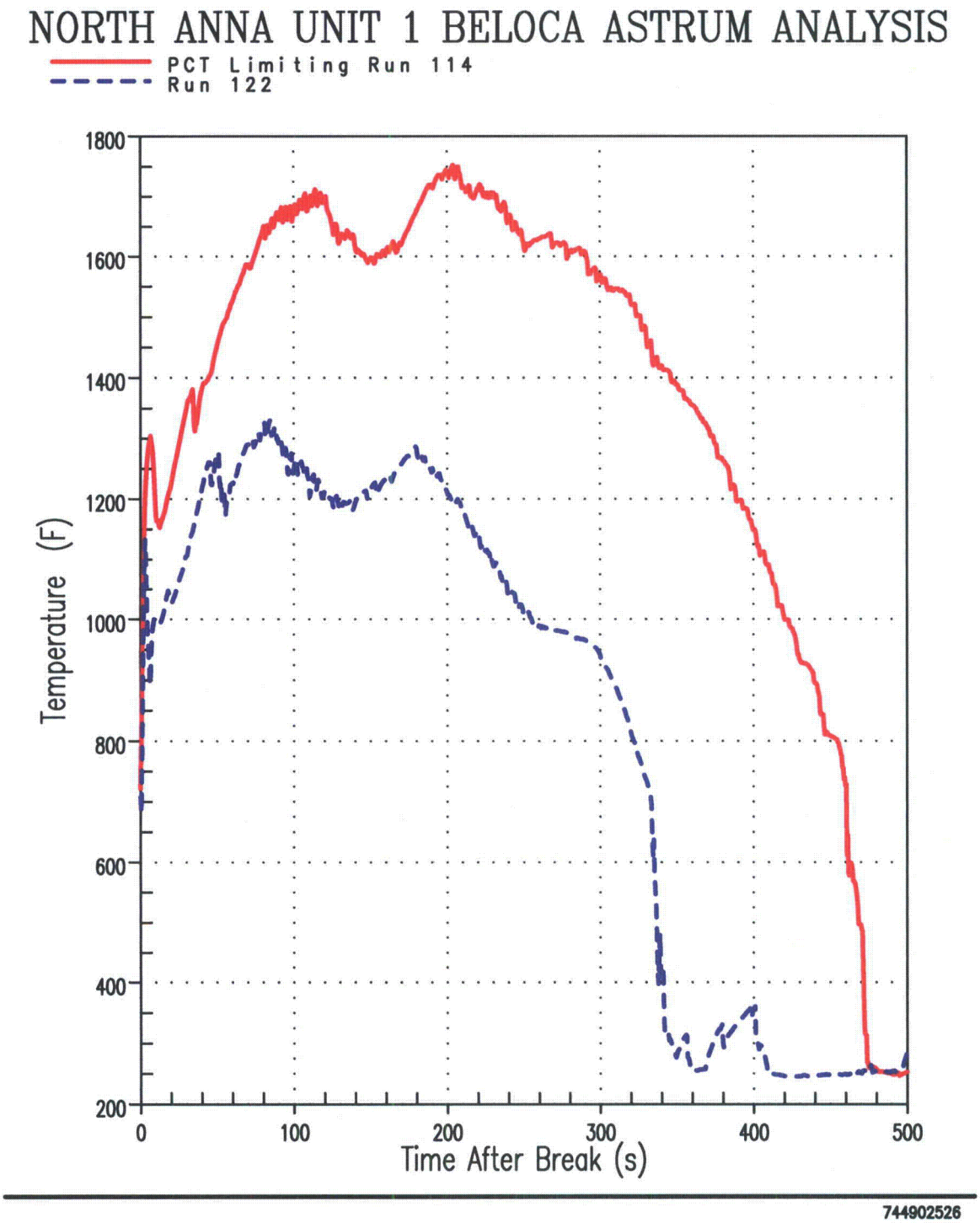

Figure 1-6: North Anna Unit 1 – Hot Rod PCT for Limiting PCT Run 114 vs. Run 122

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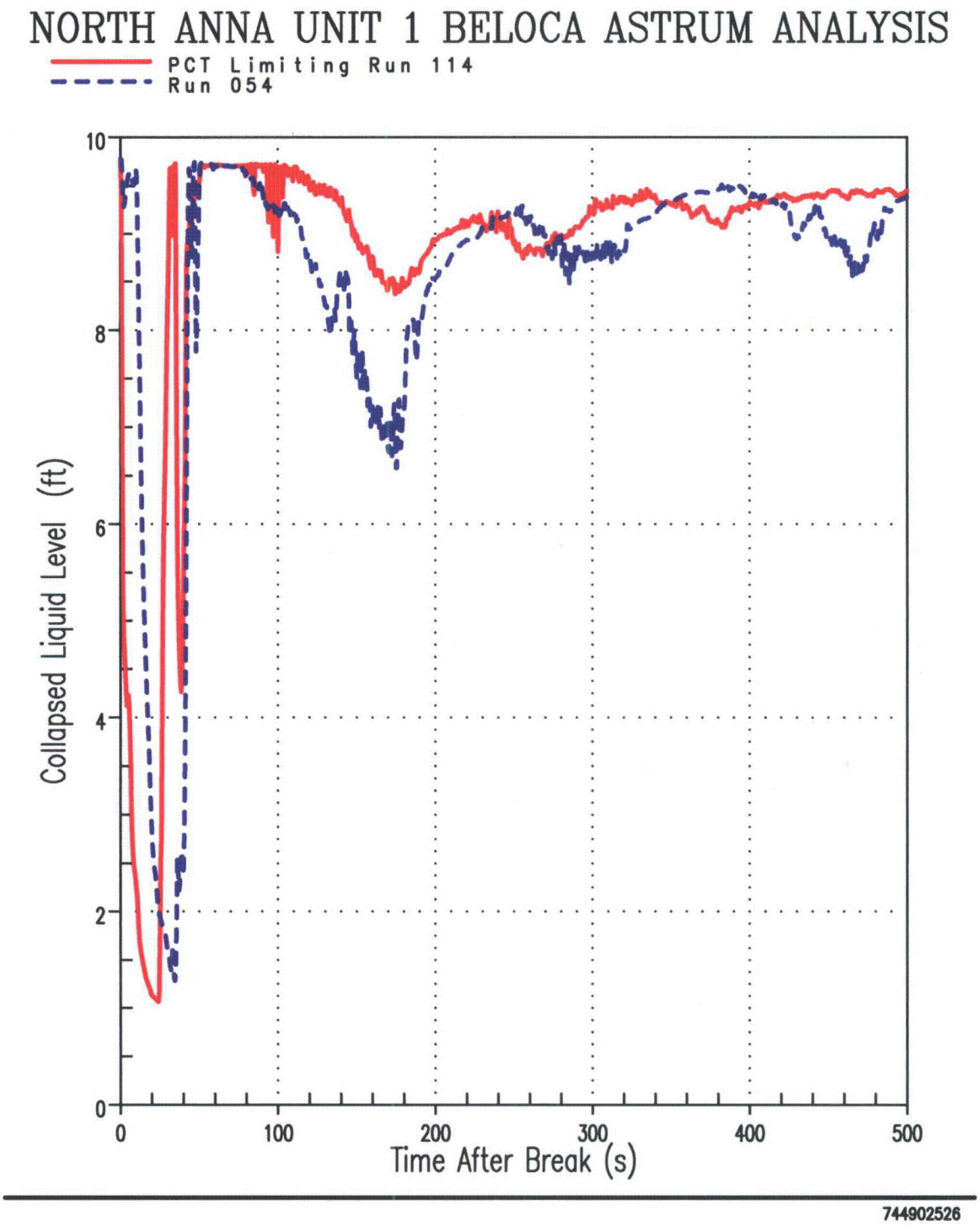

Figure 1-7: North Anna Unit 1 – Lower Plenum Collapsed Liquid Level for Limiting PCT Run 114 vs. Run 054

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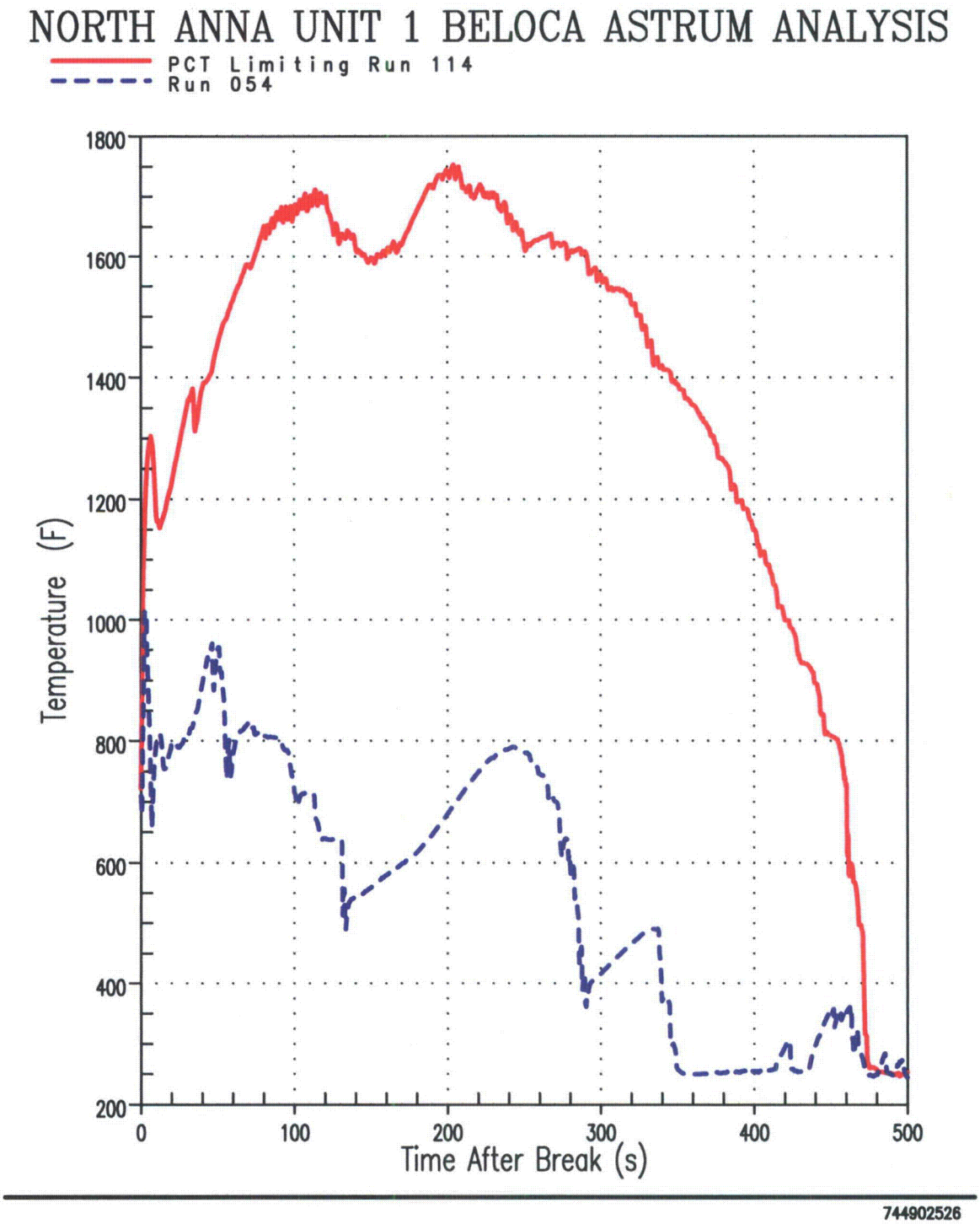

Figure 1-8: North Anna Unit 1 – Hot Rod PCT for Limiting PCT Run 114 vs. Run 054

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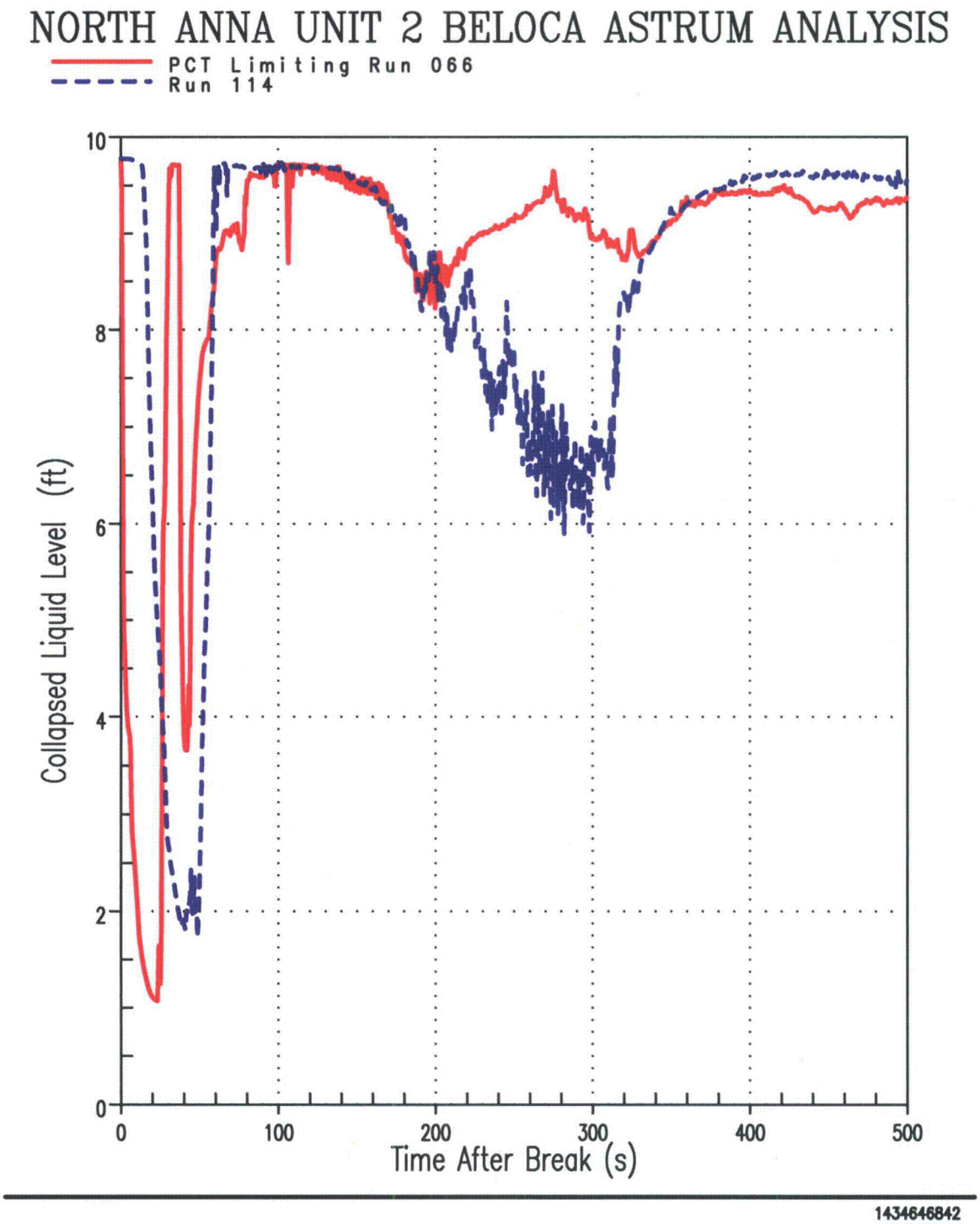

Figure 1-9: North Anna Unit 2 – Lower Plenum Collapsed Liquid Level for Limiting PCT Run 066 vs. Run 114

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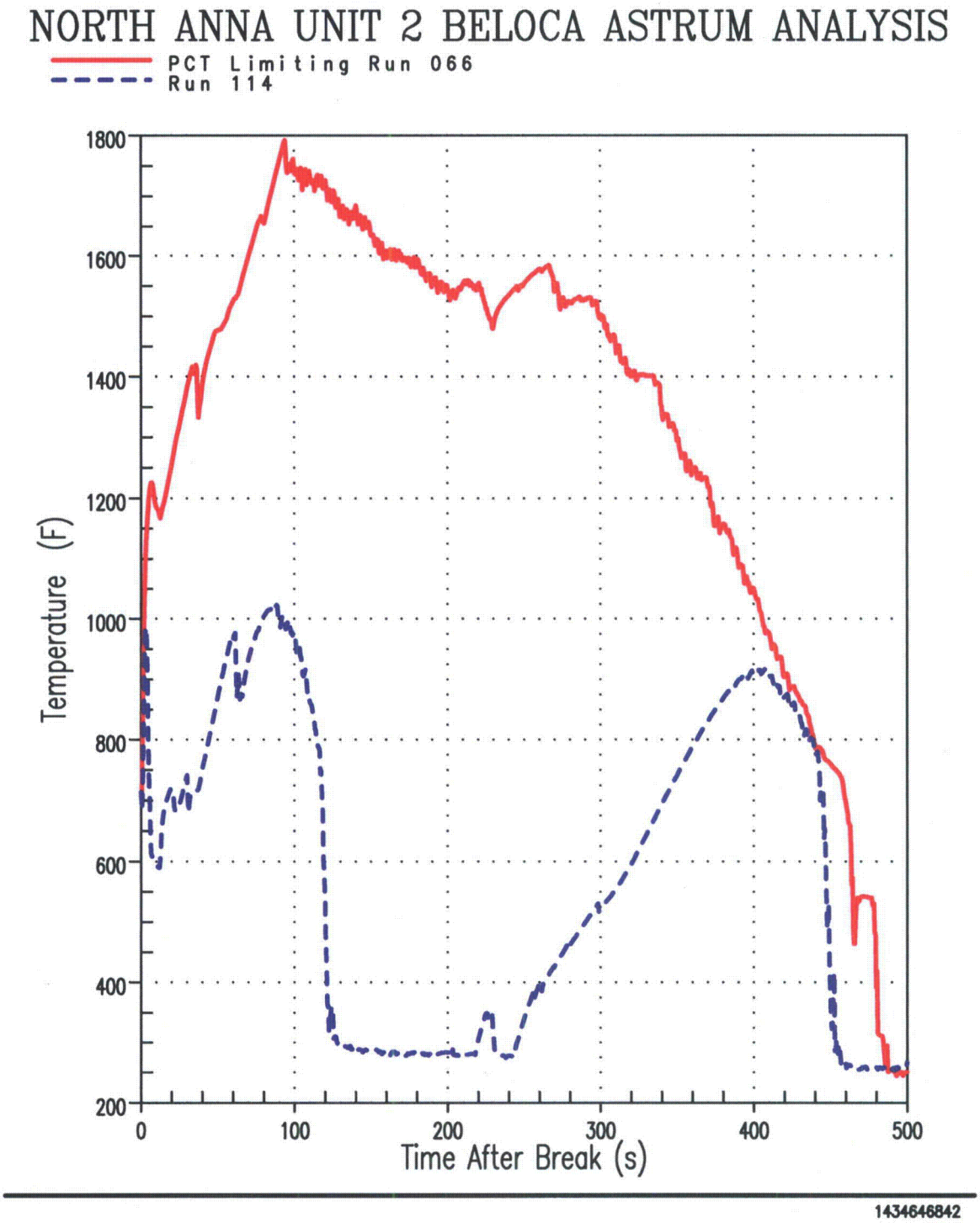

Figure 1-10: North Anna Unit 2 – Hot Rod PCT for Limiting PCT Run 066 vs. Run 114

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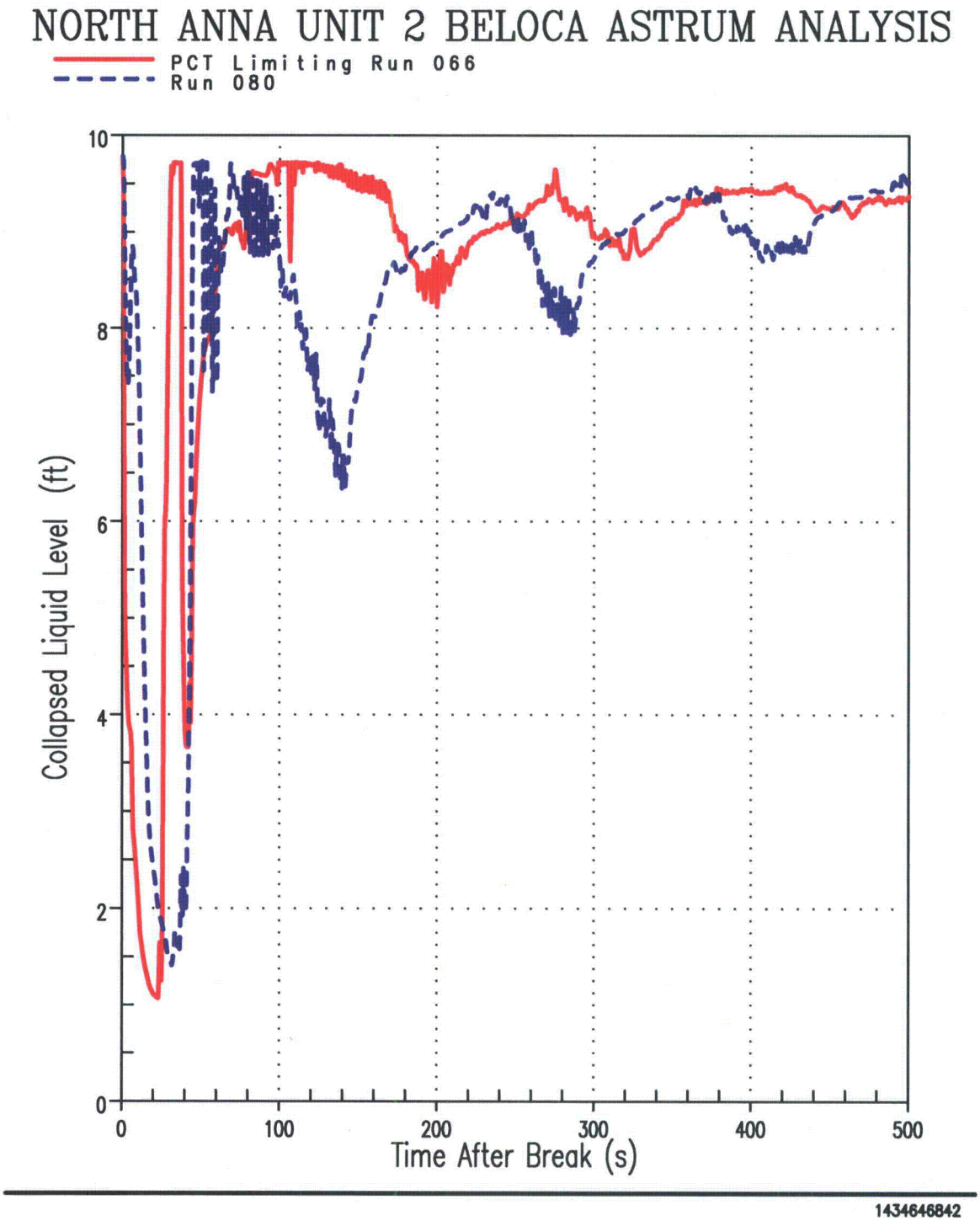

Figure 1-11: North Anna Unit 2 – Lower Plenum Collapsed Liquid Level for Limiting PCT Run 066 vs. Run 080

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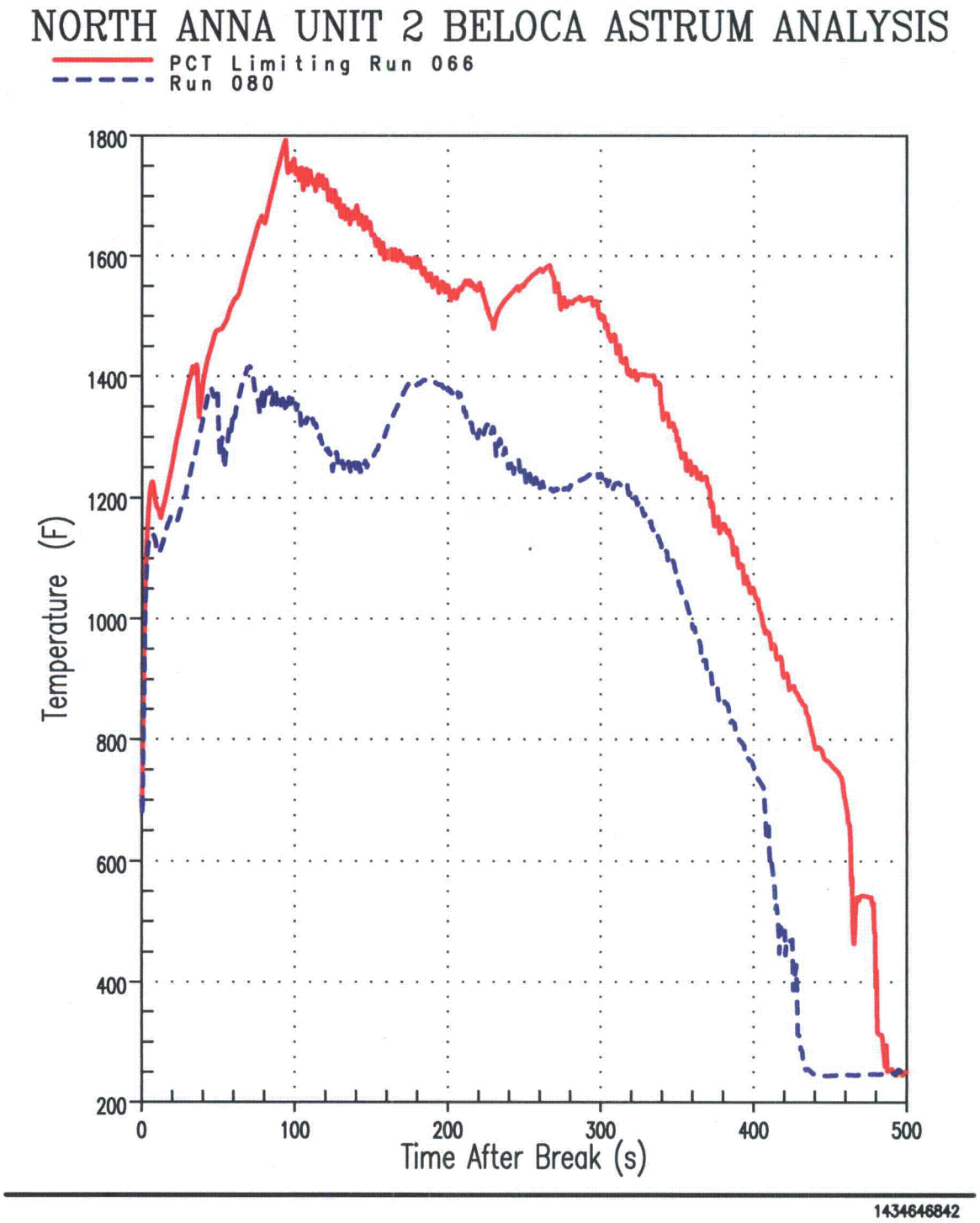

Figure 1-12: North Anna Unit 2 – Hot Rod PCT for Limiting PCT Run 066 vs. Run 080

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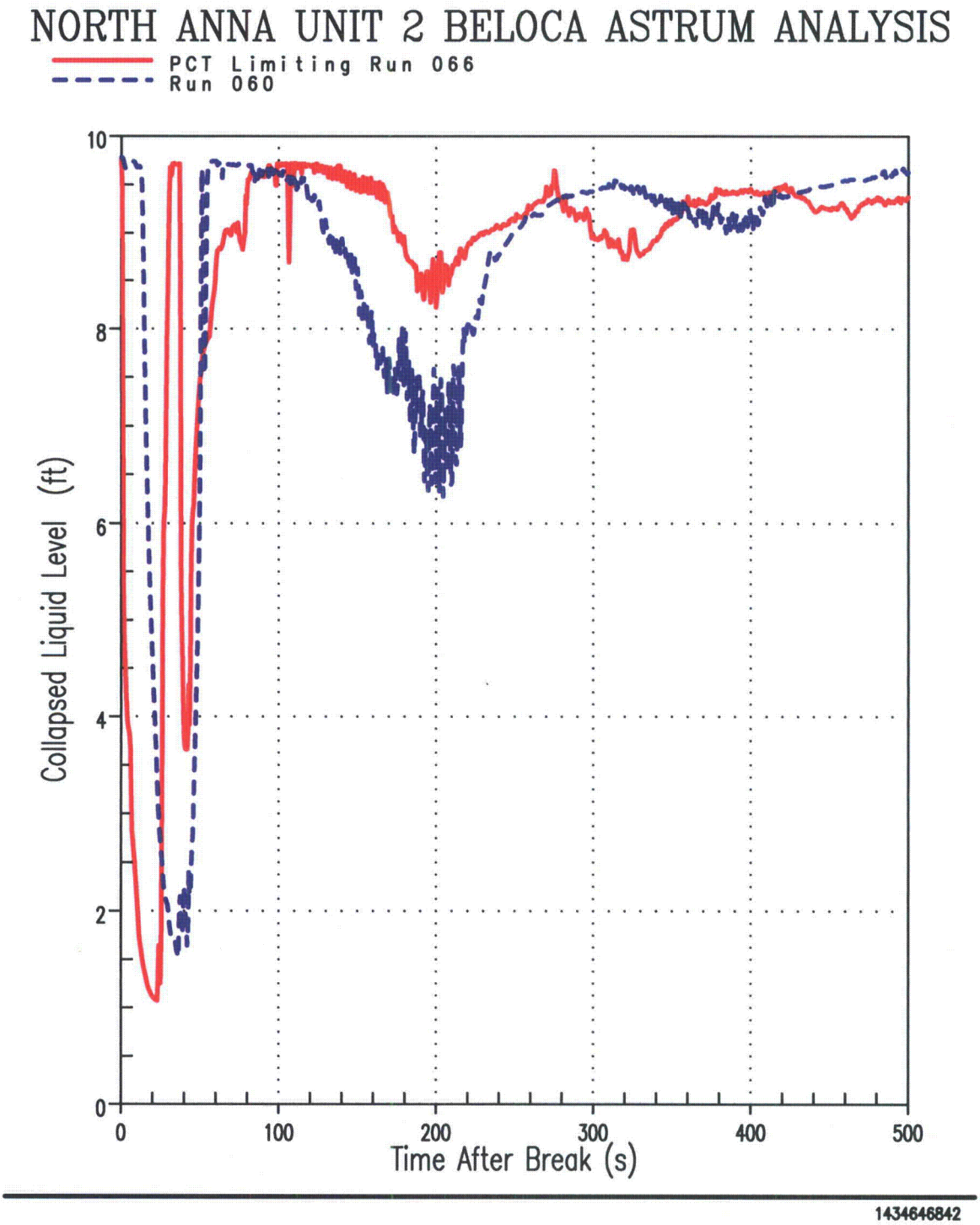

Figure 1-13: North Anna Unit 2 – Lower Plenum Collapsed Liquid Level for Limiting PCT Run 066 vs. Run 060

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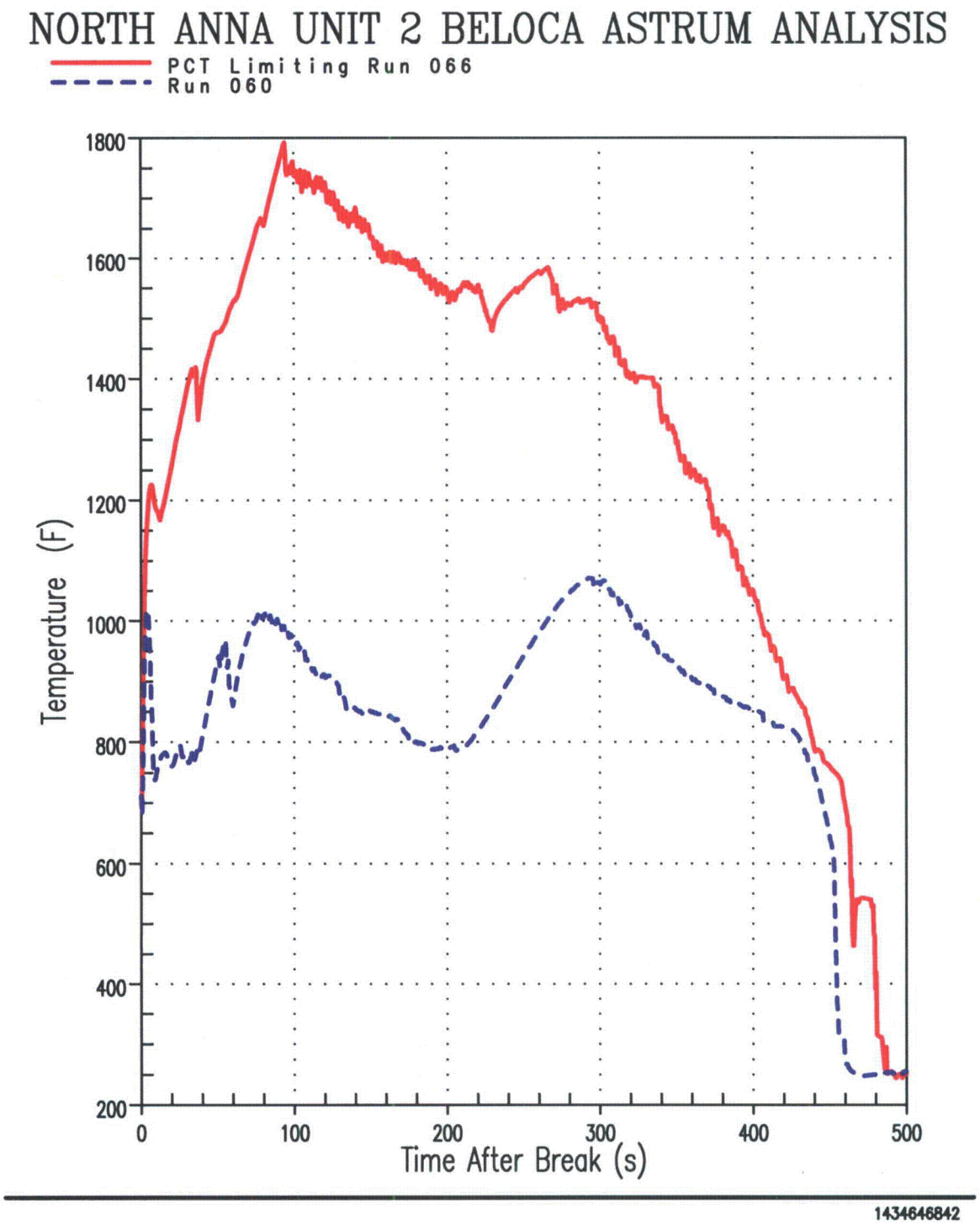

Figure 1-14: North Anna Unit 2 – Hot Rod PCT for Limiting PCT Run 066 vs. Run 060

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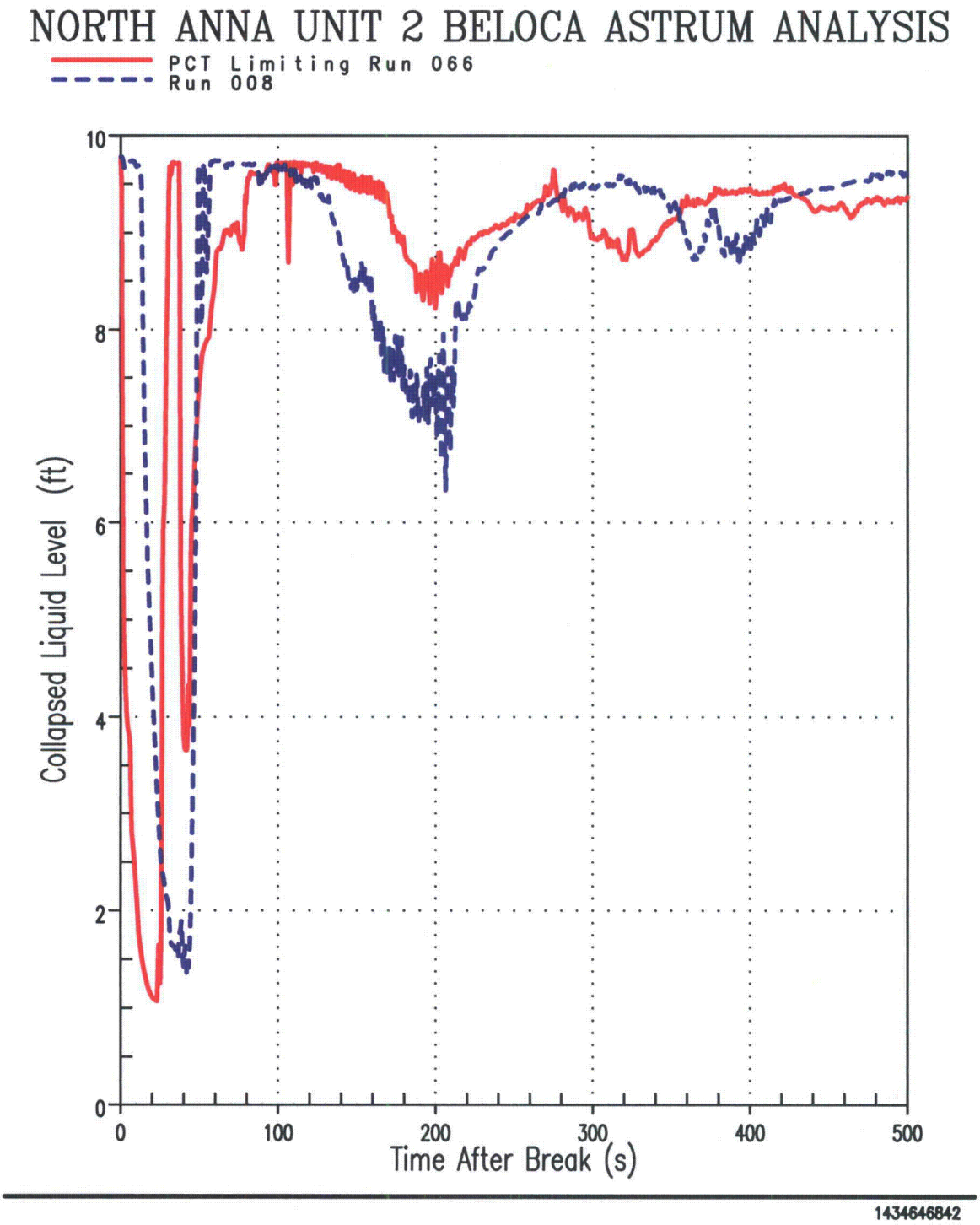

Figure 1-15: North Anna Unit 2 – Lower Plenum Collapsed Liquid Level for Limiting PCT Run 066 vs. Run 008

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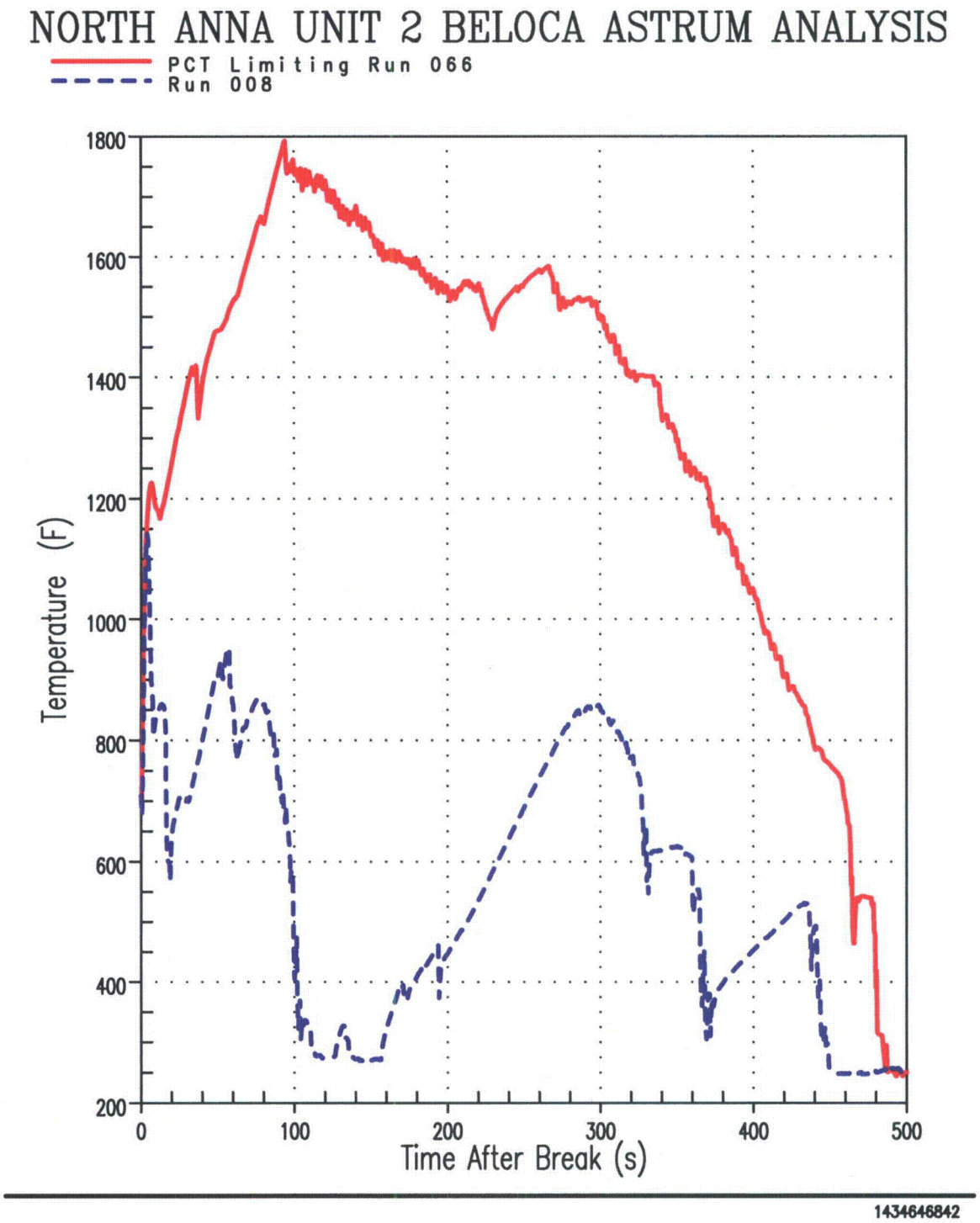

Figure 1-16: North Anna Unit 2 – Hot Rod PCT for Limiting PCT Run 066 vs. Run 008

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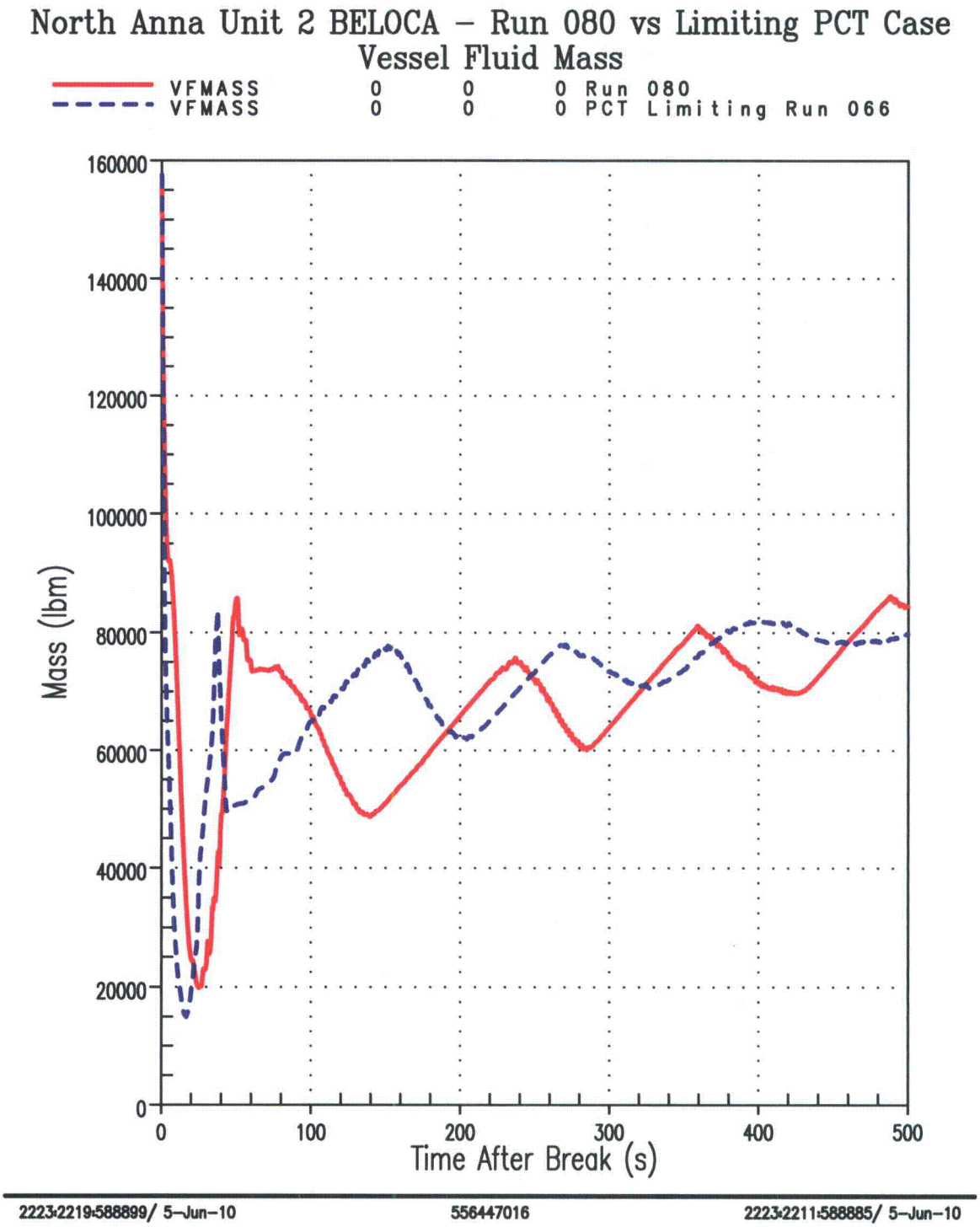

Figure 1-17: North Anna Unit 2 – Vessel Fluid Mass for Run 080 vs. Limiting PCT Run 066

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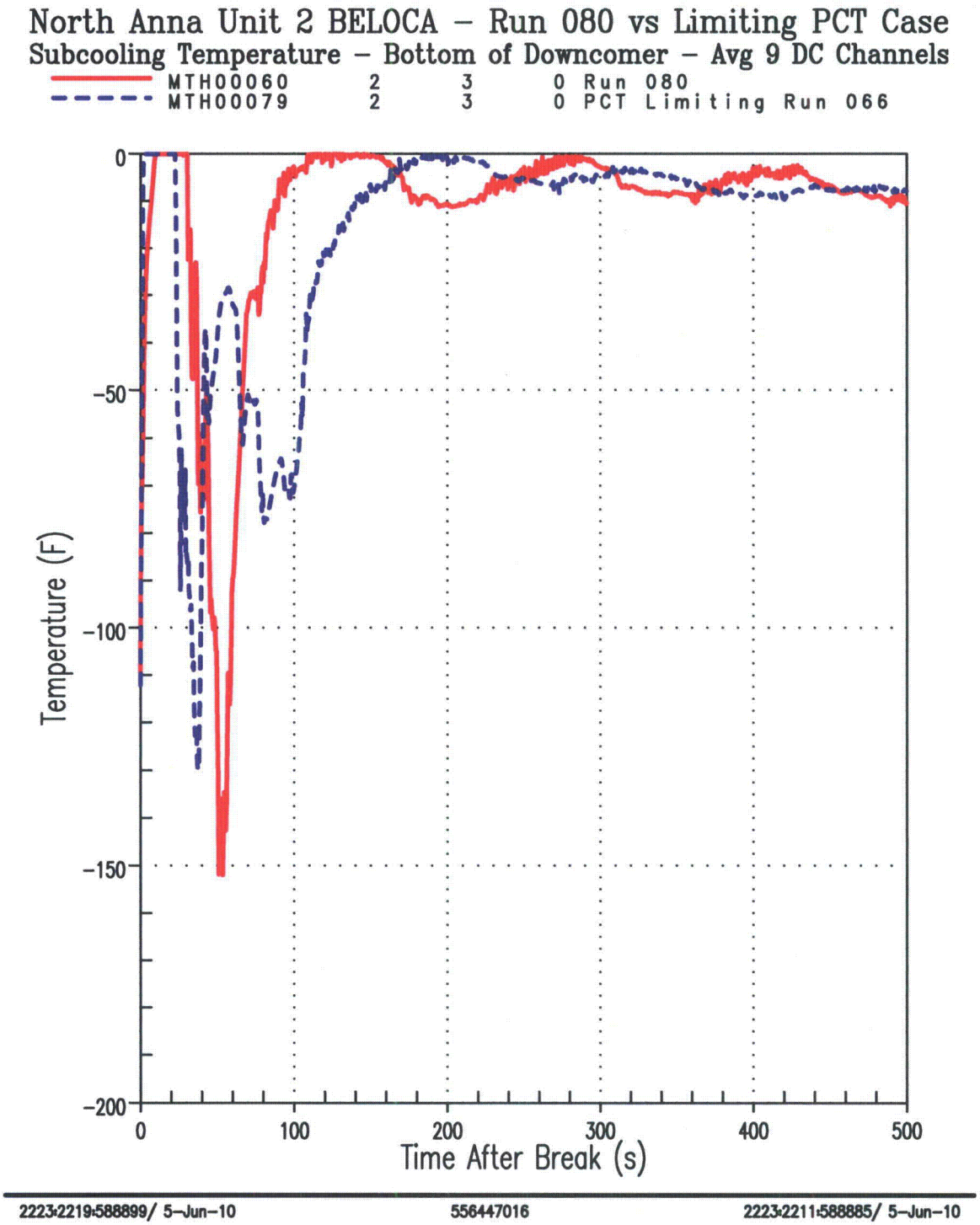

Figure 1-18: North Anna Unit 2 – Average Liquid Subcooling Comparison at the Near-Bottom of the Downcomer (Vessel Section 2, Cell 3), Run 080 vs. PCT Limiting Run 066

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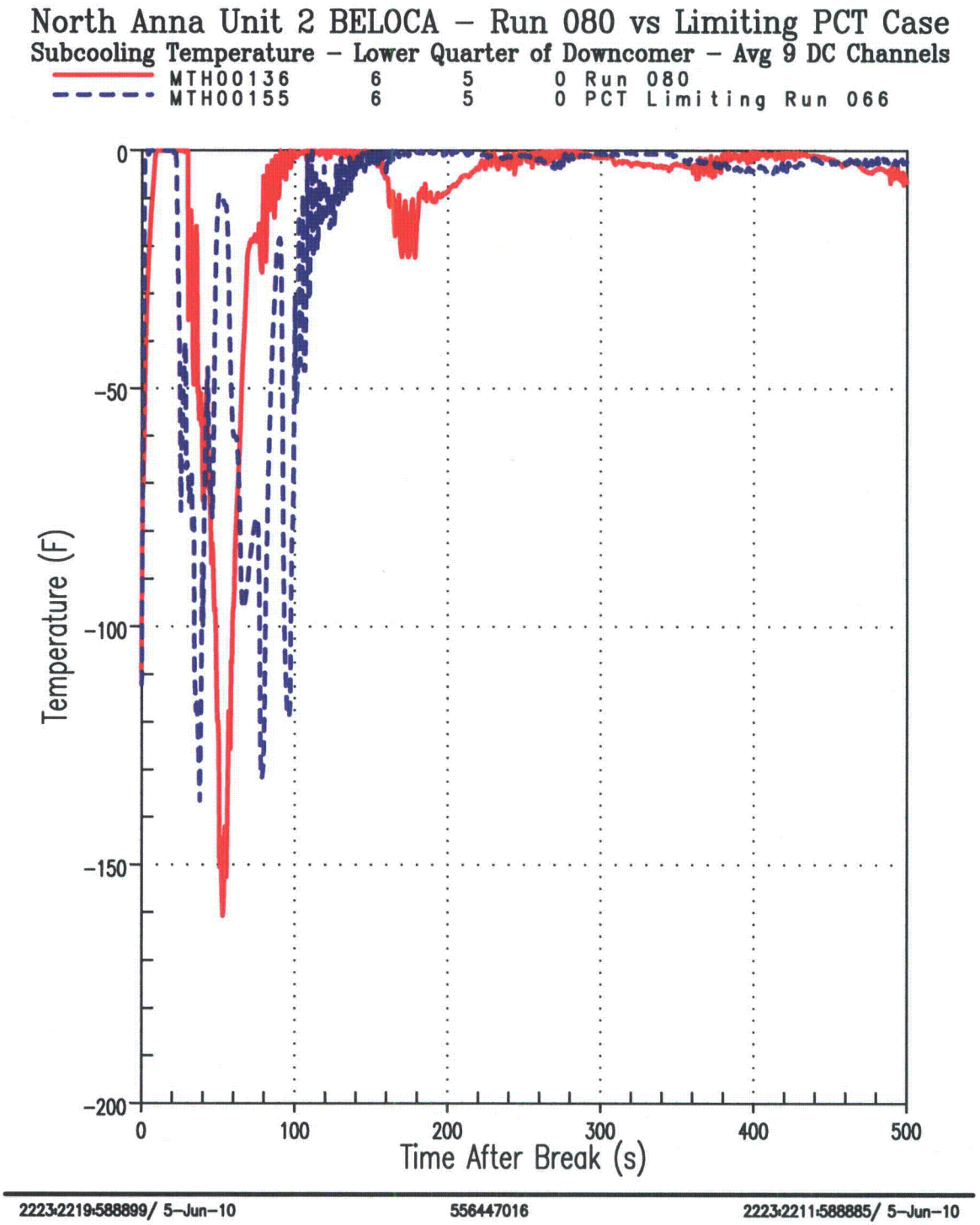

Figure 1-19: North Anna Unit 2 – Average Liquid Subcooling Comparison at the Lower Quarter of the Downcomer (Vessel Section 3, Cell 5), Run 080 vs. PCT Limiting Run 066

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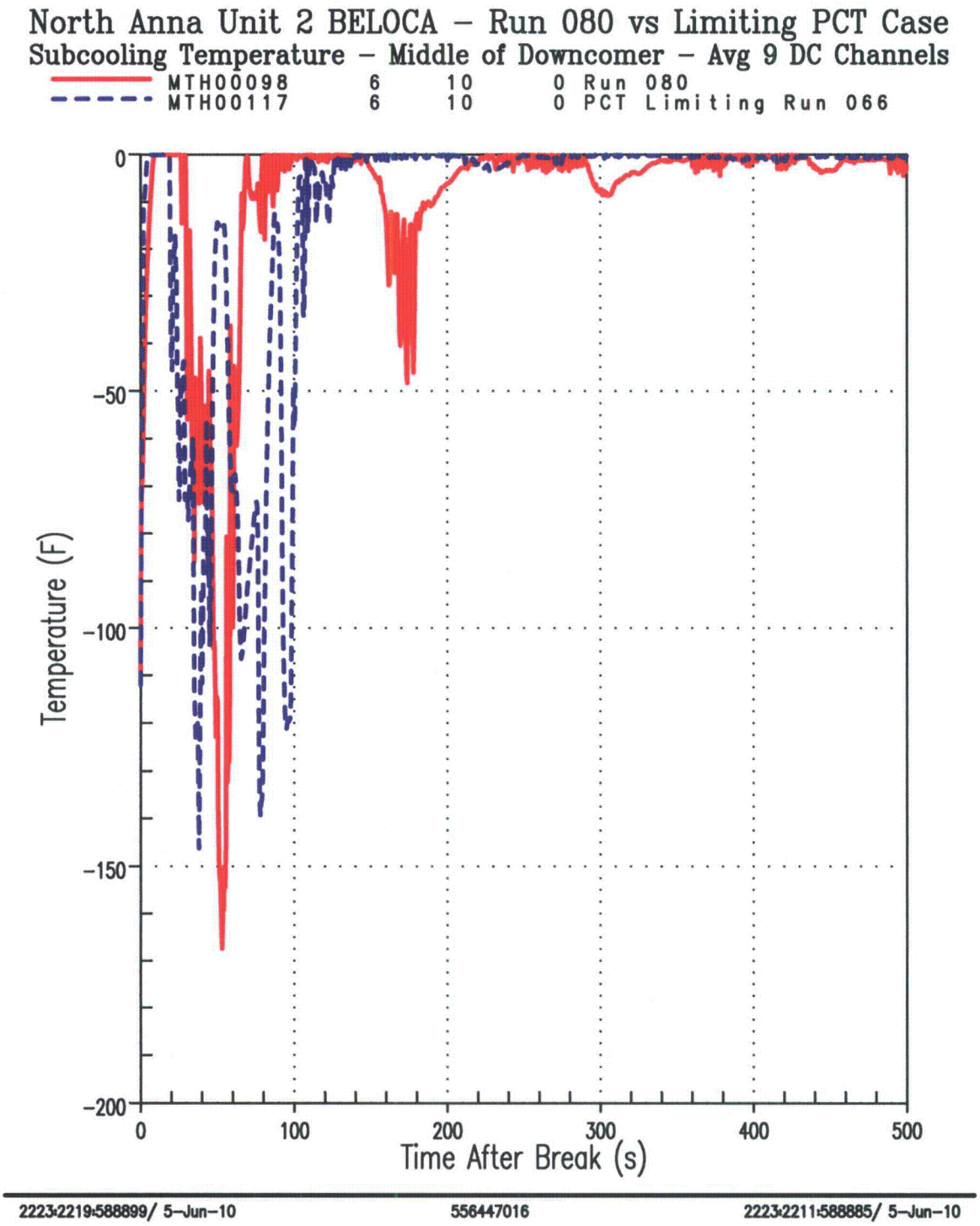

Figure 1-20: North Anna Unit 2 – Average Liquid Subcooling Comparison at the Near-Middle of the Downcomer (Vessel Section 3, Cell 10), Run 080 vs. PCT Limiting Run 066

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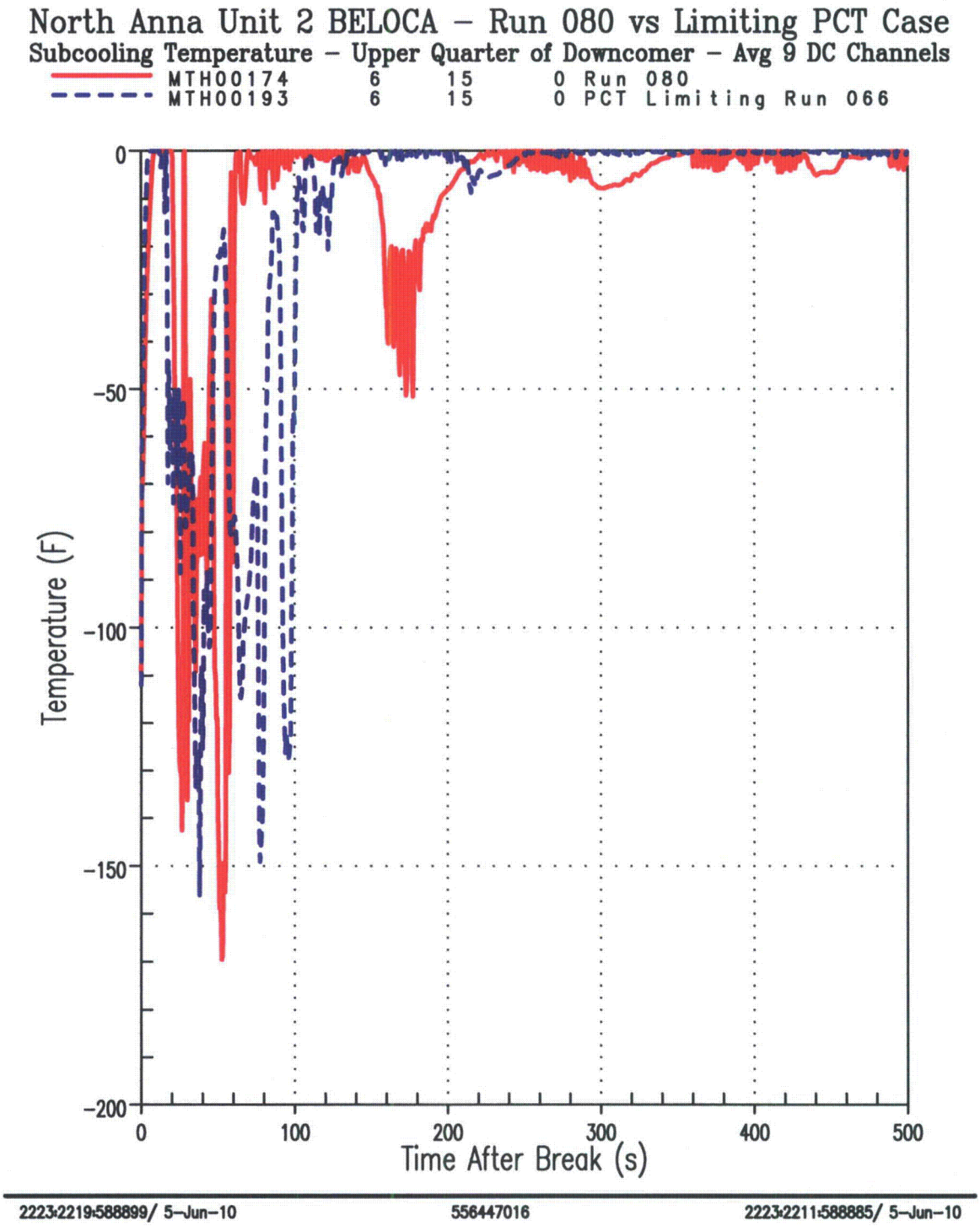

Figure 1-21: North Anna Unit 2 – Average Liquid Subcooling Comparison at the Upper Quarter of the Downcomer (Vessel Section 3, Cell 15), Run 080 vs. PCT Limiting Run 066

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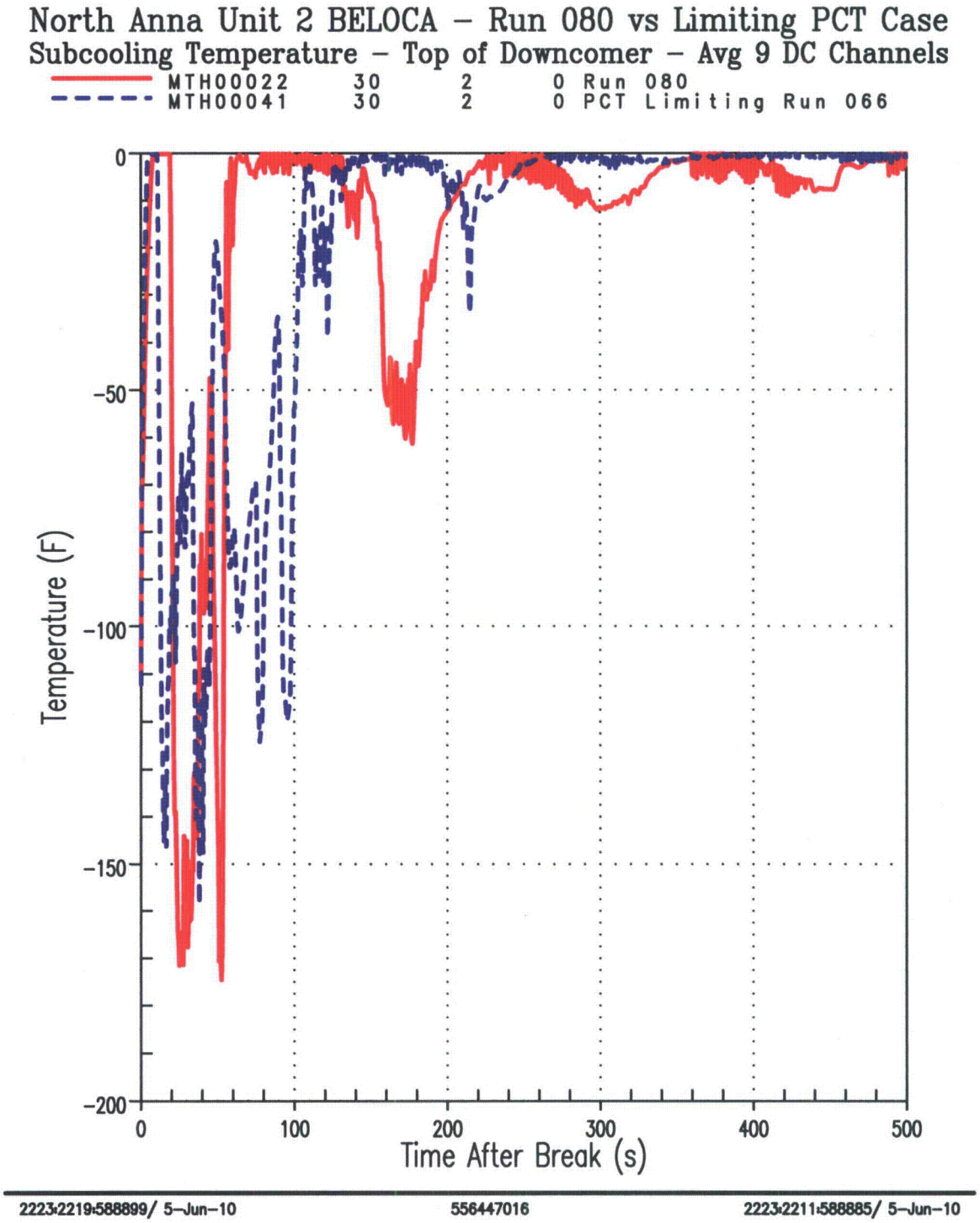

Figure 1-22: North Anna Unit 2 – Average Liquid Subcooling Comparison at the Near-Top of the Downcomer (Vessel Section 6, Cell 2), Run 080 vs. PCT Limiting Run 066

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RAI Question 2

With respect to the analysis employing a plant-specific adaptation of the ASTRUM evaluation model, please provide the following:

a) A clarification that a plant-specific adaption of the ASTRUM evaluation model is still within the approved limitations and conditions stated in the staff SE;

c) The plant nodalization scheme for NAPS; and

RESPONSE

The response to BE RAI #1 addresses the clarification desired for part (a) of BE RAI #2. For part (c), Figures 2-1 through 2-4 correspond to the Unit 1 Vessel Model, Figures 2-5 through 2-8 correspond to the Unit 2 Vessel Model, and Figure 2-9 shows the Loop Model for both Unit 1 and Unit 2. The differences between the Unit 1 Vessel Model and Unit 2 Vessel Model are:

Unit 1 is a Westinghouse 3-loop pressurized water reactor (PWR) with the barrel/baffle region converted from a downflow design to an upflow design. Unit 2 is a Westinghouse 3-loop PWR with currently/originally downflow design in the barrel/baffle region. Consequently, the Unit 2 ('VGB') model includes one additional channel (#90) in Section 4 to connect the region above the barrel/baffle region to the barrel/baffle region. Also, the downflow design includes core barrel holes (not in the upflow design) which are reflected as Gaps 14-16 and 93-98 in Section 4 to connect the downcomer channels (Channels 14-16, 64-69) to the new channel 90 lying in the region between the baffle plates and core barrel.

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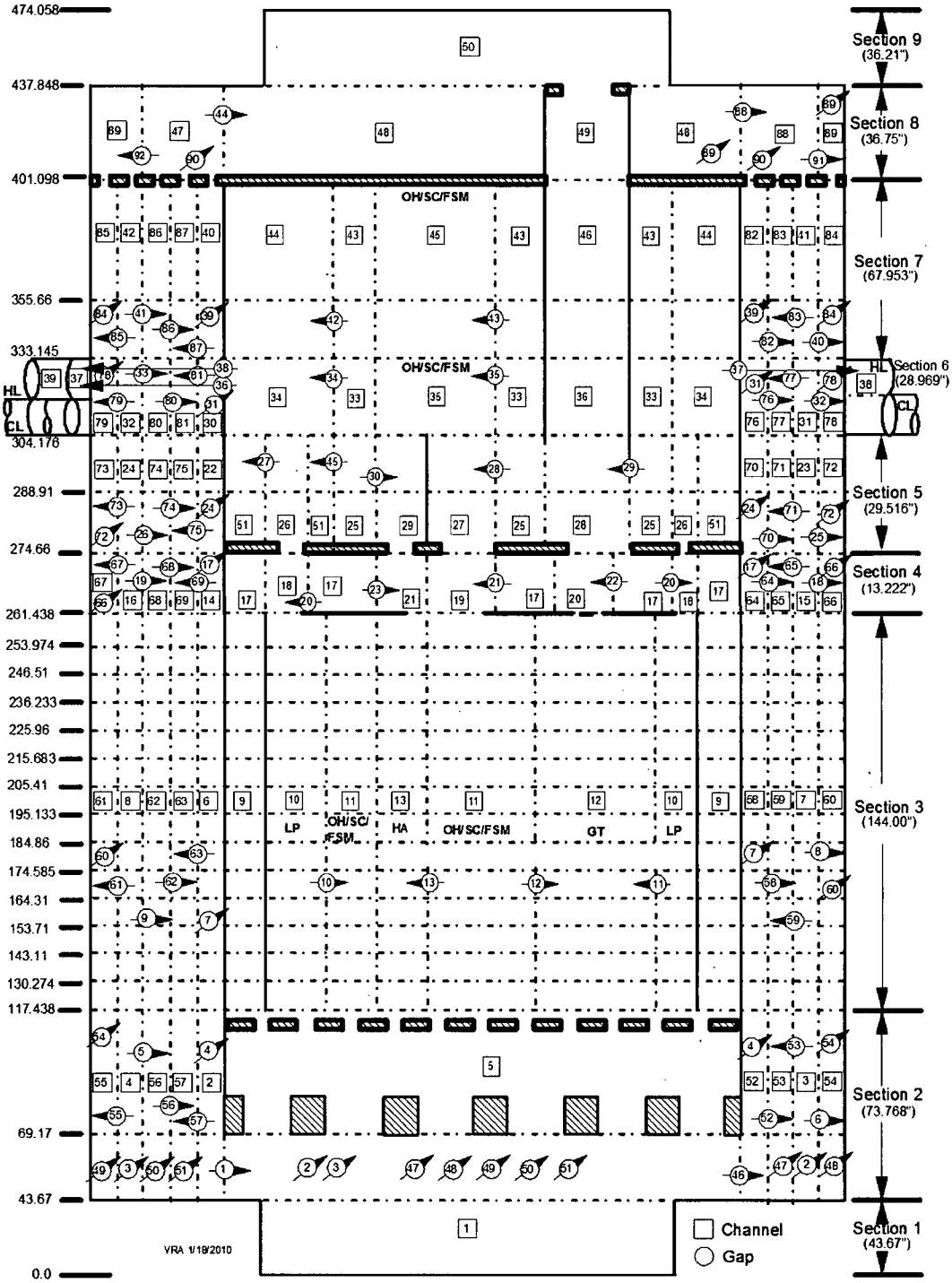

Figure 2-1: North Anna Unit 1 Vessel Noding Diagram for Nine Downcomer Channel Model

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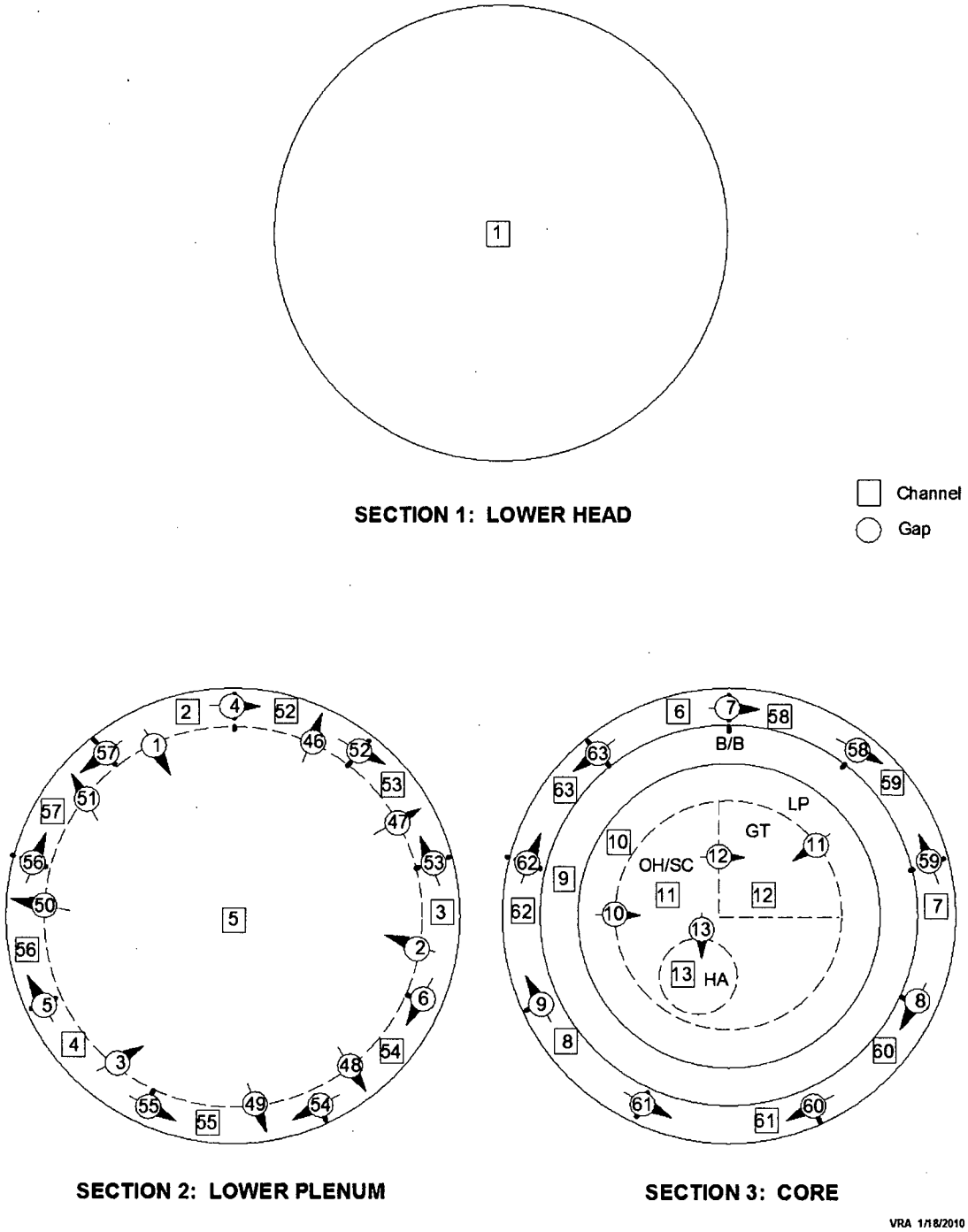

Figure 2-2: North Anna Unit 1 Vessel Sections 1, 2, and 3 for Nine Downcomer Channel Model (Horizontal View)

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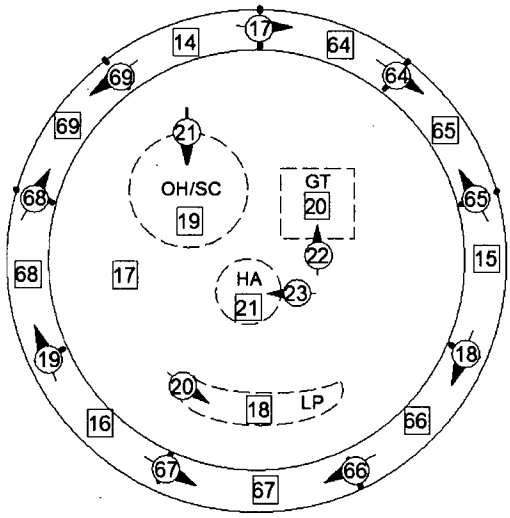
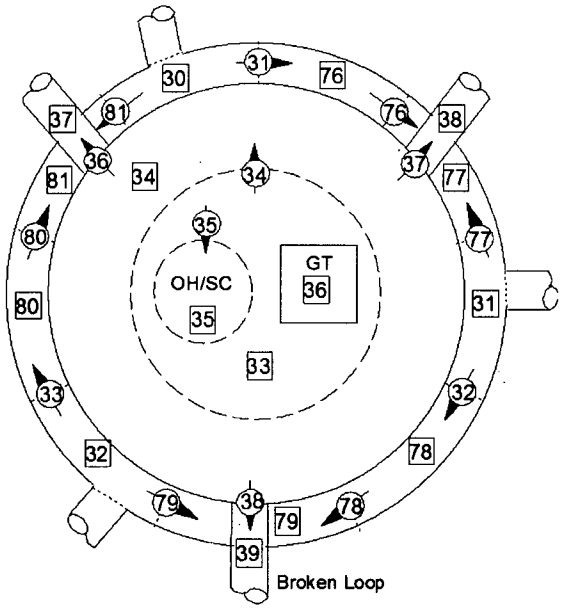


SECTION 4: CCFL REGION

Channel

Gap

SECTION 5: UPPER PLENUM BELOW NOZZLES



SECTION 6: NOZZLE REGION

[ ]a,c

Figure 2-3: North Anna Unit 1Vessel Section 4, 5, and 6 for Nine Downcomer Channel Model (Horizontal View)

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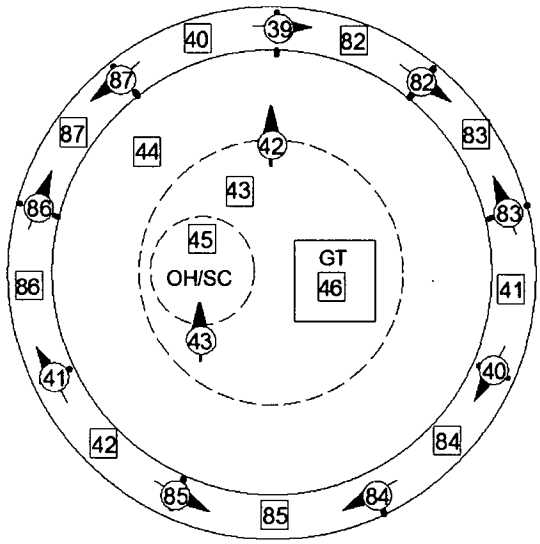
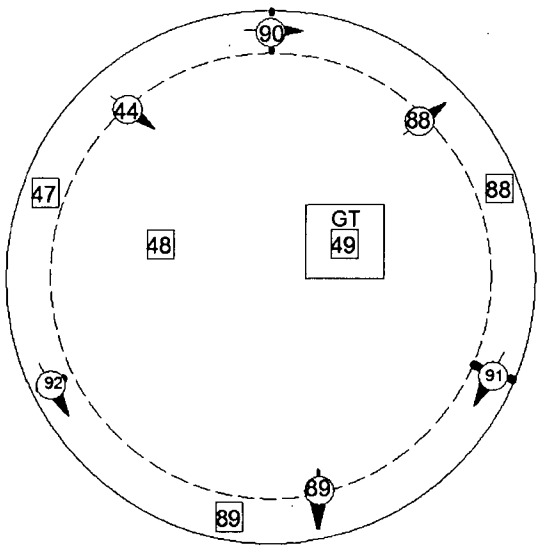


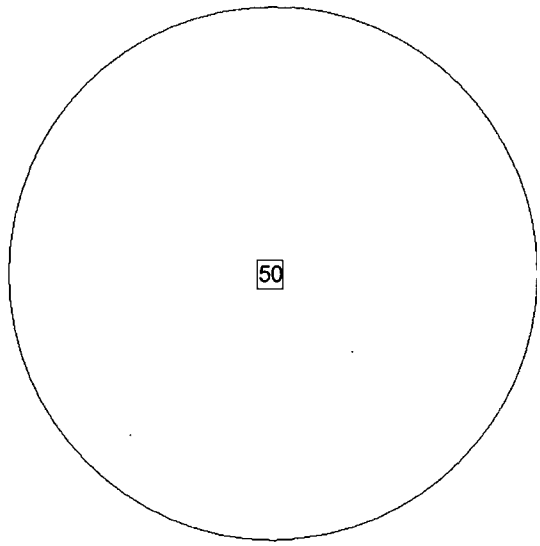
SECTION 7: UPPER PLENUM ABOVE NOZZLES

Channel

Gap



SECTION 8: UPPER HEAD UP TO TOP OF GUIDE TUBES



SECTION 9: UPPER HEAD ABOVE GUIDE TUBES

Figure 2-4: North Anna Unit 1Vessel Sections 7, 8, and 9 for Nine Downcomer Channel Model (Horizontal View)

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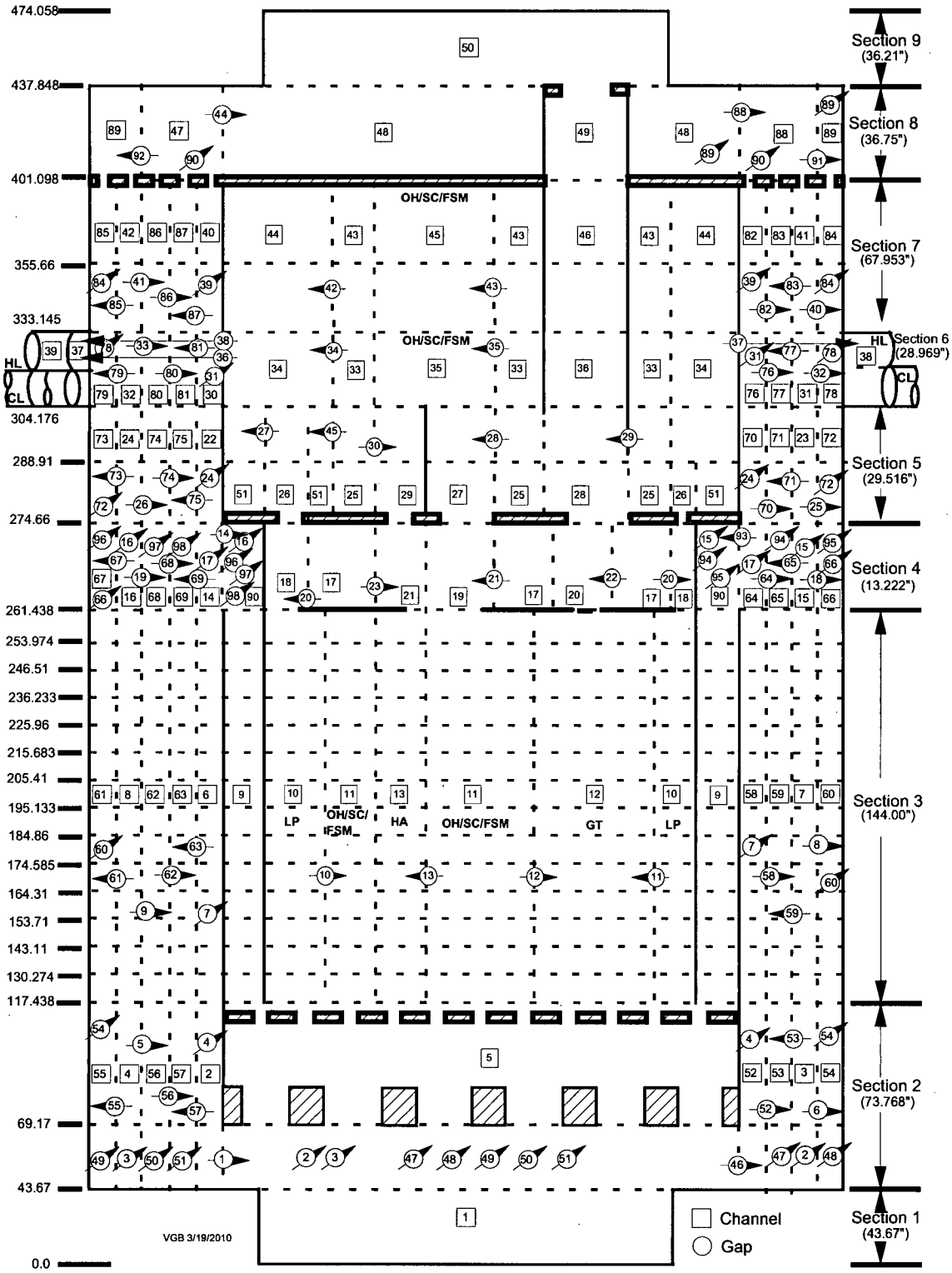

Figure 2-5: North Anna Unit 2 Vessel Noding Diagram for Nine Downcomer Channel Model

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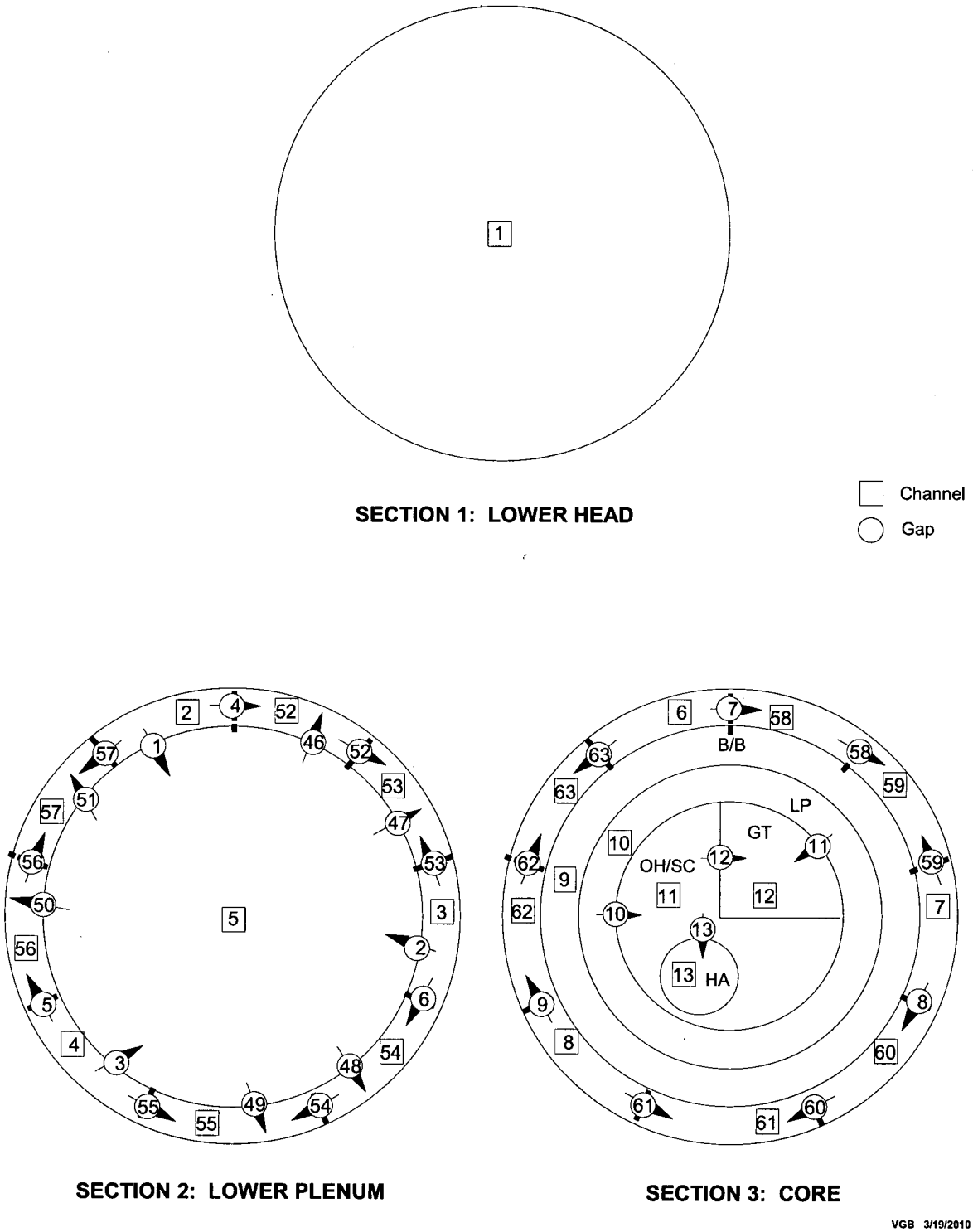

Figure 2-6: North Anna Unit 2 Vessel Sections 1, 2, and 3 for Nine Downcomer Channel Model (Horizontal View)

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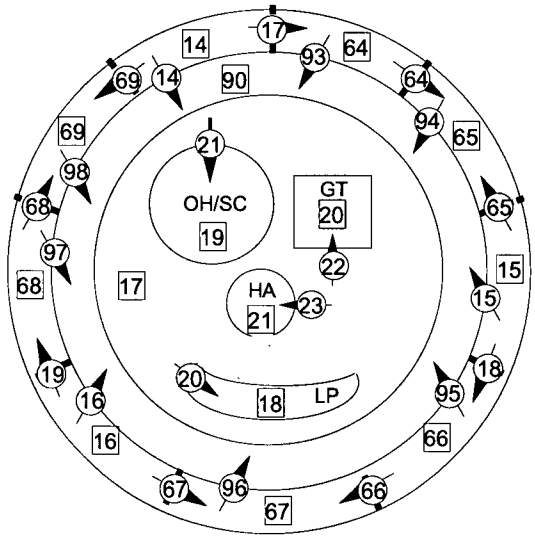
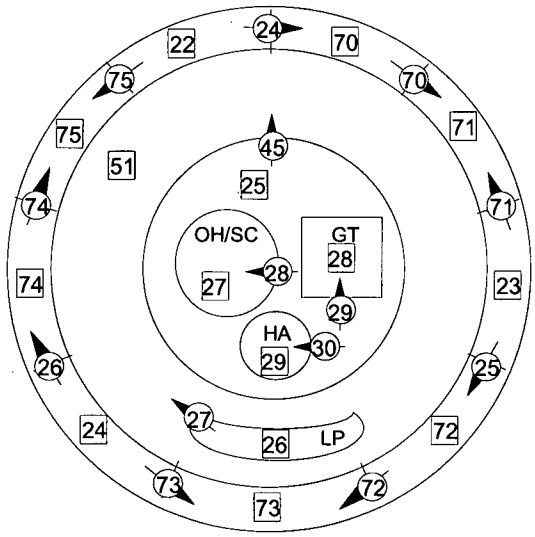


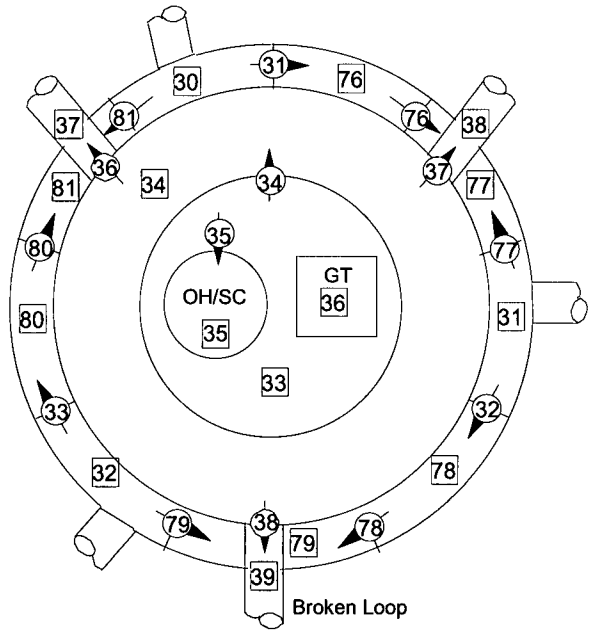
SECTION 4: CCFL REGION

Channel

Gap



SECTION 5: UPPER PLENUM BELOW NOZZLES



SECTION 6: NOZZLE REGION

[ ]a,c

Figure 2-7: North Anna Unit 2 Vessel Sections 4, 5, and 6 for Nine Downcomer Channel Model (Horizontal View)

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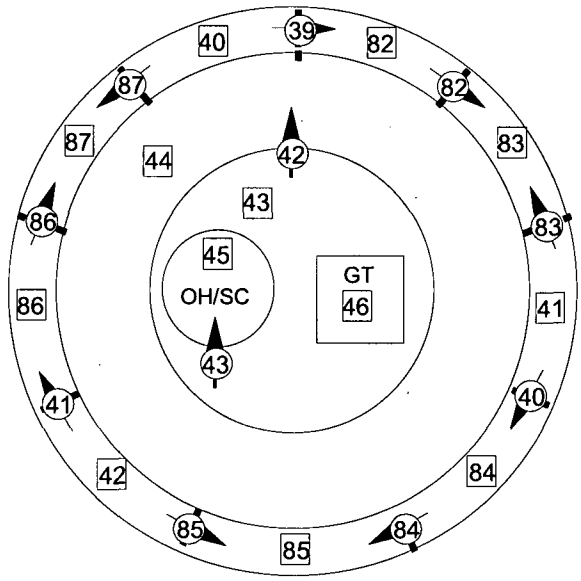
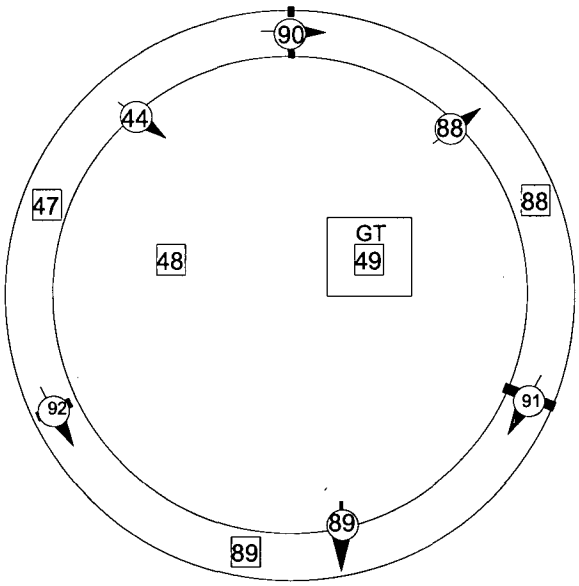


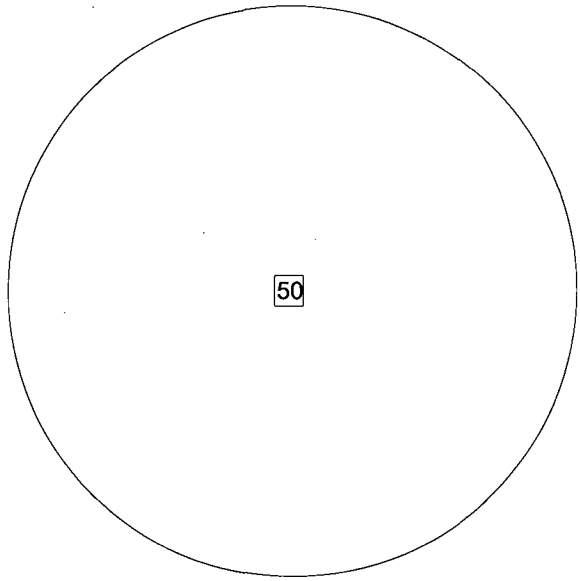
SECTION 7: UPPER PLENUM ABOVE NOZZLES

Channel

Gap



SECTION 8: UPPER HEAD UP TO TOP OF GUIDE TUBES



SECTION 9: UPPER HEAD ABOVE GUIDE TUBES

Figure 2-8: North Anna Unit 2 Vessel Sections 7, 8, and 9 for Nine Downcomer Channel Model (Horizontal View)

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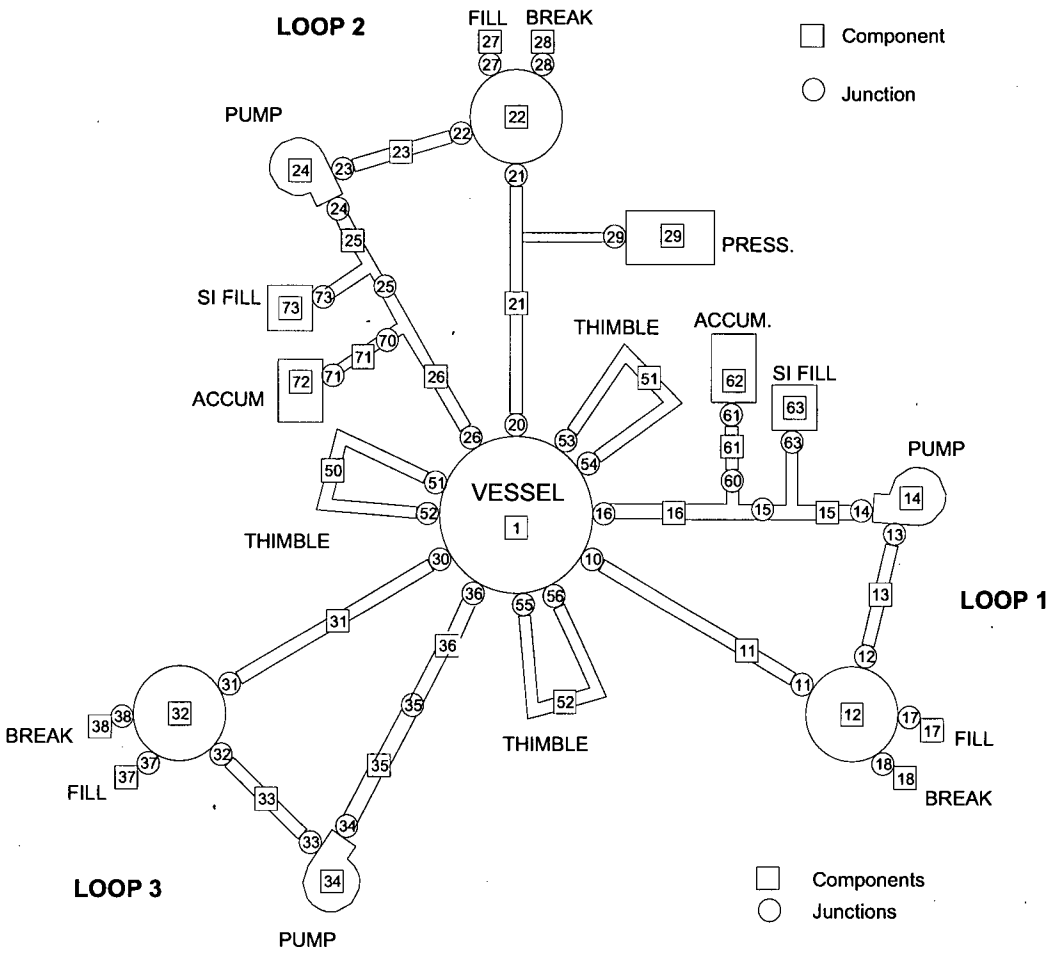

Figure 2-9: North Anna Units 1 and 2 1D Loop Noding Diagram (Steady State)

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RAI Question 4

Please describe the reason why higher peak cladding temperatures (PCTs) fall in the range of CD * Abreak/ACL values between 1.0 and 2.0 on Figure 1 for Unit 1 and between 0.9 and 2.3 on Figure 16 for Unit 2. Also, clarify that the lower break size (around 0.8) for the split break case and the higher break size (around 2.2) for the double-ended guillotine break case yield a similar high PCT for Unit 2, while the high PCTs are dominated by double-ended guillotine break at an effective break size of 1.9 for Unit 1.

RESPONSE

As a preface remark, it is emphasized that the two scatter plots do not represent 124 pairs of sensitivities to Unit 1 vs. Unit 2, because different randomly assigned seeds were applied in the attribute sampling process. The scatter plots show that the top set of results are extremely identical, perhaps not as different or as wide ranging as one might infer from the comment. The only notable difference between the two is perhaps the singular high PCT small effective break area split break PCT result seen at Figure 16 (Unit 2) that does not have a similar counterpart for Figure 1 (Unit 1). Inspecting this singular case, it is determined that its reflood heat transfer multiplier attribute was [ ]^{a,c} is given in Figure 1-5 of WCAP-16009-P-A), and is the third lowest value in the set of 124 runs for Unit 2. For a plant that is not blowdown limited, this is a particularly dominant attribute. Discounting this singular understood result, the top 7-8 cases for both analyses fall into a very similar effective break area range (1.7-2.2, overall), though it is acknowledged that Unit 2 tends to be limiting at a slightly higher effective break area that is not considered statistically significant.

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RAI Question 8

Provide the following information regarding the NAPS nuclear steam supply system:

a) Volume of the lower plenum, core and upper plenum below the bottom elevation of the hot leg, each identified separately. Also provide heights of these regions.

b) Loop friction and geometry pressure losses from the core exit through the steam generators (SGs) to the inlet nozzle of the reactor vessel. Also, provide the locked rotor reactor coolant pump (RCP) k-factor. Provide the mass flow rates, flow areas, k-factors, and coolant temperatures for the pressure losses (upper plenum, hot legs, SGs, suction legs, RCPs, and discharge legs). Include the reduced SG flow areas due to plugged tubes. Provide the loss from each of the intact cold legs through the annulus to a single broken cold leg. Also, provide the equivalent loop resistance for the broken loop and separately for the intact loop.

RESPONSE

Part A

Table 8a-1: Lower Plenum, Core, and Upper Plenum Volumes	
	Volume (ft^3)
Lower Plenum	[ ]^{a,c}
Core	[ ]^{a,c}
Upper Plenum Below the Bottom Elevation of the Hot Leg	[ ]^{a,c}

Table 8a-2: Lower Plenum, Core, and Upper Plenum Heights	
	Height (ft)
Lower Plenum	[ ]^{a,c}
Core	12.000
Upper Plenum Below the Bottom Elevation of the Hot Leg	[ ]^{a,c}

Part B

Table 8b-1: Loop Friction and Geometry Pressure Losses from the Core Exit Through the Steam Generators to the Inlet Nozzle of the Reactor Vessel				
	k (dimensionless)	Flow Area (in^2)	0% SGTP Loss Coefficient (ft/gpm^2)	7% SGTP Loss Coefficient (ft/gpm^2)
Upper Plenum to Hot Leg Nozzle	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	Same
Hot Leg Nozzle	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	Same
Hot Leg	N/A	N/A	[ ]^{a,c}	Same
Steam Generator Inlet	N/A	N/A	[ ]^{a,c}	Same
Steam Generator Tubes, Inlet to U-Bend	N/A	N/A	[ ]^{a,c}	[ ]^{a,c}
Steam Generator U-Bend	N/A	N/A	[ ]^{a,c}	[ ]^{a,c}

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Steam Generator Tubes, U-Bend Outlet	N/A	N/A	[ ]a,c	[ ]a,c
Steam Generator Outlet	N/A	N/A	[ ]a,c	Same
Pump Suction Leg	N/A	N/A	[ ]a,c	Same
Cold Leg	N/A	N/A	[ ]a,c	Same
Cold Leg Nozzle	[ ]a,c	[ ]a,c	[ ]a,c	Same
Intact Cold Leg to Broken Cold Leg	[ ]a,c	[ ]a,c	[ ]a,c	Same

Table 8b-2: Locked Rotor Reactor Coolant Pump (RCP) k-factor

	k (dimensionless)	Flow Area (in^2)	0% SGTP Loss Coefficient (ft/gpm^2)	7% SGTP Loss Coefficient (ft/gpm^2)
Locked Rotor (Forward Flow)	N/A	N/A	[ ]a,c	Same
Locked Rotor (Reverse Flow)	N/A	N/A	[ ]a,c	Same

Table 8b-3: Mass flow rates, flow areas, k-factors, and coolant temperatures for the pressure losses provided

	Mass Flow Rate (lbm/hr)	0% SGTP Flow Area (in^2)	7% SGTP Flow Area (in^2)	k-factor (ft/gpm^2)	Coolant Temperature (°F)
Upper Plenum to Hot Leg Nozzle	[ ]a,c	[ ]a,c	Same	[ ]a,c	621.9
Hot Leg Nozzle	[ ]a,c	[ ]a,c	Same	[ ]a,c	621.9
Hot Leg	[ ]a,c	[ ]a,c	Same	[ ]a,c	621.9
Steam Generator Inlet, Plenum	[ ]a,c	[ ]a,c	Same	[ ]a,c	621.9
Steam Generator Tubes, Inlet to U-Bend	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c	586.8
Steam Generator U-Bend	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c	586.8
Steam Generator Tubes, U-Bend Outlet	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c	586.8
Steam Generator Outlet	[ ]a,c	[ ]a,c	Same	[ ]a,c	551.4
Pump Suction Leg	[ ]a,c	[ ]a,c	Same	[ ]a,c	551.4
Cold Leg	[ ]a,c	[ ]a,c	Same	[ ]a,c	551.7
Cold Leg Nozzle	[ ]a,c	[ ]a,c	Same	[ ]a,c	551.7
Intact Cold Leg to Broken Cold Leg	Not Modeled	[ ]a,c	Same	[ ]a,c	551.7

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RAI Question 9

Provide the following elevation data.

a) Bottom elevation of the suction leg horizontal leg piping and cold leg diameter

b) Top elevation of the cold leg at the RCP discharge

c) Top elevation of the core (also height of core)

d) Bottom elevation of the downcomer

RESPONSE

Table 9-1: Elevation Data	
	Elevation (ft) [1]
Bottom of Suction Leg Horizontal Piping	[ ]a,c
Top of Cold Leg at Reactor Coolant Pump Discharge	[ ]a,c
Top of the Core (also Core Height)	[ ]a,c (12.000)
Bottom of the Downcomer	[ ]a,c

(1) All elevations referenced from the bottom of the reactor vessel.

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RAI Questions 11 & 12

Question 11
Discuss whether the Idlechik Handbook recommended expression for pressure loss coefficients along a curved channel was used. If so, explain why it was not used in the calculation for the k-factor. Also, provide the values of the lateral k-factors used for the downcomer lateral flow paths for the plant.

Question 12
Provide the method used to compute the azimuthal lateral k-factors and the values used in the plant calculations. The staff notes that the "Idlechik" reference for calculating k-factors presents a method to compute k-factors in annuli of various radii. Please provide the results of a k-factor study for the lateral flow paths in the downcomer if it was performed.

RESPONSE (11 & 12)

Since these two RAIs are related to each other, they are addressed in a single response.

The total lateral K-factor is made up of two components; the loss due to the curvature in the downcomer (form loss) and the frictional loss. The lateral K-factor resulting from application of Diagram 6-2 from Idelchik (Reference 11-1) is determined, and then compared to the lateral K-factor for a 3-loop PWR. A number of boundary conditions are necessary for this calculation.

1) [

]^{a,c} Also note that the geometry used for the PWR calculation was taken from North Anna Units 1 and 2.

The K-factor from the frictional losses and losses due to the curvature of the downcomer from WCOBRA/TRAC for a 3-loop PWR are presented in Figure 11-1 as a function of Reynolds number. Since the azimuthal flow rate in the downcomer and the fluid properties change significantly throughout a Large Break LOCA transient, it is desirable to compare the losses over a range of Reynolds numbers. [

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]^{a,c} is discussed later in this response.

The total loss calculated from Idelchik (including both frictional and curvature losses) is presented in Figure 11-2 as a function of Reynolds number. It is noted that the loss is higher for low Reynolds numbers in the laminar flow regime, and decreases as the regime transitions to turbulent flow.

The total loss calculated from WCOBRA/TRAC for a 3-loop PWR, the total loss calculated from Idelchik, and the difference between the two losses are presented in Figure 11-3. [

]^{a,c}

The effect of the lower lateral K-factor on mixing in the downcomer is discussed in the response to BE RAI #14.

Reference(s)

11-1) Idelchik, I. E., 1994, "Handbook of Hydraulic Resistance," 3rd Edition, CRC Press, Inc.

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a.c

Figure 11-1: K-Factor for Frictional and Curvature Losses from WCOBRA/TRAC

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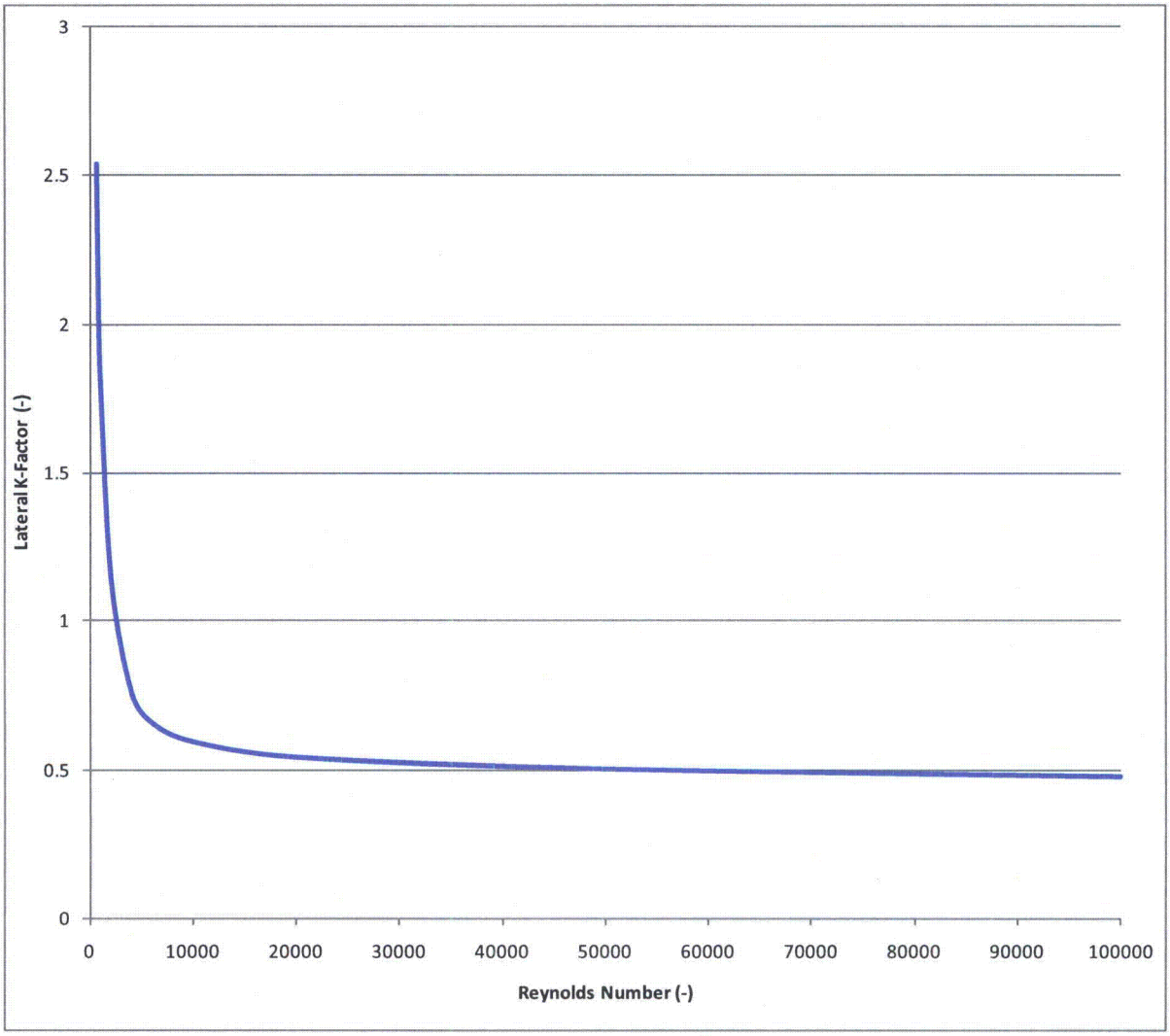

Figure 11-2: Total K-Factor for Losses Calculated from Idelchik

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a.c

Figure 11-3: Comparison of Losses from WCOBRA/TRAC and Idelchik

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RAI Question 13

Describe the azimuthal nodalization and results from the approved best estimate WCOBRA/TRAC model. Provide the results of other nodalization studies applied to the azimuthal detail in the downcomer (other than the three and nine azimuthal node studies). Also show the impact of time step on the PCT for the worst case downcomer boiling calculation.

RESPONSE

The results of the CCTF Test 62, UPTF Test 6, and UPTF Test 25A simulations for the approved CQD methodology are described in Sections 14-2-6-1, 14-4-5 through 14-4-9, and 14-4-11 of the CQD (Reference 13-1), respectively. The results of these same simulations with three downcomer channel stacks per loop (twelve total stacks) are described in Reference 13-2. These test simulations were not executed with any other number of downcomer channel stacks to support the revised downcomer noding in the PWR.

[

]a,c

The North Anna models were only developed using nine (9) downcomer channel stacks. Since the increased number of downcomer channel stacks [ ]a,c, no attempt was made to utilize the coarser downcomer model. As such, analysis results from models with different numbers of downcomer channel stacks are not available.

The impact of time step size on the Peak Cladding Temperature (PCT) calculation was discussed in the response to BE Audit Question #1.

Reference(s)

13-1) Bajorek, S. M., et al., March 1998, "Code Qualification Document for Best Estimate LOCA Analysis," Volume 1 Revision 2, and Volumes 2 through 5, Revision 1, WCAP-12945-P-A (Proprietary).

13-2) Letter from Jensen, J. N. to USNRC, December 27, 2007, "License Amendment Request Regarding Large Break Loss-of-Coolant Accident Analysis Methodology," Enclosure 3, AEP:NRC:7565-01.

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RAI Question 14

The NRC staff completed its sensitivity study on downcomer boiling and the effect of lateral k-factor on this phenomenon. The case with zero lateral k-factor in the downcomer cross flow paths joining the azimuthal cells resulted in a 400 degrees F reduction in PCT. This was due to the maximization of mixing between the downcomer azimuthal cells, which severely limited downcomer boiling. The cold water entering the downcomer during the long term readily mixed into the adjacent downcomer volumes and reduced boiling and the resulting core uncover and clad temperatures. Emergency core coolant bypass and liquid sweep-out that dominate the very early portion of the event (the first 100-200 seconds) does not prevail during the longer term when the downcomer fills with liquid and vapor velocities are no longer high enough to entrain and sweep out the injected liquid. Provide a detailed analysis of impact of the lateral k-factor values on PCT during downcomer boiling following an LBLOCA.

RESPONSE

1. Introduction

Westinghouse has previously stated that [

]a,c

However, the bypass and liquid sweep-out behavior only dominates during the early portion of the transient. The NRC has observed (based on a sensitivity study with a different code) that reduced lateral K-factor can allow for mixing of subcooled liquid in the downcomer which reduces downcomer boiling in the later portion of the LOCA transient.

Westinghouse previously executed a lateral K-factor sensitivity study for an ice condenser plant, as discussed in Reference 14-1. An additional sensitivity study is executed for North Anna. [

]a,c

The results of the lateral K-factor sensitivity study are discussed in Section 2. Some additional discussion regarding the validation of the WCOBRA/TRAC condensation prediction is provided in Section 3, and all references called out in these discussions are cited in Section 4.

2. Sensitivity Study Results Discussion

The discussion of the lateral K-factor sensitivity study results is divided into the short-term impact, and the impact on the later portion of the transient. In the short-term, the primary effect if the lateral K-factor is expected to be on the ECC bypass and sweepout behavior. In the longer term, the NRC has questioned the impact of the lateral K-factor on downcomer boiling.

The increase in the downcomer lateral K-factor for the North Anna Unit 2 limiting transient resulted in an overall 30°F benefit in the peak cladding temperature (PCT). The change to the lateral K-factor had only a minor impact on the calculated short-term bypass and sweepout behavior.

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The NRC noted in this RAI that a sensitivity study with a different code showed that reducing the lateral K-factor in the downcomer promoted mixing of subcooling liquid, thereby reducing the amount of downcomer boiling and correspondingly the PCT. This behavior was not observed in the sensitivity study with WCOBRA/TRAC, for the following reasons.

A significant amount of condensation is calculated in the intact cold legs, such that the safety injection liquid entering the vessel is already well above the injection temperature. A combination of condensation and heat transfer from the metal structures in the downcomer continues to heat the injected liquid. This causes the injected liquid to be near or at the saturation temperature as it reaches the lower elevations of the downcomer. Since the bulk liquid temperature is near or at saturation, the existence or lack of mixing does not impact the calculation of downcomer boiling as was observed in the NRC sensitivity study. As such, the change to the lateral K-factor was not found to significantly impact the calculation of downcomer boiling for WCOBRA/TRAC.

In summary, it has been shown that modeling no lateral K-factor in the downcomer beyond frictional losses tends to produce a similar (albeit slightly more conservative) calculation of the peak cladding temperature with WCOBRA/TRAC for this Large Break LOCA simulation for North Anna.

3. Validation of WCOBRA/TRAC Condensation Prediction

The WCOBRA/TRAC prediction of condensation during reflood is addressed in Section 15-3 of the CQD (Reference 14-2). The degree to which condensation occurs in the cold leg and downcomer is important in calculating the steam flowrate and temperature of the water flowing into the vessel during reflood. If the condensation rate is high, the steam flow will be reduced and the water temperature will be increased. The hotter water will reach saturation and begin to boil sooner in the downcomer, lower plenum, and core. The lower steam flow may entrain less water from the downcomer out the break, and may result in a smaller pressure drop across the broken cold leg nozzle. This in turn will reduce the downcomer pressure (allowing liquid to boil at a lower temperature, and potentially reducing the reflood rate to some small degree). If the condensation rate is low, the colder water will contribute to continued subcooling of the water in the downcomer.

The WCOBRA/TRAC predicted condensation rates were evaluated by comparing the predicted and available fluid temperature measurements for UPTF Tests 8 and 25A, and by calculating an overall condensation efficiency. The predicted liquid temperatures at the exit of the cold leg for both UPTF Test 8 and Test 25A [ ]^{a,c}

Condensation efficiency is defined as the actual condensation rate divided by the potential condensation rate. Analysis of the tests indicates that the condensation efficiency for these large scale tests [ ]^{a,c} (Reference 14-3). The predicted condensation efficiencies for these tests [ ]^{a,c} with the measured efficiency.

4. References

14-1) Letter from Gebbie, J. P. to USNRC, February 24, 2011, "Response to Second Request for Additional Information Regarding a License Amendment Request Associated With the Large-Break Loss-Of-Coolant Accident Analysis Methodology (TAC No. ME1017)," Enclosure 4, AEP-NRC-2011-15.

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14-2) WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.

14-3) MPR-1208, "Summary of Results from the UPTF Cold Leg Flow Regime Separate Effects Tests, Comparison to Previous Scaled Tests, and Application to U.S. Pressurized Water Reactors," October 1992.

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Figure 14-1: Comparison of Sensitivity Study Downcomer Lateral K-factors to Idelchik

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RAI Question 15

Note that the staff will review the results of the applicable small break LOCA (SBLOCA) break spectrum analysis for NAPS in a forthcoming audit activity. This will include the analysis supporting RCP trip timing, which supports the emergency operating procedure for tripping these pumps following a SBLOCA and a description of the methods and identification of the break sizes and limiting location and other pertinent assumptions supporting the RCP trip timing for North Anna.

RESPONSE

The License Amendment Request submitted on October 21, 2010 (Reference 15-8) requested the addition of the Westinghouse ASTRUM methodology documented in WCAP-16009-P-A (Reference 15-9) to the list of methodologies approved for reference in the Core Operating Limits Report in North Anna Technical Specification (TS) 5.6.5.b. The ASTRUM methodology is used for Best-Estimate Large Break LOCA analysis; the methodology is not used for small break LOCA (SBLOCA) analysis. The NRC question on SBLOCA analyses was asked during an NRC audit of the North Anna transition to the Westinghouse RFA-2 fuel product that occurred May 24-26, 2011.

To support the North Anna transition to the Westinghouse RFA-2 fuel product, SBLOCA analysis was performed using the analytical methods in WCAP-10054 (Reference 15-6) and WCAP-10079 (Reference 15-7), which are identified in TS 5.6.5, Core Operating Limits Report, and in Section 15.3.1.2 of the North Anna Updated Final Safety Analysis Report.

The results of the applicable small break loss-of-coolant accident (SBLOCA) break spectrum analysis for the Westinghouse RFA-2 fuel product for North Anna Units 1 and 2 are summarized in Tables 1 and 2. NOTRUMP runs were performed for 1.5, 2, 2.25, 2.5, 2.75, 3, 4, and 5.189 inch equivalent diameter breaks. Table 1 presents the time sequence of events from the NOTRUMP calculations and Table 2 presents the results of the beginning of life (BOL) SBLOCTA fuel cladding heat up calculations. During the fuel transition audit meeting, Nuclear Regulatory Commission (NRC) staff asked for clarification on which reactor coolant system loop seals clear in the analysis for the limiting peak cladding temperature (PCT) case (2.75 inch break). The SBLOCA analysis for the 2.75 inch break only assumes clearing of the broken loop seal, consistent with WCAP-10054 (Reference 15-6). Refer to footnotes (2) and (3) in Table 1.

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Table 1
TIME SEQUENCE OF EVENTS

Event (Sec)	1.5 in	2.0 in	2.25 in	2.5 in	2.75 in	3.0 in	4.0 in	5.189 in
Transient Initiated	0	0	0	0	0	0	0	0
Reactor Trip Signal	76.7	41.3	30	23.4	19.8	16.4	10.1	7.4
Safety Injection Signal	91.8	52.7	40.4	33.4	29.4	25.9	18.6	14.5
Safety Injection Begins(1)	118.8	79.7	67.4	60.4	56.4	52.9	45.6	41.5
Loop Seal Clearing(2)(3)	1788.8	986.9	780.3	573.4	473.1	402.1	233.1	155.4
Top of Core Uncovered	N/A	1261.3	944.3	899.8	606.3	541.1	479.7	228.7
Accumulator Injection	N/A	N/A	2598.3	1974.1	1456	1214.2	648.6	347.1
RWST Low Level	4015.1	3990.6	3974.3	3965.2	3954.6	3948	3930.2	2938.6
Top of Core Recovered	N/A	5148.6	5436.1	N/A(4)	N/A(4)	N/A(4)	3178.7	760.8

(1) Safety Injection is assumed to begin 27.0 s after the Safety Injection Signal.
(2) [ ]a,c
(3) This time is representative of loop seal clearing in the broken loop. [ ]a,c
(4) The core has not fully recovered by the end of the transient but the mixture level is steadily increasing and is nearing the top of the core.

Table 2
SBLOCTA BOL RESULTS

Results	1.5 in	2.0 in	2.25 in	2.5 in	2.75 in	3.0 in	4.0 in	5.189
PCT, °F	N/A(2)	1453.9	1639.9	1719.3	1782.7	1728.4	1386.6	1347.4
PCT Time, sec		2755.4	2579.3	1996.2	1738.3	1504	805.8	404.6
PCT Elevation, ft		11.75	11.75	11.75	12.00	12.00	11.50	11.00
Burst Time(1), sec		N/A	N/A	N/A	N/A	N/A	N/A	N/A
Burst Elevation(1), ft								
HR Maximum ZrO_2, %		0.91	2.19	3.25	3.75	2.58	0.22	0.09
HR Maximum ZrO_2 Elevation, ft		11.75	11.75	11.75	12.00	11.75	11.25	11.25
HR Axial Average ZrO_2, %		0.12	0.29	0.41	0.48	0.35	0.04	0.02

(1) Neither the hot rod nor the hot assembly average rod burst during the BOL SBLOCTA calculations.
(2) The core does not uncover; therefore, SBLOCTA calculations are not warranted for this break size.

SBLOCA Analyses for RCP Trip Timing

Since the SBLOCA event at Three Mile Island Unit 2 on March 28, 1979, operation of the reactor coolant pumps (RCPs) during such transients has been called into question. In the post-accident assessment, it was noted or hypothesized that several aspects of the event timeline affected the final core cooling outcome. Among these aspects was RCP trip. The morning of the event, the RCPs were left in operation for over an hour until cavitation due to highly voided suction conditions was causing severe pump vibration. This raised concerns that reactor coolant system (RCS) integrity could become challenged in that specific area and the pumps were ultimately shut-down. When the last RCPs were tripped approximately 1 hour and 40 minutes into the event, the fluid conditions in the RCS progressed rapidly from a quasi-homogenous saturated state, to a stratified one. This resulted in a significant core uncovery due to a mass shortfall in the RCS.

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The forensics of the event raised questions over this and many other design and operational aspects of pressurized water reactors (PWRs). These concerns prompted the NRC staff to issue many requests for action from the nuclear industry. This included directives for significant operator training improvements, control room staffing requirements, auxiliary feedwater system pedigree, new component design requirements and operation of the RCPs during RCS transients. Specific to operation of the RCPs, among the documents issued were NRC Bulletin 79-06C and Generic Letters 83-10 C and D. These documents presented questions for the need of automated RCP trip and/or improved guidance and operator training to facilitate proper RCP operation under various accident scenarios. Prior to this, Westinghouse had undertaken work to quantify the effects of RCP operation during SBLOCA transients. This work is documented in WCAP-9600 and more notably, WCAP-9584 (References 15-1 and 15-2).

In response to the NRC communications, the Westinghouse Owner's Group (WOG) issued several documents (OG-110 and OG-117, References 15-3 and 15-4) which recommended RCP trip criteria on a generic basis. This work was partially based on analyses presented in WCAP-9584 and supplemented in OG-110 and OG-117. The analysis work itself was based on the WFLASH Evaluation Model (EM). The work in these documents strived to demonstrate the following:

1. The effects of longer term operation of the RCPs during SBLOCAs.
2. What the impacts of such operation may be on the SBLOCA licensing basis analysis.
3. Establish generic RCP trip criteria that could be provided in the emergency response guidelines, thus not necessitating an automated RCP trip under SBLOCA conditions.

Relative to this, the following conclusions were reached:

a) Automated RCP trip under SBLOCA conditions is not required.

b) Three RCP trip criteria were presented that could be utilized by the plant staff in the Emergency Operating Procedures (EOPs). They are as follows:

    1. An absolute RCS pressure with normal uncertainties
    2. Loss of hot leg sub-cooling
    3. Primary-to-secondary pressure differential

c) On a best estimate basis, the RCPs can be tripped at any time with acceptable SBLOCA results.

d) If the RCPs can remain operational throughout the entire small break transient, significant benefits relative to a maximum clad temperature occurs due to enhanced steam cooling. Note that the steam dump system, which is also operational with off-site power available, also provides benefit because of the additional energy removal it provides and is discussed herein.

e) For any given break size, tripping the RCPs after the time in the final safety analysis report (FSAR) calculation when break flow becomes all steam (i.e., breakdown of two phase natural circulation/reflux cooling and progression to loop seal clearing) can make the SBLOCA results worse. The reason being the break flow remains at a low quality, two phase mixture which increases RCS mass loss with respect to time for a given pressure. The two main effects of this are; 1) deeper core uncovery, and 2) reduced total time of uncovery. These two characteristics have opposing effects on PCT giving rise to a maximum function and a worst time interval of RCP trip. It was noted that maximum clad temperature becomes worse for RCP trip during this interval than compared to FSAR type calculations where loss of off-site power is assumed.

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Sometimes these temperatures exceeded 2200°F (Note that these results are based on Appendix K LOCA model assumptions rather than best-estimate).

f) As break size increases, the PCT penalty resulting from delaying the RCP trip decreases or vanishes. As small break size decreases, the PCT penalty can increase.

g) Per WCAP-9584, when considering the spectrum of possible small break sizes, there exists a critical time such that, if RCPs are tripped no later than that time, PCTs will remain below 2200°F for that plant type regardless of the assumed break size. A 10 minute critical RCP trip time was determined for all Westinghouse NSSS designs. This was determined through an extensive analysis performed for the Westinghouse 3-Loop Plant which included many conservative analysis assumptions. In addition, the concept of an equivalent break size was utilized to conclude the critical time for 2-Loop and 4-Loop Plants. Therefore, this critical time can be applied on a generic basis. Note that in follow-up studies performed in Reference 15-4, it was determine that the critical time could in some cases be as low as 5 minutes for a specific break size.

h) If the RCPs are tripped in conformance with the Westinghouse EOP Guidelines, the thermal-hydraulic system behavior and calculated peak clad temperature (or maximum clad temperature, PCT) will be almost identical to the FSAR calculation assuming RCP trip at reactor trip time.

[

]a,c

Again, the main issue with RCPs running is that if the pumps remain in operation too long, additional mass loss from the reactor coolant system can be expected as compared to cases where the RCPs are tripped. This is because the break flow quality can remain relatively low for an additional operational period which will increase mass loss for a given RCS pressure. However, if the RCPs can be tripped before system mass loss is equivalent to the liquid phase inventory that remains after the loop piping has drained (post-loop seal clearing), the SBLOCA transient results are very similar between RCPs operating vs. RCPs tripped. The transient event sequence timing may shift, however, the overall RCS response remains very similar. This is not necessarily based on code results, but rather physics confirmed by the system codes. For a cold leg SBLOCA with loss of off-site power, the RCS response is generally as follows: When RCPs trip due to loss off-site power from turbine trip, the SBLOCA transient will progress to a single phase natural circulation period which maintains core cooling. As saturation is reached due to mass loss and subsequent depressurization, natural circulation will transition to a two-phase state. When over-all mass loss exceeds approximately 40% (note this will vary somewhat depending on break size and decay heat power), the relative velocity between the liquid and vapor phases becomes too great in the

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steam generator (SG) vertical tube runs to support co-current flow. At that point the two-phase mixture circulation breaks down into a counter-current reflux cooling period. As mass loss progresses, the liquid trapped in the RCP suction cross-over legs is purged and venting of the vapor being generated by the core begins. At this time, the amount of liquid inventory that remains in the system following initial loop seal clearing is basically confined to the vessel. [

]^{a,c} Review of the North Anna fuel transition indicates this occurs at approximately 473 sec for the limiting break. As break size increases, this window narrows, however, the time rate of change on RCS pressure becomes larger which reduces break flow, increases emergency core cooling system (ECCS) flow and leads to earlier accumulator injection all of which minimize the duration of core uncovery. Therefore the effect of continued RCP operation becomes diminished. The time rate of change of RCS pressure for a given break size can be impacted by RCPs running depending on when the pumps are tripped and how much mass remains in the system at time of trip. This variation is not extreme though. Again, as long as the RCPs are tripped at a point before loop seal clearing would occur in the loss of off-site power cases, the operation of the pumps up to that time will have little effect on the analysis. These are all physical phenomena which are considered independent of thermal-hydraulic system codes and models. Therefore, regardless of the changes and/or error corrections made to NOTRUMP since 1985, the basis of the WOG generic RCP trip criteria is upheld. That is, for the cases of significance in the SBLOCA analysis, the RCPs will be tripped in a time frame before the transient is adversely impacted. If the trip is not performed in a timely manner with regard to the transient, that is, the criteria exists almost instantaneously to trip the RCPs, the break size is large enough such that the impact of operating RCPs will be minimal because of the significant depressurization that would have occurred and the accompanying benefits of such phenomena. Note that this rationale applies to hot leg breaks as well. That is, until the liquid phase inventory reaches the break elevation, the differences in cases between RCPs in operation and those where off-site power is lost at turbine trip, will essentially be the same. This is due to the break donor quality remaining basically the same during that time frame, i.e., at or near a saturated liquid state. There could be some subtle differences in break flow due to differences in depressurization rates, but these are not considered to cause major differences in system mass loss with respect to time.

Another aspect that must be considered in SBLOCA scenarios where the RCPs are operating is the availability of the steam dump system. If off-site power is available, the main condenser and steam dump system will also be available. Upon reactor trip, the steam dump system will attempt to bring the plant conditions to a no-load T_{ave} value of 547°F (in the case of the North Anna Units). This will have a significant benefit in SBLOCAs since it will reduce RCS pressure beyond what the loss of off-site power case would experience since P_{sat} of 547°F is lower than the main steam safety valve (MSSV) lowest setpoint. This pressure reduction will reduce break flow, increase ECCS flow and allow the accumulator set point to be reached more quickly in the transients. In addition, steam condensed in the SGs by operation of the steam dump system will provide liquid mass back to the vessel. This is considered to be a significant benefit relative to the FSAR transients which assume loss of off-site power.

In conclusion, the WOG RCP trip criteria, since their inception in mid 1980's, remain valid for successful mitigation of SBLOCA transients should one occur in Westinghouse's NSSS design. Since the North Anna Units are part of the plant population represented in the generic studies, no further work is required to support the existing RCP trip criteria on a plant specific basis.

North Anna Units 1 and 2 responded to NRC Generic Letter 85-12 in a letter dated February 14, 1986 (Reference 15-10). The response identified that North Anna Units 1 and 2 would implement RCS subcooling based on wide range hot leg RTDs as the criterion for manually tripping the RCPs and included the justification for this selection. In addition, the NRC Generic Letter 85-12 response identified

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the EOPs that would be revised to implement the RCP trip criterion. The current revisions of EOPs 1/2-E-0, "Reactor Trip or Safety Injection" direct RCP trip based on RCS subcooling at Step 10 and on the Continuous Action Page. This procedure design supports an early manual RCP trip during a SBLOCA after a reactor trip and confirmation of loss of hot leg subcooling. North Anna's operator training program uses the Westinghouse Owner's Group Critical Task Documentation E-1-C as a basis for controlling SBLOCA RCP Trip as a critical task with an acceptance criterion of 5 minutes from loss of subcooling for SBLOCAs smaller than 4.5 inches. The North Anna EOPs and operator training program ensure that RCP trip occurs within the timing requirements of the generic SBLOCA analyses in WCAP-9584 (Reference 15-2) and OG-117 (Reference 15-4), such that the standard SBLOCA UFSAR analyses with an assumption of loss of offsite power remain bounding.

References:

15-1) WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS System," June 1979.
15-2) WCAP-9584, "Analysis of Delayed Reactor Coolant Pump Trip During Small Loss of Coolant Accidents for Westinghouse Nuclear Steam Supply Systems," August 1979.
15-3) OG-110, "Evaluation of Alternate RCP Trip Criteria," September 1983.
15-4) OG-117, "Justification of Manual RCP Trip for Small Break LOCA Events," March 1984.
15-5) GL-85-12, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps'," June 1985.
15-6) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.
15-7) WCAP-10079-P-A, "NOTRUMP – A Nodal Transient Small Break and General Network Code," August 1985.
15-8) Letter from Leslie N. Hartz (Dominion) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed License Amendment Request (LAR), Addition of Analytical Methodology to COLR, Best-Estimate Large Break Loss of Coolant Accident (BE-LBLOCA)," Serial No. 10-575, October 21, 2010.
15-9) WCAP-16009-P-A, Revision 0, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.
15-10) Letter from W. L. Stewart (Virginia Power) to Harold R. Denton (USNRC), "Virginia Electric and Power Company, North Anna Power Station Unit Nos. 1 and 2, Response to Generic Letter 85-12, Automatic Trip of Reactor Coolant Pumps," Serial No. 85-510AA, February 14, 1986.

RAI Question 16

Provide the decay heat multiplier of the limiting LBLOCA and a detailed description of how decay heat is sampled for each LBLOCA that was analyzed.

RESPONSE

The decay heat multiplier corresponds to "Decay Heat" in ASTRUM Topical (WCAP-16009-P-A) Table 1-10, [ ]a,c with mean (μ) and standard deviation (σ) applied as a function of burnup and enrichment according to the ANSI/ANS 5.1-1979 standard. The as-sampled decay heat multiplier for the PCT-limiting case corresponds to -0.08σ and -0.81σ for North Anna Unit 1 and Unit 2, respectively (decay heat below nominal in both cases). Depending on what paperwork the NRC may have been looking at during the audit, as intermediate steps during the sampling process, the intermediate sampling values of 0.9976 and 0.9758 for Unit 1 and Unit 2, respectively, might have been observed.

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The decay heat model is described in ASTRUM Topical Sections 8-2 and 8-4, while Section 8-7 provides information on the associated uncertainties provided by the ANSI/ANS 5.1-1979 standard. The relative contribution to the decay heat from U-235, Pu-239, and U-238 is a function of burnup and enrichment (see Figures 8-1 to 8-3 of WCAP-16009-P-A). As a result, the uncertainty contribution from the decay groups (shown in Table 8-14 of WCAP-16009-P-A) yields an overall uncertainty that varies with burnup and enrichment.

Application of the uncertainty is through the multiple decay groups shown in ASTRUM Topical Table 8-14.

In summary, the code has included all the individual nominal and uncertainty elements for each decay heat contributor as given in the ANSI/ANS standard (as a function of burnup and enrichment), and applies the same sampled sigma to each contributor at each point in time, with the total decay heat being the sum of all contributors. There is not a single time independent percentage multiplier applied to the nominal at all points in time.

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ATTACHMENT 4

WESTINGHOUSE AFFIDAVIT

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CAW-11-3228

August 5, 2011

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: VRA-11-49 P-Attachment, "North Anna Nuclear Power Station Units 1 and 2 – Response to the Request for Additional Information (RAI) from the Nuclear Regulatory Commission (NRC) Related to the May 2011 Audit of the Fuel Transition Project and the Best-Estimate Large Break LOCA (BE LBLOCA) License Amendment Request (LAR) No. 10-575 (TAC Nos. ME4933 and ME4934)" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-11-3228 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Dominion.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-11-3228, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance

Enclosures

CAW-11-3228

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 5th day of August 2011

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Joyce A. Szepessy, Notary Public
Parks Twp., Armstrong County
My Commission Expires April 16, 2013
Member. Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(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 the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, 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.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

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.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390; it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in VRA-11-49 P-Attachment, "North Anna Nuclear Power Station Units 1 and 2 – Response to the Request for Additional Information (RAI) from the Nuclear Regulatory Commission (NRC) Related to the May 2011 Audit of the Fuel Transition Project and the Best-Estimate Large Break LOCA (BE LBLOCA) License Amendment Request (LAR) No. 10-575 (TAC Nos. ME4933 and ME4934)" (Proprietary) for submittal to the Commission, being transmitted by Dominion letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for use by North Anna Nuclear Power Station Units 1 and 2 is expected to be applicable for other licensee submittals in response to certain NRC requirements for fuel transition project submittals and may be used only for that purpose.

This information is part of that which will enable Westinghouse to:

(a) Provide input to the U.S. Nuclear Regulatory Commission for review of the North Anna fuel transition submittals.

(b) Provide results of customer specific calculations.

(c) Provide licensing support for customer submittals.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of the information to its customers for the purpose of meeting NRC requirements for licensing documentation associated with fuel transition submittals.

(b) Westinghouse can sell support and defense of the technology to its customer in licensing process.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

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 information 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.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.