WCAP-16996-P, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)"
Request for Additional Information – (Non-Proprietary)
Set 8 RAIs 127, 132-135 and 137-139

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RAI Question #127: Single Failure Assumptions in Loss-of-Coolant Accident Analyses

Criterion 35, "Emergency Core Cooling," in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 to Code of Federal Regulation Part 50, requires that a single failure be assumed when analyzing safety system performance. The NRC RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989, states that "Appendix A to 10 CFR Part 50 requires that a single failure be considered when analyzing safety system performance and that the analysis consider the effect of using only onsite power and only offsite power."

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.3, "Reactor Accident Boundary Conditions," in its part "Single Failure Assumption," states that "the loss of a train may be assumed for the determination of pumped ECCS flow during the LOCA, while the train will be assumed to operate in the calculation of containment backpressure." It is also explained that "alternatively, a more complete analysis using consistent assumptions may be performed on a plant-by-plant basis."

(1) Please provide a table that lists single failure assumptions considered possible and applicable for the purposes of LOCA analyses using the FSLOCA™ methodology. Please use a separate raw for each combination and formulate the single failure assumptions in two separate columns. Include the single failure assumption applicable for modeling the RCS response with WCOBRA/TRAC-TF2 in the first column and the one applicable for the containment backpressure modeling in the second column. Please consider small and large break LOCA analysis applications. Explain the identified single failure assumptions along with pertinent conditions and supporting considerations.

(2) Subsection 25.3, "Reactor Accident Boundary Conditions," in its part "Single Failure Assumption," states that "a more complete analysis using consistent assumptions may be performed on a plant-by-plant basis." Please explain what such "a more complete analysis," when "performed on a plant-by-plant basis," includes. Identify and describe the "consistent assumptions" that are considered applicable for performing the analysis. Describe the types of plant-specific information that are considered in individual plant analyses. If failure-related assumptions that can be identified and used in analyses on a on a plant-by-plant basis are not included in the response to Item (1) above, please provide a table that describes them. Explain the identified single failure assumptions along with pertinent conditions and supporting considerations.

Response:

Criterion 35, "Emergency Core Cooling," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, requires suitable redundancy in components, features, interconnections, etc. such that the system safety function can be accomplished assuming a single failure. As such, when considering possible single failures, it is assumed that there is no single failure which can cause two independent trains of the Emergency Core Cooling System (ECCS) to be inoperable. Additionally, single failures are only considered for active components.

A list of various single failures that could be of interest to different licensing-basis analyses is typically provided in a plant's Final Safety Analysis Report (FSAR). However, there are only a limited number of active systems credited for mitigating LOCA transients in the FULL SPECTRUM™ LOCA (FSLOCA) evaluation model (EM). [

]a,c

Therefore, many of the possible failures are not of significance for the LOCA analysis. Considering the previously discussed active systems, the resulting list of single failures considered relevant for the generic FSLOCA EM is presented in Table 127-1. The impact of the failure on both the RCS and containment is provided.

Table 127-1: Single-Failures Considered for the FULL SPECTRUM LOCA EM

Failure	RCS Impact	Containment Impact

[

]a,c

[

]a,c

As stated in Section 25.3 of WCAP-16996-P, if additional margin is desired on a plant-specific basis, a more complete analysis using consistent assumptions may be performed. This statement refers to modeling the consistent RCS and containment impacts from a particular failure, rather than a bounding combination of the impacts. [

]^{a,c}

Reference(s)

1) WCAP-16996-P, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2010.

2) LTR-NRC-01-6, "10 CFR 50.46 Annual Notification and Reporting for 2000," March 13, 2001.

RAI Question #132: Steam Generator Decay Heat Removal during Small Break Loss-of-Coolant Accidents

The main steam safety valves (MSSVs) are direct-acting valves (actuated only by pressure) that provide overpressure design protection and backup decay heat removal capability when the steam dumps and secondary atmospheric dump valves cannot be used. Each main steam line has several safety valves with staggered set pressures to provide an increased relieving capacity with an increasing overpressure. As described by T. E. Wierman, et al., "Industry Performance of Relief Valves at U.S. Commercial Nuclear Power Plants through 2007," NUREG/CR-7037, March 2011, the set pressures for these valves are a nominal 1170, 1200, 1210, 1220, and 1230 psig with the highest setpoint being less than 110 percent of the SG design pressure.

WCAP-16996-P/WCAP-16996-NP, Volumes I, II and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.3, "Reactor Accident Boundary Conditions," in summarizing the modeling approach with regard to the SG secondary side boundary conditions, states that [

]^{a,c} Specifically, it is explained that [

]^{a,c}

Please provide additional information related to the modeling of SG secondary conditions in WCOBRA/TRAC-TF2 plant analyses to predict decay heat removal via primary to secondary heat transfer in the SGs during SBLOCAs as follows.

(1) State the method of decay heat removal due to heat transfer from the RCS to the SG secondary sides in SBLOCA analyses with WCOBRA/TRAC-TF2 as credited in the FSLOCA™ methodology. Identify the credited systems, conditions for their operation, and introduced assumptions as applied in modeling decay heat removal via the SGs in SBLOCA analyses.

(2) Describe the approach to determine the set pressures for the MSSVs. In particular, please explain how the implemented approach takes into consideration and models the following factors: (a) uncertainty in the setpoint characteristics of the safety valves, (b) pressure drop from the SG to the safety valves, (c) uncertainty in rated relief capacities of the safety valves, and (d) criteria for using the next highest pressure setpoint. For example, Section 7.2.2, "Westinghouse-Designed Plants," in NUREG-0623 (see B. Sheron, "Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors," NUREG-0623, November 1979), in discussing the system pressure for manual pump trip, describes that the next highest pressure setpoint for the secondary system safety valves is used when the calculated relief flow is greater than 60 percent of the rated valve capacity at the previous pressure setpoint.

(3) Present a table that documents the parameters identified in Item (2) above and used in the demonstration plant small break LOCA analyses for the FSLOCA methodology presented in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 28, "Scoping And Sensitivity Studies," and in Section 31, "Full Spectrum LOCA Demonstration Analysis." Subsection 31.1.1, "Break Area Ranges," identifies only a quantity described as "the lowest MSSV set pressure" and provides its value as [ ]^{b} for the V. C. Summer case analysis.

Response:

(1) The only systems credited to remove decay heat due to heat transfer from the reactor coolant system (RCS) to the steam generator (SG) secondary-side after the reactor trips due to onset of the LOCA transient are [

]^{a,c}

(2) The response to part (2) is separated into the setpoint uncertainties, the relief capacity, and the pressure drop.

<u>Setpoint Uncertainties</u>

[

]^{a,c}

Relief Capacity

[

]a,c for several cases from the V C Summer small break spectrum sensitivity studies (presented in Section 27.1.1.3 of WCAP-16996-P [132-1]) is considered. It is expected that the decay heat removal via the steam generator would be more significant for smaller breaks, since more energy is removed via the break for larger breaks.

[

]a,c

Pressure Drop

The pressure drop from the secondary-side of the SG to the safety valves is [

]a,c There was an information notice on this topic (IN-97-09 [132-2]) issued in 1997, where it was stated that MSSV configurations which have relatively long and relatively small piping between the SG and the valves could cause a full-flow pressure drop as high as 100 psi. [

]a,c

(3) Table 132-1 provides the MSSV parameters for V C Summer, and Table 132-2 provides the MSSV parameters for Beaver Valley Unit 1. Note that only the parameters [ ]^{a,c} as described in part (2) of this response.

Table 132-1: Main Steam Safety Valve Parameters for V C Summer

[				
				]^{b}

Table 132-2: Main Steam Safety Valve Parameters for Beaver Valley Unit 1

[				
				]^{b}

Summary

In summary, the only systems modeled to remove energy from the secondary-side of the steam generators are [

]^{a,c} As such, the modeling of the MSSVs and heat removal from the steam generator secondary within the FULL SPECTRUM LOCA evaluation model is judged to be acceptable.

Reference(s)

132-1) WCAP-16996-P, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2010.

132-2) IN-97-09, "Inadequate Main Steam Safety Valve (MSSV) Setpoints and Performance Issues Associated with Long MSSV Inlet Piping," March 12, 1997.

RAI Question #133: Steam Generator Heat Transfer Modeling

According to Section 3.2.7, "Primary to Secondary Heat Transfer (Not Applicable to Boiling Water Reactors)," in the NRC RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989, "heat transferred between the primary and secondary systems through the steam generators should be considered in the calculation and should be calculated in a best-estimate manner."

(1) Please describe the mechanisms that participate in the primary to secondary heat transfer through the SG heat exchange tubes. Identify factors that can have an impact on the primary to the secondary heat transfer mechanisms during normal plant operation and under LOCA conditions. Include consideration of deposits fouling of SG heat transfer tubes and supporting structures and associated effects on the overall heat resistance and thermal performance degradation.

(2) Please present a table that identifies the correlations used in WCOBRA/TRAC-TF2 to model the heat transfer mechanisms on the outer side of the SG heat transfer tubes. Identify the experimental database for each provided correlation and provide its applicability range. Compare the applicability ranges for the correlations against typical conditions expected during LOCAs. Explain how the factors affecting the heat transfer mechanisms as identified in Item (1) above are accounted for in the implemented heat transfer models. Summarize the technical basis for these models along with the supporting data and analyses. In particular, consider the effect of thermal performance degradation due to deposits fouling.

(3) Please present a table that identifies the correlations used in WCOBRA/TRAC-TF2 to model the heat transfer mechanisms on the inner side of the SG heat transfer tubes. Identify the experimental database for each provided correlation and provide its applicability range. Compare the applicability ranges for the correlations against typical conditions expected during LOCAs. Explain how the factors affecting the heat transfer mechanisms as identified in Item (1) above are accounted for in the implemented heat transfer models. Summarize the technical basis for these models along with the supporting data and analyses. In particular, consider the effect of non-condensable gas on heat transfer inside the SG tubes.

Response

Heat Transfer Correlations

The steam generator (SG) tubes are modeled with an HTSTR component in WCOBRA/TRAC-TF2. The WCOBRA/TRAC-TF2 HTSTR component is identical to the HTSTR component from the base TRAC-P Version 5.4.28 code.

The HTSTR component is described in Section 10.10 of WCAP-16996-P [1], which is largely based on the discussion in Section 4.3 of LA-UR-00-834 [2]. The HTSTR solution as described therein is dependent on the value of several parameter settings for the component. For convenience, the standard values utilized for the parameters described in Section 10.10 of WCAP-16996-P or Section 4.3 of LA-

UR-00-834 for the FULL SPECTRUM™ LOCA (FSLOCA™) Evaluation Model (EM) to model the SG tubes are provided as follows. [

]a,c

The calculation of heat conduction in the HTSTR component is described in Section 2.2.4 of LA-UR-00-910 [3]. There are four numerical calculation options described in this section; the [

]a,c

The wall heat transfer models utilized for the HTSTR component are described in Section 3.8 of LA-UR-00-910. A summary of the different correlations used to determine the heat transfer coefficients for the different regimes is provided in Table 3-7 on pages 3-23 and 3-24 of LA-UR-00-910. All of these regimes are encountered for the SG tubes in PWR analysis except for film boiling (regime 4). Additional information on the wall-to-fluid heat transfer solution is provided in Appendix F.2.1 of LA-UR-00-910 (Appendix F.2.2 does not apply since the core reflood model is not used for the SG tubes).

The set of correlations used to model the heat transfer on both the inside and outside of the steam generator tubes is the same.

Effect of Non-Condensables on Steam Generator Heat Transfer

There is a heat transfer suppression model resulting from non-condensables as described in Section 6.2.11 of WCAP-16996-P. This model will impact the heat transfer if non-condensables are transported to the steam generator tubes. The effect of non-condensable gas on heat transfer inside the steam generator tubes [ ]a,c of the Small Break LOCA (SBLOCA) transient behavior based on the following discussion.

The concern with non-condensable gas disrupting decay heat removal through the steam generator under SBLOCA conditions was addressed at least as far back as the late 1970's. This concern was studied in WCAP-9600 [4] and it was [

]a,c

The same situation is present for the FSLOCA EM. [

]a,c

[

]a,c

Effect of Fouling on Steam Generator Heat Transfer

The steam generators are modeled [

]a,c

The steam generators play a larger role in decay heat removal for smaller breaks as discussed and illustrated previously in this response. A sensitivity study was executed with the V C Summer 2-inch break case to illustrate the impact of SG fouling. [

]a,c

Summary

In summary, the heat transfer across the steam generator tubes is modeled using accepted correlations consistent with those in TRAC-P Version 5.4.28 code. The impact of [ ]^{a,c} Overall, the modeling of the heat transfer across the steam generator tubes is judged to be acceptable for the FULL SPECTRUM LOCA evaluation model.

Reference(s)

1) WCAP-16996-P, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2010.

2) LA-UR-00-834, "TRAC-M/FORTRAN 90 (Version 3.0) User's Manual," Steinke, R. G., et al., February 2000.

3) LA-UR-00-910, "TRAC-M/FORTRAN 90 (Version 3.0) Theory Manual," Spore, J. W., et al., July 2000.

4) WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS System," June 1979.

a,c

Figure 133-1: Void Fraction in the Bottom Cell of the Accumulator for the V C Summer 2-inch, 3-inch, 4-inch, and 5-inch Break Transients

a,c

Figure 133-2: Partial Pressure of Non-Condensable Gas in the Steam Generator for the V C Summer 2-inch, 3-inch, 4-inch, and 5-inch Break Transients

a,c

Figure 133-3: Steam Generator Secondary-Side Pressure for the V C Summer 2-inch, 3-inch, 4-inch, and 5-inch Break Transients

a,c

Figure 133-4: Liquid Heat Transfer Coefficient on the Outside of the Steam Generator Tubes for the V C Summer 2-inch Break Steam Generator Fouling Sensitivity Study

a,c

Figure 133-5: Steam Generator Secondary-Side Pressure for the V C Summer 2-inch Break Steam Generator Fouling Sensitivity Study

a,c

Figure 133-6: Pressurizer Pressure for the V C Summer 2-inch Break Steam Generator Fouling Sensitivity Study

a,c

Figure 133-7: Vessel Fluid Inventory for the V C Summer 2-inch Break Steam Generator Fouling Sensitivity Study

RAI Question #134: Steam Generator Tube Plugging Levels

WCAP-16996-P/WCAP-16996-NP, Volumes I, II and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.1, "Plant Physical Configuration," in summarizing the modeling approach with regard to SG tube plugging, states that [

]a,c

WCAP-16996-P/WCAP-16996-NP, Volumes I, II and III, Revision 0, Section 28, "Scoping and Sensitivity Studies," Subsection 28.1.6, "Steam Generator Hydraulics: Tube Plugging – LBLOCA," present results for tube plugging levels of 0 percent, 10 percent, and 20 percent for a nominal double-ended guillotine break demonstration plant analyses for V. C. Summer, a three-loop Westinghouse PWR plant.

Please provide the following additional information regarding the approach to SG tube plugging modeling implemented in the FSLOCA™ methodology for performing LOCA analyses.

(1) Please explain how the [ ]a,c SG tube plugging fraction is established and used in determining relevant boundary conditions for performing LOCA analyses using the FSLOCA methodology.

(2) WCAP-16996-P/WCAP-16996-NP, Volumes I, II and III, Revision 0, Section 28, "Scoping and Sensitivity Studies," referring to [ ]a,c discussed in Section 28.1.6 and in Section 28.2.9 respectively, states that [ ]a,c Please identify and describe the type of plant-specific information that is considered in determining "a plant-specific [ ]a,c for the SG tube plugging used in LOCA and associated uncertainty analyses. In particular, please explain SG design-specific information and SG operational history are taken into account when determining the [ ]a,c SG tube plugging fraction on a plant-specific basis.

(3) Please provide the information identified in Item (2) above and the plant-specific [ ]a,c for the SG tube plugging levels used in the demonstration plant analyses for the FSLOCA methodology presented in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 28, "Scoping And Sensitivity Studies," and in Section 31, "FSLOCA Demonstration Analysis." Describe the introduced assumptions, explain and discuss the applicability of the analysis results, and compare the assessments.

RAI Question #135: SG Tube Plugging Impact on Core Flow Stagnation for Large Break Loss-of-Coolant Accidents

Considering the modeling approach with regard to SG tube plugging, WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 29, "Assessment of Uncertainty Elements," Subsection 29.3.1, "Bounded Parameters," explains that [

]a,c

Recognizing an additional phenomenon of importance for LBLOCA analyses, Subsection 29.3.1, "Bounded Parameters," acknowledges that [

]a,c Specifically, Subsection 29.3.1 explains that [

]a,c

(1) Please identify and describe the deterministic studies of SG tube plugging level that are referred to in the above provided citation from Subsection 29.3.1. Summarize and present results from such analyses that examine the impact of the assumed SG tube plugging level on large break LOCA predictions with a focus on core flow stagnation and possible flow reversal. Include plots that capture such predicted conditions for the core and the hot channel in particular. Analyze and show identified effects of flow stagnation on prediction results taking into consideration applicable safety criteria.

(2) Please identify best-estimate studies that have been performed with WCOBRA/TRAC-TF2 to analyze the effect of SG tube plugging on LBLOCA predictions associated with core flow stagnation and possible flow reversal in the core. Provide a list of references that identify the source documents containing these analyses. In addition to the analyses discussed in Subsection 28.1.6, "Steam Generator Hydraulics: Tube Plugging – LBLOCA," please present results from calculations obtained with WCOBRA/TRAC-TF2 that examine the impact of SG tube plugging on large break LOCA predictions due to core flow stagnation and possible flow reversal in the core.

(3) In presenting the analyses requested in Item (2) above, please provide additional information addressing the following items: (a) the analyses should cover a spectrum of large break sizes over which the examined effect takes place and code predictions show most sensitivity to break change, if so observed, and (b) the analyses should cover the entire range of possible SG tube plugging levels. Please provide consideration of different PWR plant types and show predictions results for a PWR plant design for which the sensitivity to SG tube plugging due to core flow stagnation is expected to be most pronounced, as applicable. Show the impact on prediction results with relevance to the safety criteria and include zoomed plots of computed results over a time window during which the stagnation effect is predicted to occur, as applicable.

RAI Question #137: Steam Generator Tube Plugging Impact on Steam Generator Reversed Heat Transfer for Large Break Loss-of-Coolant Accidents

Considering the modeling approach with regard to SG tube plugging, WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 29, "Assessment of Uncertainty Elements," Subsection 29.3.1, "Bounded Parameters," explains that [

]^{a,c}

Following a large break LOCA, liquid entrainment from the upper plenum through the hot legs and into the SG tube bundles can lead to significant evaporation of entrained liquid due to reversed heat transfer from the hot SG secondary systems to the primary side of the SG tubes. In turn, this process can cause a primary pressure increase thus impacting the core thermal response. Such reversed heat transfer through the SGs can be dependent on the available SG heat transfer area, which is directly affected by the assumed SG tube plugging level.

(1) Please present results from calculations obtained with WCOBRA/TRAC-TF2 that examine the effect of the assumed SG tube plugging level on LBLOCA predictions with a focus on the effect of reversed heat transfer from the hot SG secondary systems and evaporation of entrained liquid on the SG tubes primary side.

(2) In presenting the analyses requested in Item (1) above, please provide additional information addressing the following items: (a) the analyses should cover a spectrum of large break sizes over which the examined effect takes place and code predictions show most sensitivity to break change and (b) the analyses should cover the entire range of possible SG tube plugging levels. Please provide consideration of different PWR plant designs and show predictions results for a PWR plant design for which the sensitivity of primary pressure increase to SG tube plugging is expected to be most pronounced, as applicable. Show the impact on prediction results with relevance to the applicable safety criteria and include zoomed plots of computed results over a time window during which the examined effect is predicted to take place, as applicable.

Response:

1.0 Introduction and Problem Statement

The as-submitted FULL SPECTRUM LOCA™ (FSLOCA™) evaluation model (EM) analyzes [ ]^{a,c} as described in Sections 25.1 and 29.3.1 of WCAP-16996-P [1]. The Nuclear Regulatory Commission (NRC) requested additional information supporting the decision to [ ]^{a,c} The additional information to support the modeling decision is provided in this response.

This response addresses Requests for Additional Information (RAIs) 134, 135, and 137 which are related to the [ ]^{a,c}

2.0 Effects of Steam Generator Tube Plugging

Steam generator tube plugging can influence a number of phenomena during a LOCA transient. The phenomena which are impacted are dependent on the break size. The most important phenomena which could be influenced by the SGTP (regardless of break size) are considered to be the following (based on the Phenomena Identification and Ranking Table, Table 2-1 in WCAP-16996-P):

[

]a,c

The first four phenomena are primarily of significance for larger breaks. The fifth can impact both larger and smaller breaks. The sixth item is of significance for smaller breaks. These different phenomena are discussed in the following subsections. [

]a,c

2.1 Core Flow Stagnation Elevation (During Blowdown)

During the blowdown phase of a Large-Break LOCA (LBLOCA) transient, there are two primary paths from the core to the break. The first is out the bottom of the core, up the downcomer, into the broken cold leg and to the break. The second is up through the top of the core, through the upper plenum into the broken hot leg, around the loop and to the break. Any variations in relative resistance between these two paths will tend to impact the stagnation elevation.

The tube plugging level affects the resistance in the loop, so it will impact the stagnation point during blowdown. [

]a,c

This was shown in Sections 28.1.4.1 (V. C. Summer) and 28.1.4.2 (Beaver Valley Unit 1) of WCAP-16996-P [ ]a,c Please refer to these sections of WCAP-16996-P for additional discussion.

2.2 Mass and Energy Release to Containment

The SGTP level impacts the fluid volume on the primary side of the RCS. A higher tube plugging level reduces the RCS volume, which results in less mass and energy released from the break during the blowdown phase of a LOCA transient.

For smaller breaks, the containment pressure does not have a significant impact on the LOCA transient progression. However, the containment pressure becomes increasingly important for larger breaks. The reduced RCS primary side volume resulting from higher SGTP causes less mass and energy to be released into containment during blowdown, subsequently resulting in a lower containment pressure than for a lower tube plugging level. [

]a,c

2.3 Loop Resistance to Reflood and Steam Venting

[

]a,c

2.4 Steam Generator Heat Transfer Area and Steam Binding

The SGTP level impacts the heat transfer area between the primary side and secondary side of the reactor coolant system (RCS). For larger breaks, this can influence the vaporization of entrained liquid which is carried from the upper plenum into the steam generator (i.e. steam binding). The steam binding further serves to increase the resistance through the steam generator and thus slow the reflood rate.

Cylindrical Core Test Facility (CCTF) Run 62 is a prototypical integral effects test (IET) for LBLOCA. The predicted versus measured cold leg nozzle to upper plenum differential pressure is presented in Figure 19.6-28 of WCAP-16996-P. It is observed that the [

]a,c

A LBLOCA break spectrum study for Beaver Valley Unit 1 was presented in Section 27.1.2.1 of WCAP-16996-P. The average void fraction of the fluid flowing into and out of the steam generator is inspected for the largest and smallest break sizes of the break spectrum. The void fractions for all three loops are presented in Figure 2-1 for the largest break from the spectrum and in Figure 2-2 for the smallest break from the spectrum. [

]a,c

a,c

Figure 2-1: Average Post-Blowdown Void Fraction into and out of the Steam Generators for the Largest Break in the Beaver Valley Unit 1 LBLOCA Break Spectrum Study

a,c

Figure 2-2: Average Post-Blowdown Void Fraction into and out of the Steam Generators for the Smallest Break in the Beaver Valley Unit 1 LBLOCA Break Spectrum Study

3.0 Sensitivity Studies

A sensitivity study on SGTP for SBLOCA was presented in Section 28.2.9 of WCAP-16996-P. It was concluded therein that the [

]a,c

No LBLOCA sensitivities to SGTP were presented in WCAP-16996-P using WCOBRA/TRAC-TF2; however, there are a number of LBLOCA SGTP sensitivity studies available with WCOBRA/TRAC from the Code Qualification Document (CQD) [2] and Automated Statistical Treatment of Uncertainty Method (ASTRUM) [3] evaluation models. The results from a subset of these studies are presented in Table 3-1; the studies were selected to cover a range of different plant designs (i.e., 2-loop, 3-loop, 4-loop) and a range of different tube plugging levels.

Table 3-1: Steam Generator Tube Plugging Sensitivity Study Results with WCOBRA/TRAC

[

]a,c

It is observed from Table 3-1 that there is [

]a,c

The PCT transients associated with these seven LBLOCA SGTP sensitivity studies are presented as Figures 3-1 through 3-7.

a,c

Figure 3-1: Steam Generator Tube Plugging Sensitivity Study for Beaver Valley Unit 1

a,c

Figure 3-2: Steam Generator Tube Plugging Sensitivity Study for Beaver Valley Unit 2

a,c

Figure 3-3: Steam Generator Tube Plugging Sensitivity Study for Diablo Canyon Unit 2

a,c

Figure 3-4: Steam Generator Tube Plugging Sensitivity Study for Indian Point Unit 3 (Solid line is 10% SGTP, Dashed line is 0% SGTP)

a,c

Figure 3-5: Steam Generator Tube Plugging Sensitivity Study for Kewaunee

a,c

Figure 3-6: Steam Generator Tube Plugging Sensitivity Study for Seabrook

a,c

Figure 3-7: Steam Generator Tube Plugging Sensitivity Study for Watts Bar Unit 1

4.0 Summary and Conclusions

It was shown in this RAI response [

]a,c

5.0 Other Pertinent Information

The steam generator tube plugging level is an input into the FSLOCA EM which is provided by the utility. The utility typically assigns a value that accounts for historical rates of plugging as determined and monitored during outages. For current Westinghouse Best-Estimate EMs, the analyzed plugging level is confirmed every reload to ensure that it remains valid. Any intended changes in the plugging levels beyond the analyzed values are evaluated.

The tube plugging levels provided by the utilities for the FSLOCA scoping studies and demonstration analysis are 22% for Beaver Valley Unit 1 and 10% for V. C. Summer.

6.0 Reference(s)

1) WCAP-16996-P, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2010.

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

3) WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment Of Uncertainty Method (ASTRUM)," January 2005.

RAI Question #138: Safety Injection Pump Flow during Loss-of-Coolant Accidents

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.3, "Reactor Accident Boundary Conditions," in summarizing the modeling approach with regard to the SI flow, states that [ ]^{a,c} Specifically, it is stated that "safety injection (SI) flow varies depending on the single failure assumed, and on the specific plant pump and injection line configuration. Current methods, which are also used in currently accepted evaluation models, provide conservative estimates of minimum and maximum flow, which take into account several uncertainties."

Please provide additional information related to the modeling of pumped SI flow in performing LOCA analyses using WCOBRA/TRAC-TF2 as identified below.

(1) Describe the approach to determine the [ ]^{a,c} that are assumed for LOCA analyses on a plant-specific basis. In particular, please identify and describe these "several uncertainties" that are taken into consideration by the "current methods, which are also used in currently accepted evaluation models" in order to provide conservative estimates as stated in Subsection 25.3, "Reactor Accident Boundary Conditions." In addition, please explain how the implemented approach accounts for and models the following factors: (a) uncertainties in safety injection pump characteristics, (b) injection line configuration, and (c) flow resistance and pressure drop along the injection lines.

(2) Please document the parameters identified in Item (1) above and used in the demonstration plant analyses of the FSLOCA™ methodology presented in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 28, "Scoping and Sensitivity Studies," and in Section 31, "FSLOCA Demonstration Analysis." State the assumptions and relevant conditions, as implemented in assessing parameters related to the SI pump flow modeling in the performed LOCA analyses. Include graphs of all pump flow characteristics and tables documenting assessments for other relevant parameters, as appropriate.

Response

(1) The response to RAI #18 provided in LTR-NRC-13-37 [1] on the conservative assumptions / uncertainties the High Head Safety Injection / Charging Pump applies to all the Safety Injection Pumps including the Low Head and Intermediate Head Pumps.

In summary, all Safety Injection (SI) pump performance curves (the High Head Safety Injection / Charging Pump, the low head and the intermediate pump) utilized in the FULL SPECTRUM LOCA (FSLOCA) plant analyses are adjusted to account for uncertainties in measurement of pump developed head and pump flow rate as described in LTR-NRC-13-37.

Additional discussion is provided in this response regarding the assumptions for the injection lines / spilling line configuration and the mini flow (recirculation path around the pump to prevent deadheading) valve treatment.

The assumptions for defining the spilling branch line resistance for the Emergency Core Cooling System (ECCS) LOCA cases are as follows:

- [
-]a,c

The high head safety injection pump mini flow path operation during a LOCA varies from plant to plant as follows:

- Automatic isolation signal results in closure of the mini flow path on receiving a safety injection actuation signal ("S" signal), or
- The mini flow paths are locked open to prevent pump deadheading, or
- The system is configured to be automatically open at lower flow rates and closed at high flow rates.

The intermediate head safety injection pump mini flow path is designed to remain open during the RCS injection phase and is automatically closed when switching to the recirculation phase. The residual heat removal (RHR) mini flow valve is automatically controlled by the flow through the RHR pump (i.e., low head pump) and will open on a low RHR pump flow rate and close on a high RHR pump flow rate.

The fluid system hydraulic modeling of the mini flow path being open or closed is based on the individual plant design and is therefore plant-specific. In the safeguards flow cases, the mini flow path is modeled to use the maximum flow orifice tolerance limit plus plant instrument uncertainties in setting the mini flow path resistance in the model. For the minimum ECCS flow cases utilized in the FSLOCA evaluation model, the mini flow path resistance is minimized to maximize the flow through the mini flow path, thereby reducing the RCS injected flows.

(2) The safety injection flows for Beaver Valley Unit 1 are provided by the customer as input into the FSLOCA analysis. The safety injection flows for V. C. Summer are calculated by Westinghouse as input into the FSLOCA analysis. These V. C. Summer safety injection flows are calculated as described in the response to RAI #18 provided in LTR-NRC-13-37 and as described in Part (1) of this response. The V. C. Summer safety injection flows assuming that the spilling line spills to RCS pressure [ ]a,c are provided in Table 138-1. Flows consistent with the spilling line spilling to containment pressure are provided in Table 138-2. As previously discussed, the [ ]a,c

Table 138-1: V. C. Summer Minimum Cold Leg Injection Flows with Spilling to RCS Pressure

a,b,c

Table 138-2: V. C. Summer Minimum Cold Leg Injection Flows with Spilling to Containment Pressure (Assumed to be 0 psig)

a,b,c

Reference(s)

1) LTR-NRC-13-37, "Submittal of Westinghouse Responses to 'WCAP-16996-P, 'Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)' Request for Additional Information' (Proprietary/Non-Proprietary), Project 700, TAC No. ME5244," June 5, 2013.

RAI Question #139: Asymmetrical Predictions in Modeling of Parallel Flow Configurations

Thermal hydraulic system codes, such as RELAP5, have been found to predict asymmetrical results when modeling parallel flow configurations, usually under low-flow conditions. Recognition of such modeling difficulties is presented by G. W. Johnsen, "RELAP5-3D Development & Application Status," Presentation at the 2002 RELAP5 International User's Seminar, September 4-6, 2001, Park City, Utah. In PWR plant analysis, such flow configurations can be related to parallel flow paths representing the cold legs in the same primary coolant loop of a Combustion Engineering (CE) PWR plant, parallel flow channels representing different azimuthal sections of a reactor vessel downcomer, such representing steam generator secondary side volumes or other regions of the reactor system. A possible solution approach in modeling a simple flow problem between parallel pipes is discussed by D. Lucas, "Recirculating Flow Anomaly Problem Solution Method," Proceedings of 8th International Conference on Nuclear Engineering ICONE8, Paper ID 8479, April 2-6, 2000, Baltimore, Maryland.

Please show that WCOBRA/TRAC-TF2 does not predict anomalous behaviors as described above for other codes when using three- and one-dimensional components. As part of the response, present predictions for an illustrative parallel pipe flow problem as implemented in the RELAP5 dual pipe flow input model presented below.

```
=Flow Anomaly Test Problem
*
*-----------------------------------------------------------------
*crdno        problem type     problem option
0000100          new            transnt
*-----------------------------------------------------------------
*crdno        input units        output units
0000102         british            british
*-----------------------------------------------------------------
*crdno   time 1    time 2
0000105  10.       40.    10000.
*-----------------------------------------------------------------
0000110  nitrogen
*-----------------------------------------------------------------
*crdno end time  min dt   max dt  control  minor ed  major ed  restart
0000201  5000.  1.0e-6  2.0     3   1      250        500
**********************************************************************
**********************************************************************
* minor edit requests
**********************************************************************
**********************************************************************
*
*crdno      variable      parameter
*
301  count   0
302  dt      0
303  dtcrnt  0
304  cputime 0
305  errmax  0
306  emass   0
307  tmass   0
310  mflowj  145010000
```

```
311 mflowj  145020000
312 mflowj  716000000
313 mflowj  711000000
314 mflowj  175010000
315 mflowj  175020000
316 tempf   130010000
317 tempf   160010000
318 cntrlvar 1
319 cntrlvar 2
320 testda  2
321 testda  3
322 testda  4
20800001 testda 2
20800002 testda 3
20800003 testda 4
*
*****************************************************************
*****************************************************************
* hydrodynamic components
*****************************************************************
*****************************************************************
1300000  pmpsuca2       pipe   * loop a2 rc pump suction
1300001  1
1300101   4.2761  1
1300301     25.956   1
1300401     0.0   1
1300601        -90.   1
1300701   -25.956   1
1300801  .00030       0.   1
1301001   00    1
1301201  3  2200.0  550.0    0.0          0.0     0.0     1
*
1450000  clbrcha2      branch
1450001  2       0
1450101  10.0  5.4064     0.    0. -90.0  -5.4064  .00015     0. 00
1450200  3  2200.0   550.0
1451101  160010000 145000000  4.2761  1.0    1.0   0100
1452101  130010000 145000000  4.2761  1.0    1.0   0100
1451201  0.0     0.0      0.0
1452201  0.0     0.0      0.0
*
1600000  pmpsuca1       pipe
1600001  1
1600101   4.2761  1
1600301     25.956   1
1600401     0.0   1
1600601        -90.   1
1600701   -25.956   1
1600801  .00030       0.   1
1601001   00    1
1601201  3 2200.0  550.0  0.0          0.0     0.0     1
```

```
*
1750000  clbrcha1      branch
1750001  2       0
*1750101  10.0  5.4064      0.    0. -90.0  -5.4064 .00015     0. 00
1750101  10.0  5.4064      0.    0. -90.0  -5.4064 .01000     0. 00
1750200  3  2200.0   550.0
1751101  175010000 160000000  4.2761  1.0    1.0   0100
1752101  175010000 130000000  4.2761  1.0    1.0   0100
1751201  0.0      0.0      0.0
1752201  0.0      0.0      0.0
*
*
7100000  lpa1hpit     tmdpvol
7100101  1.0e6   10.0 0.0    0.   -90.0  -10.0   0.     0.    00
7100200   3
7100201     0.      2200.0         90.
*
*
7110000  lpa1hpif   tmdpjun
7110101  710010000  175000000       .0246
7110200   1
7110201   0.0   0.0      0.0      0.0
7110202   10.0  96.0     0.0      0.0
*
*
7150000  lpa2hpit     tmdpvol
7150101  1.0e6  10.0  0.0    0.   -90.0  -10.0   0.     0.    00
7150200   3
7150201     0.      2200.0         550.
*
*
7160000  lpa2hpif   sngljun
7160101  145010000  715000000      10.0    1.0 1.0 0
7160201 0   0.0    0.0      0.0
*
20500100  dtempf  sum  1.0 0.0 1
20500101  0.0  1.0 tempf  160010000  -1.0  tempf  130010000
*
20500200  dtempf  sum  1.0 0.0 1
20500201  0.0  1.0 tempf  130010000  -1.0  tempf  160010000
*
.    * end of input stream
```

Response:

Six different numerical problems are executed with WCOBRA/TRAC-TF2 to address Request for Additional Information (RAI) 139 on the FULL SPECTRUM™ LOCA (FSLOCA™) evaluation model (EM). First, two 3D (vessel) problems are executed similar to the situation presented in (Lucas, 2000) [1]. Next, two 1D problems are executed with parallel pipes. Finally, an additional 3D and 1D case are executed where the boundary condition flow is ramped down to zero. For all cases zero resistance is used in the hydraulic system.

3D (Vessel) Parallel Channel Low Flow, Zero Resistance Numerical Problems

The noding diagram for the vessel parallel channel problems is presented in Figure 139-1 (dummy 1D components required to execute the code were omitted from the diagram). Four channels were used to model the problem. Vessel Sections 1 and 3, which represent the inlet and outlet of the test section, are both modeled with a single channel. Vessel Section 2, which is in between the inlet and outlet regions, is modeled with two identical channels. No loss coefficients were specified for any of the channels at any elevation.

For the first problem, the boundary conditions were specified as shown in Figure 139-1. A constant flow and enthalpy boundary condition was specified at the bottom of the vessel component, with a mass flow rate of 1 lbm/s and an enthalpy of 500 BTU/lbm. A constant pressure and enthalpy boundary condition was specified at the top of the vessel, with a pressure of 2,000 psia and an enthalpy of 500 BTU/lbm. These boundary conditions cause the flow path to be from the bottom of the vessel to the top of the vessel. For the second problem, the boundary conditions were switched such that the flow was from the top of the vessel to the bottom.

The results for the upward flow case are presented in Figure 139-2. It can be seen that the inlet and outlet flow are both 1 lbm/s, as expected. The flow distributes evenly between the two parallel channels, with half the flow travelling upward through each channel. The results for the downward flow case, presented in Figure 139-3, are similar to the upward flow case. The flow at the inlet and the outlet are 1 lbm/s (negative values indicate flow is downward). The flow distributes evenly between the two parallel channels, with half the flow travelling downward through each channel.

The low flow anomaly presented in (Lucas, 2000) is not observed in either of the 3D vessel low flow, zero resistance numerical problems that were investigated.

1D Parallel Pipe Low Flow, Zero Resistance Numerical Problems

The noding diagram for the vessel parallel channel problems is presented in Figure 139-4. Six different components were used to model the problem. Component 1 is a FILL component, which provides a constant flow and enthalpy condition. The mass flow rate is just less than 1 lbm/s, and the enthalpy is 500 BTU/lbm. Component 2 is a TEE component, which connects the FILL component to two parallel PIPE components. Components 3 and 4 are parallel PIPE components. Component 5 is a TEE component which connects the parallel PIPE components and a BREAK component. Component 6 is a BREAK component which provides a constant pressure condition of 2,000 psia. No loss coefficients or friction entries were specified for any cells of any of the components.

Similar to the 3D numerical problems, for the first 1D problem the components were oriented such that the flow was upward vertically through the components. For the second 1D problem the components were inverted such that the flow was downward vertically through the components.

The results for the 1D components case with upward flow are presented in Figure 139-5. It can be seen that the inlet and outlet flow are both just under 1 lbm/s, as expected. The flow distributes evenly between the two parallel pipe components, with half the flow travelling upward through each pipe. The results for the downward flow case, presented in Figure 139-6, are similar to the upward flow case. The flow at the inlet and the outlet are again just under 1 lbm/s (negative values indicate flow is downward). The flow distributes evenly between the two parallel pipe components, with half the flow travelling downward through each pipe.

No anomalous behavior is observed in either of the 1D component low flow, zero resistance numerical problems that were investigated.

3D and 1D Parallel Channel / Pipe Zero Flow Numerical Problems

Since reasonable results were obtained for the low flow, zero resistance numerical problems, an additional 3D vessel and 1D component case was executed where the boundary condition flow was ramped to zero. For both cases, the boundary condition was held constant for 1,000 seconds, then ramped to zero flow over the next 4,000 seconds, and finally held constant at zero for 5,000 seconds until the end of the problem at 10,000 seconds.

The results for the 3D case are presented in Figure 139-7, and for the 1D case in Figure 139-8. It can be seen for the 3D case, the inlet and outlet flow follow the changes in the boundary conditions, and the flow in the parallel channels follow at half the flow of the boundary conditions. Once the boundary condition flow reaches zero, there is limited, anomalous behavior observed. Rather than the system reaching a state of identically zero flow, there are small flow oscillations observed at the cell faces in the system. However, the nature of the observed instability is different from that presented in the Lucas, 2000 paper.

For the 1D case, the inlet and outlet flow follow the changes in the boundary conditions, and the flow in the parallel pipe components follow at half the flow of the boundary conditions. Once the boundary

condition flow reaches zero, the flow at the cell faces in the system also goes to and remains at zero for the duration of the problem.

Conclusions

The 1D component and 3D vessel low flow numerical problems discussed previously in this response produce the expected solutions and do not exhibit the anomalies observed for similar problems using other thermal-hydraulic codes (e.g. Lucas, 2000). The additional 1D component zero flow numerical problem produces the expected result, as the system settles into a constant state of zero flow. The 3D vessel zero flow numerical problem does exhibit limited numerical noise as it does not settle into a constant state of zero flow. However, this is not a concern for the FSLOCA EM since a sustained zero flow condition will not exist during a LOCA transient. As such, the demonstrated WCOBRA/TRAC-TF2 code performance for these numerical problems is judged to be better than the results presented for other thermal-hydraulic codes, and certainly acceptable relative to the calculation of LOCA transients.

Reference(s)

1) Lucas, D. S., "INEEL-Advanced Test Reactor," *Proceedings of ICONE8*, Paper #8479, April 2-6, 2000.

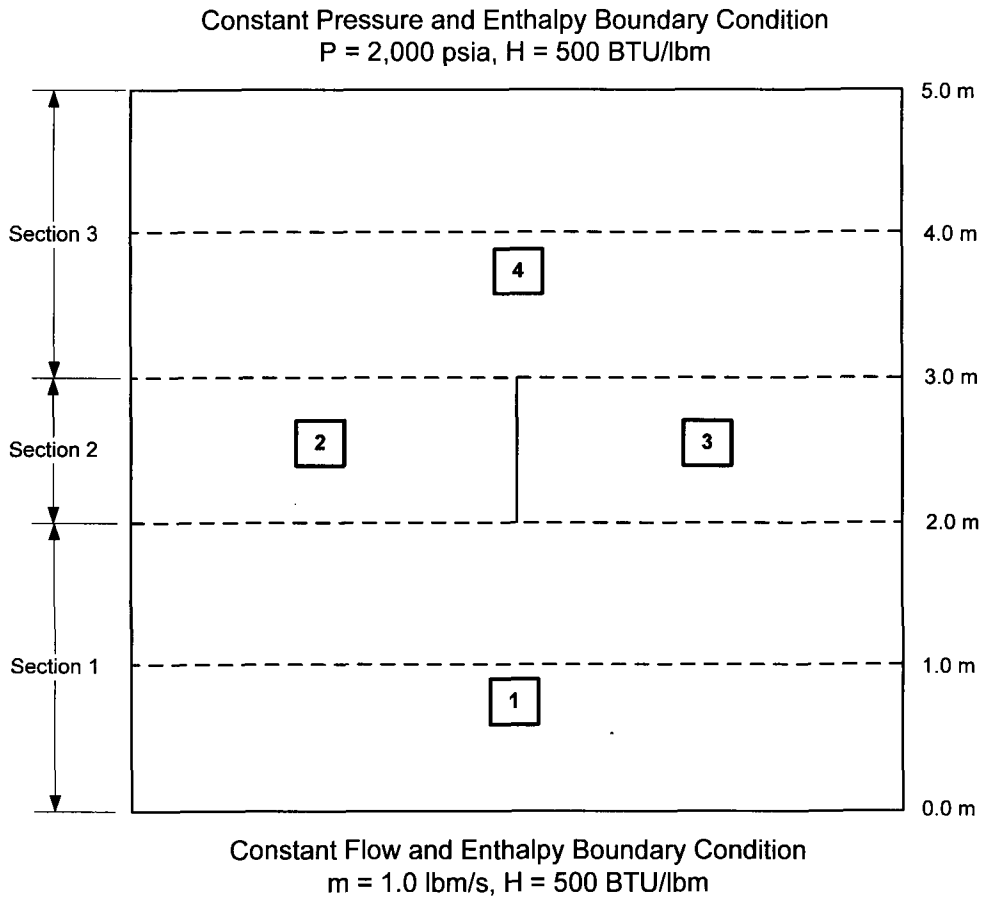

Figure 139-1: Noding Diagram for the 3D (Vessel) Parallel Channel Numerical Problems

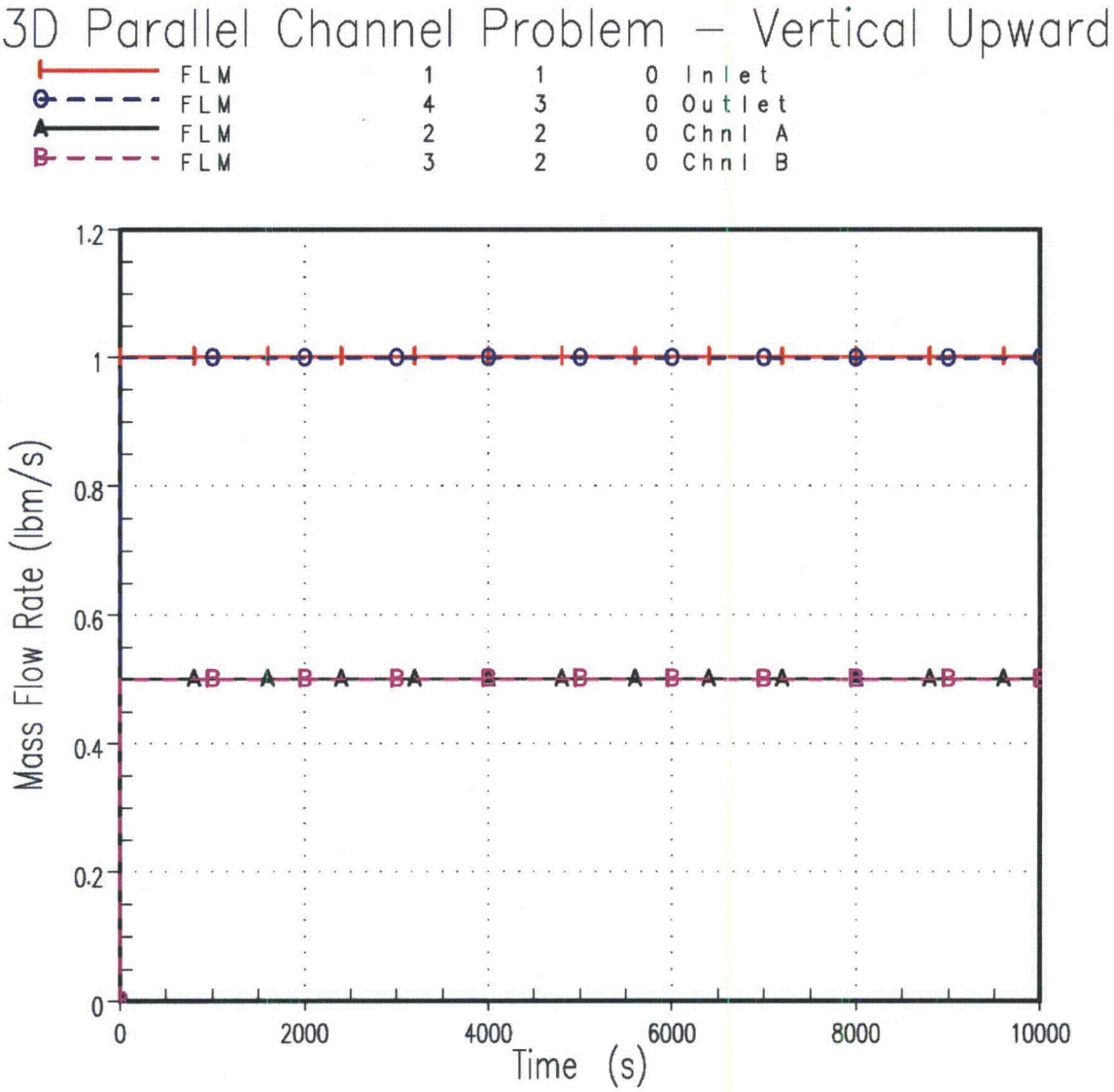

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Figure 139-2: WCOBRA/TRAC-TF2 Results for the 3D Parallel Channel Numerical Problem with Upward Flow through the Vessel Component

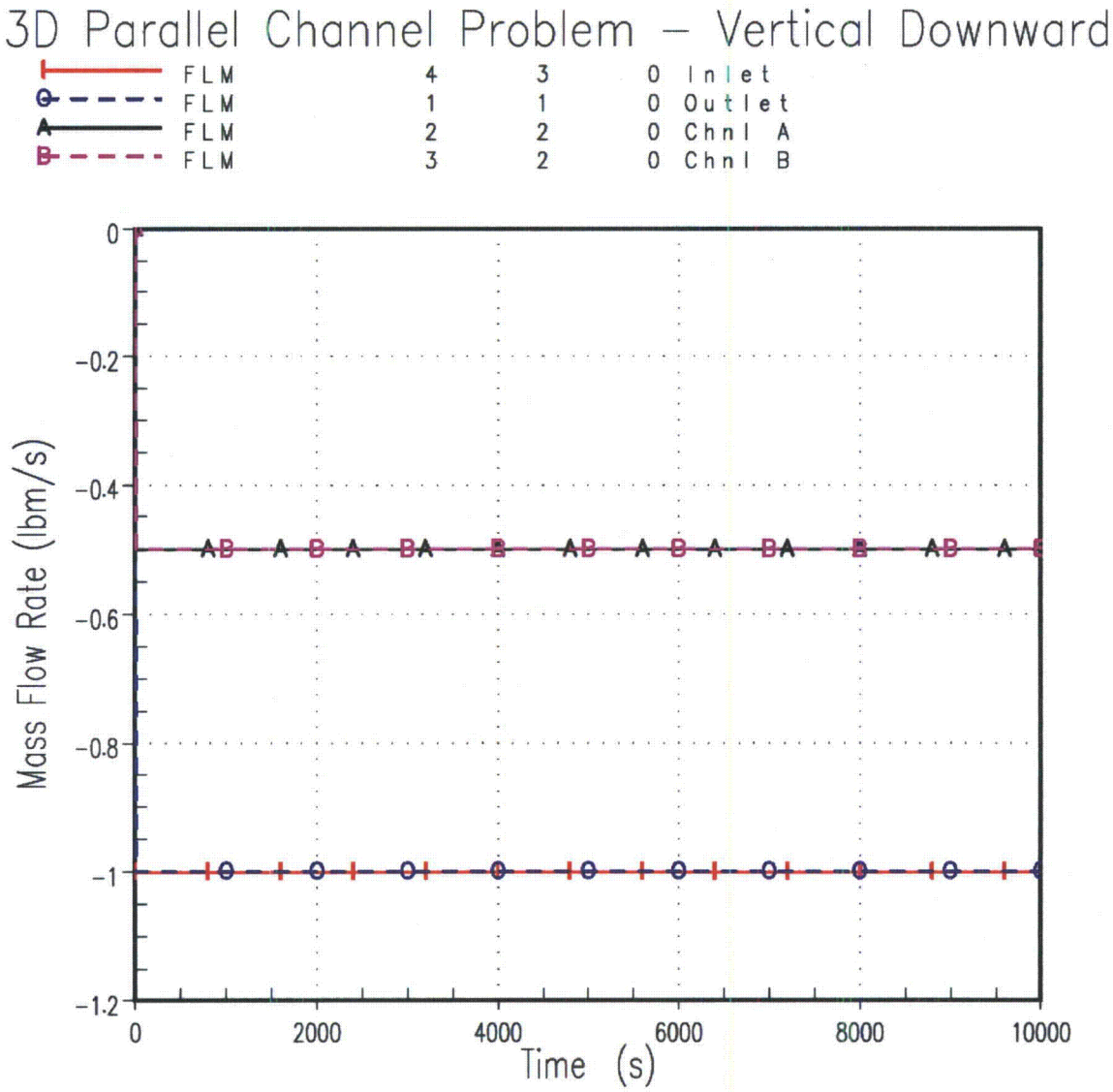

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Figure 139-3: WCOBRA/TRAC-TF2 Results for the 3D Parallel Channel Numerical Problem with Downward Flow through the Vessel Component

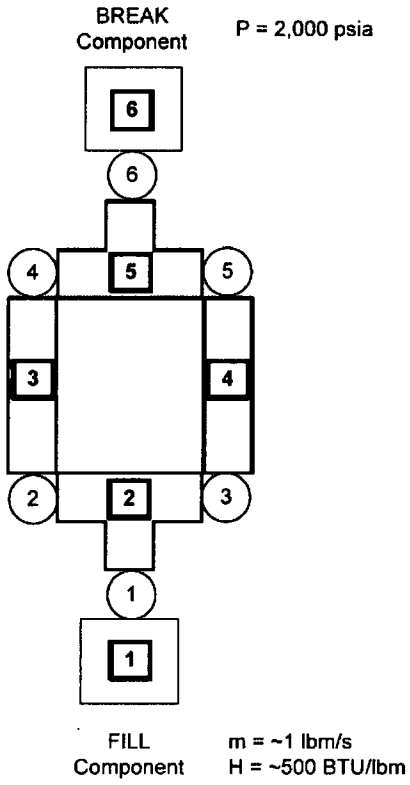

Figure 139-4: Noding Diagram for the 1D Parallel Pipe Numerical Problems

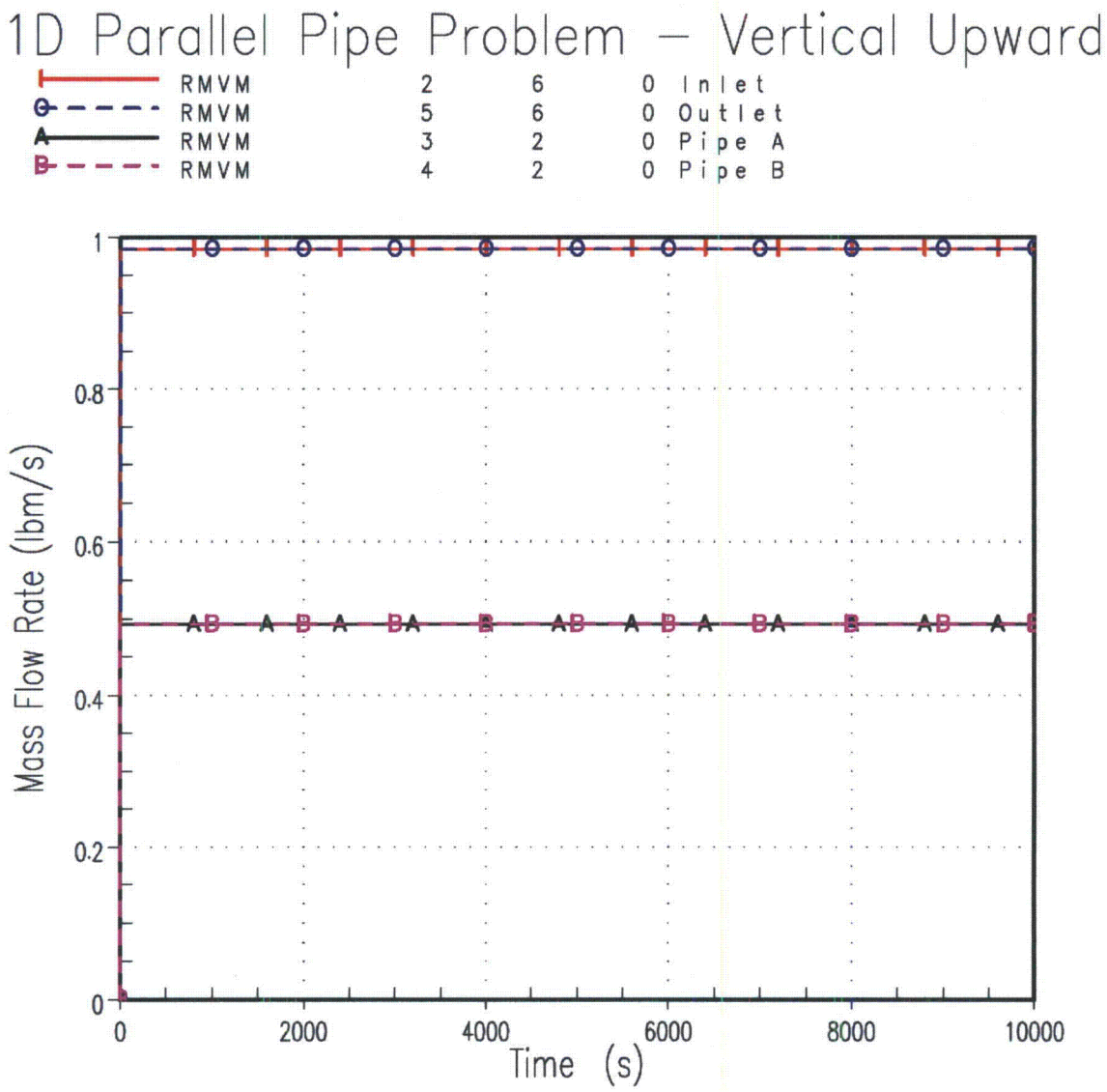

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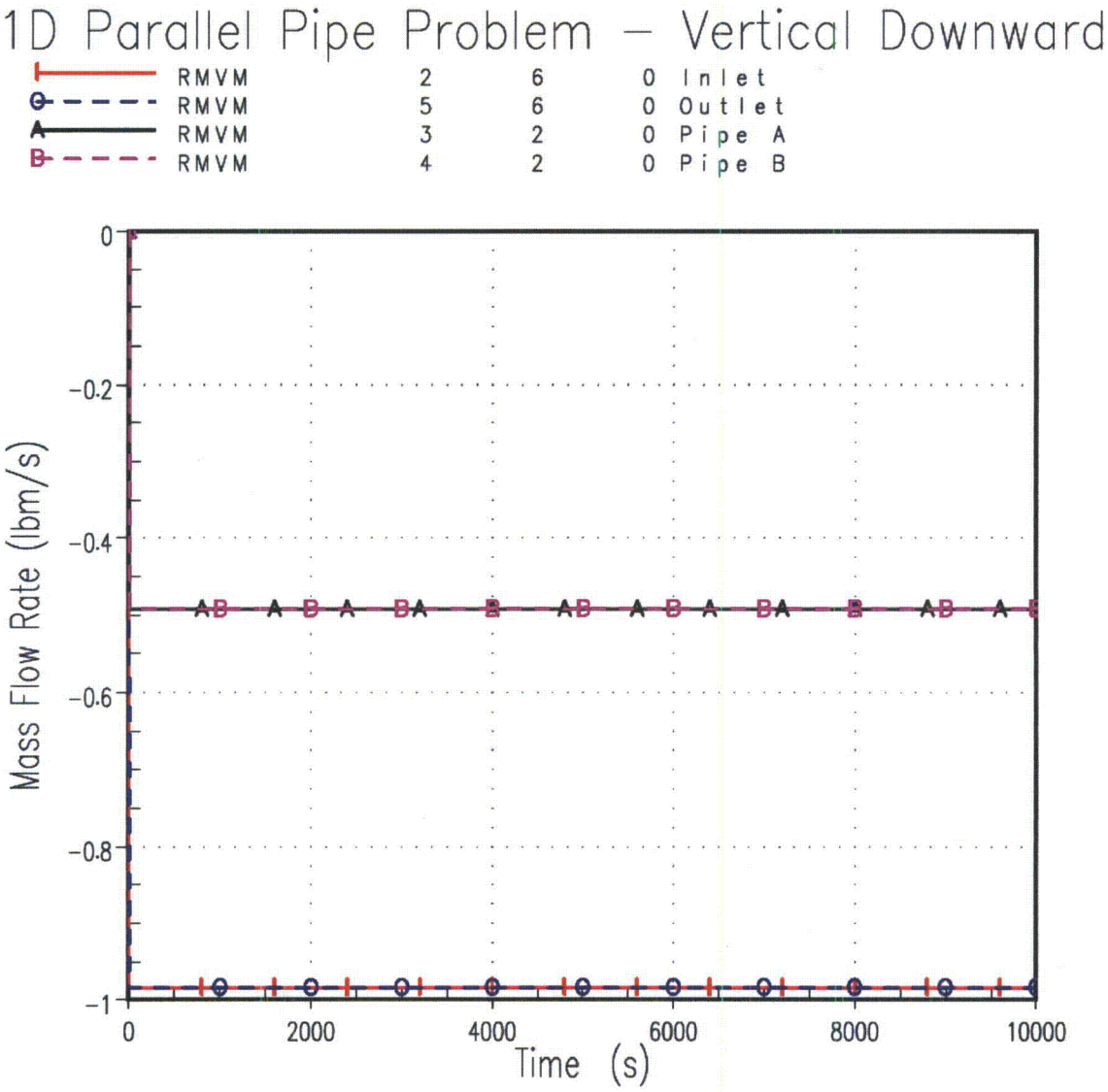
Figure 139-5: WCOBRA/TRAC-TF2 Results for the 1D Parallel Pipe Numerical Problem with Upward Flow through the 1D Components



288929177

Figure 139-6: WCOBRA/TRAC-TF2 Results for the 1D Parallel Pipe Numerical Problem with Downward Flow through the 1D Components

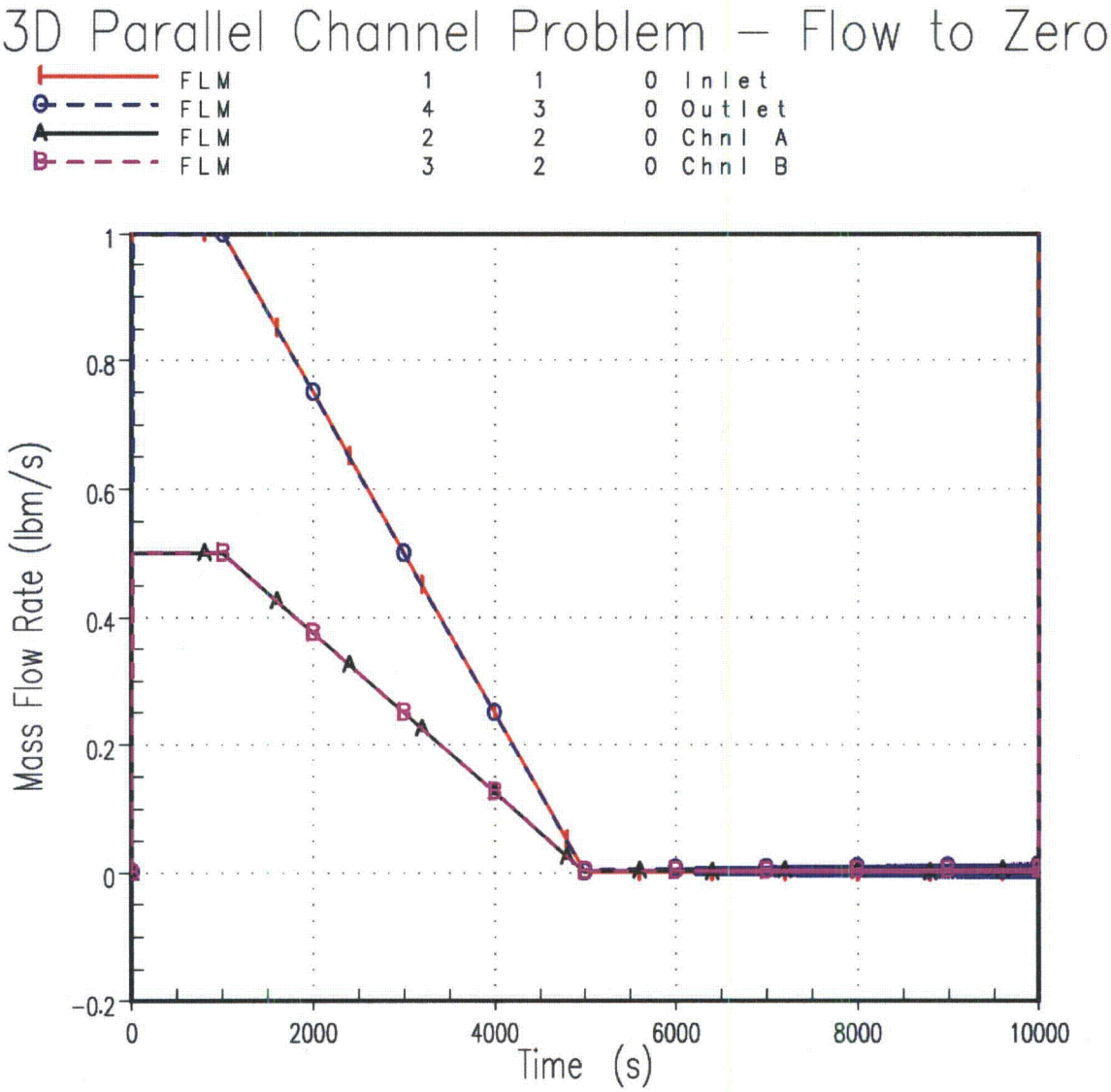


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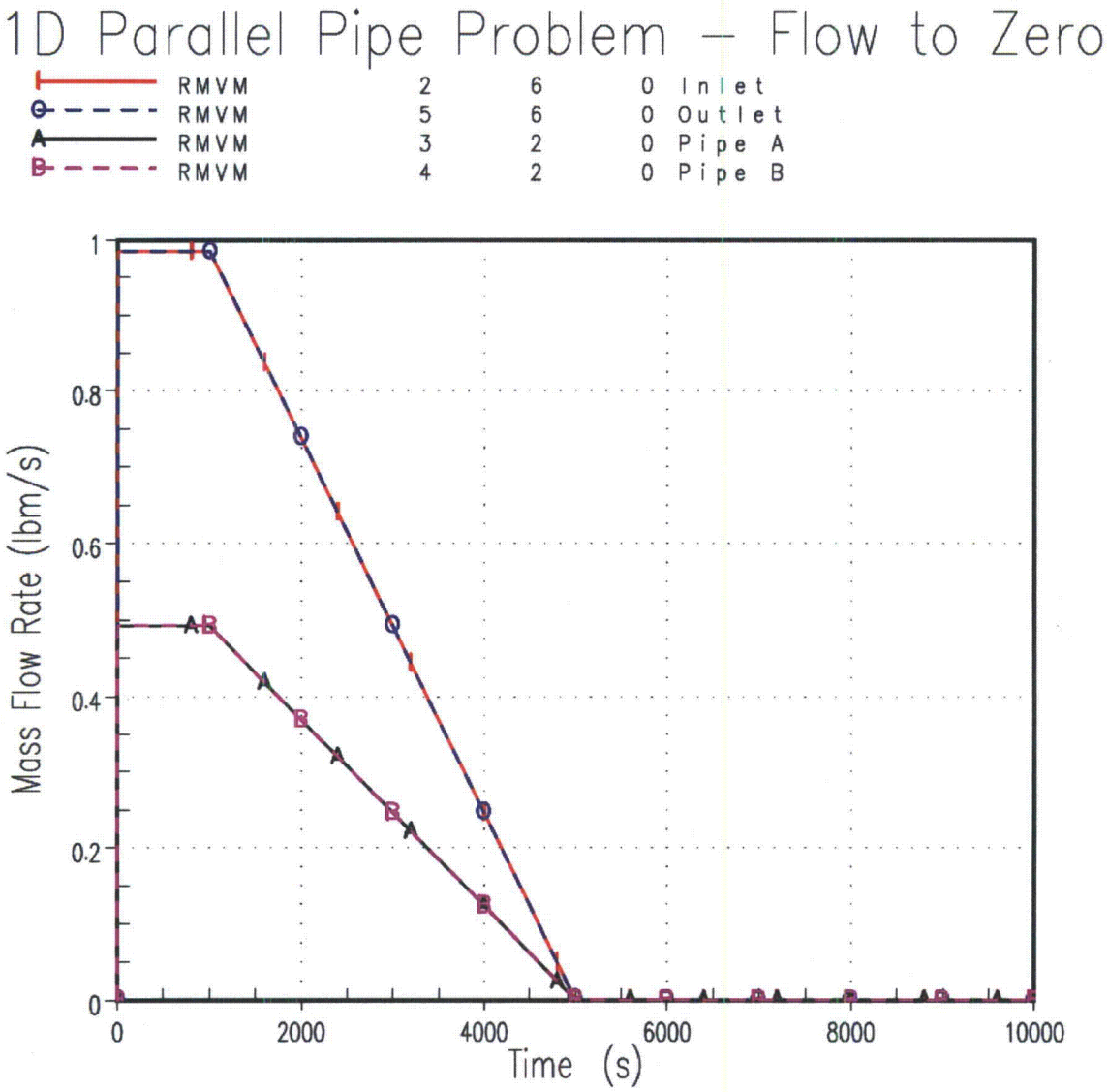
Figure 139-7: WCOBRA/TRAC-TF2 Results for the 3D Vessel Parallel Channel Numerical Problem with Flow Ramped Down to Zero



420174754

Figure 139-8: WCOBRA/TRAC-TF2 Results for the 1D Components Parallel Pipe Numerical Problem with Flow Ramped Down to Zero