AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.173

Question:

Please provide additional information on the simulations reported in Section A.4 on sensitivity studies on upper plenum entrainment rate and interfacial drag. In particular, show:

(a) Axial continuous liquid and steam flow rates into the upper plenum cell(s).

(b) Void fraction in each of the upper plenum cell(s).

(c) Collapsed liquid level in the upper plenum.

(d) Lateral flows (WLM, WEM, and WGM) into each hot leg from the upper plenum.

(e) Axial continuous and entrained liquid flows, and steam flow in the upper plenum at the bottom of the hot legs, and also the entrained drop size.

Westinghouse Response:

The requested plots are provided for the sensitivity study cases in this order:

1. Base Case upper plenum model (Figures 440.173-1 through 20)
2. 1.3 * upper plenum interfacial drag (Figures 440.173-1a through 20a)
3. 0.65 * upper plenum interfacial drag (Figures 440.173-1b through 20b)
4. 0.5 * upper plenum entrainment rate (Figures 440.173-1c through 20c)
5. 2.0 * upper plenum entrainment rate (Figures 440.173-1d through 20d)
6. 4.0 * upper plenum entrainment rate (Figures 440.173-1e through 20e)

The sequence of the plots for each case follows the order listed in the question. The time scales correspond to the scale of the WCAP-15833 Section A.4 figures, and the Base Case model is the three vertical cell upper plenum representation case from Section A.4.

In interpreting the plots, please recall that there are [

]^{a,c} the bottom elevation of the hot legs. The entrained droplet size is provided in the top cells of Channels 15 and 47 and also the bottom cell 2 of Channel 20.

Each of these cases exhibits a liquid circulation pattern in the upper plenum, but the magnitude varies. Both the continuous and entrained liquid fields show a positive flow upward at the hot leg elevation in Channel 15, together with a downward flow of continuous and entrained liquid fields in the Channel 47 junction. In all cases, the magnitude of these liquid flows is greatest at the opening of the first ADS-4 valve, and diminishes thereafter.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Figure 440.173-1: Core Exit Continuous Liquid Flow Rate.
Base Case Upper Plenum Model

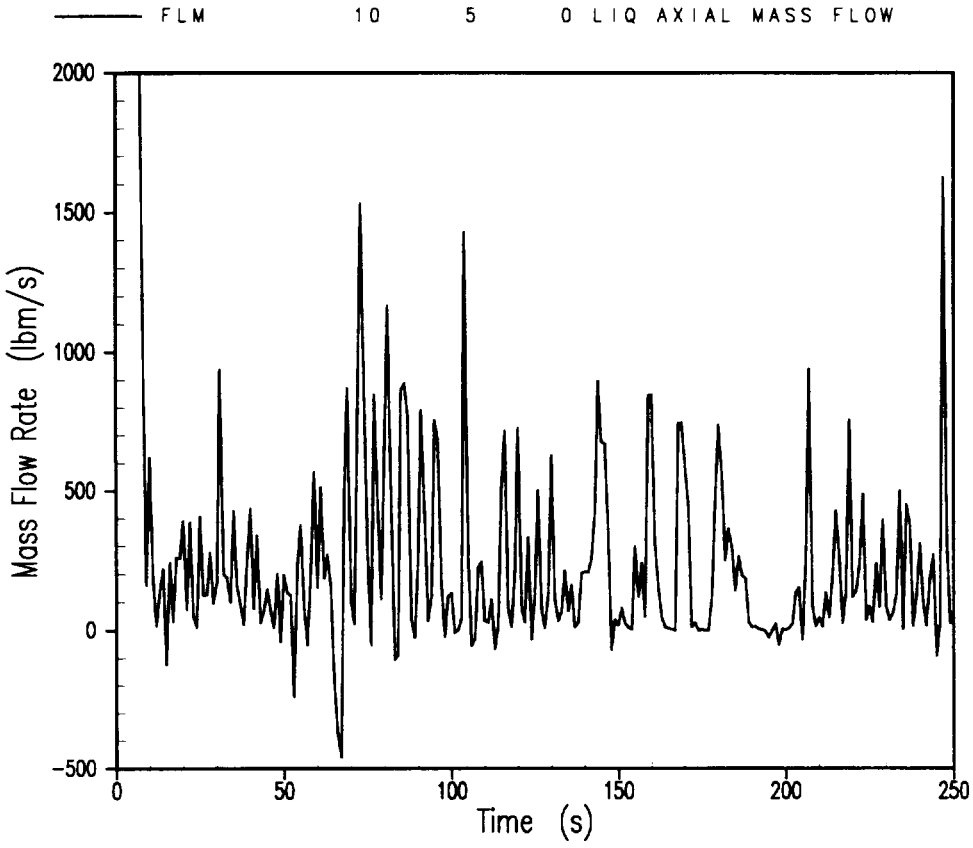


Figure 440.173-2: Core Exit Entrained Liquid Flow Rate.
Base Case Upper Plenum Model

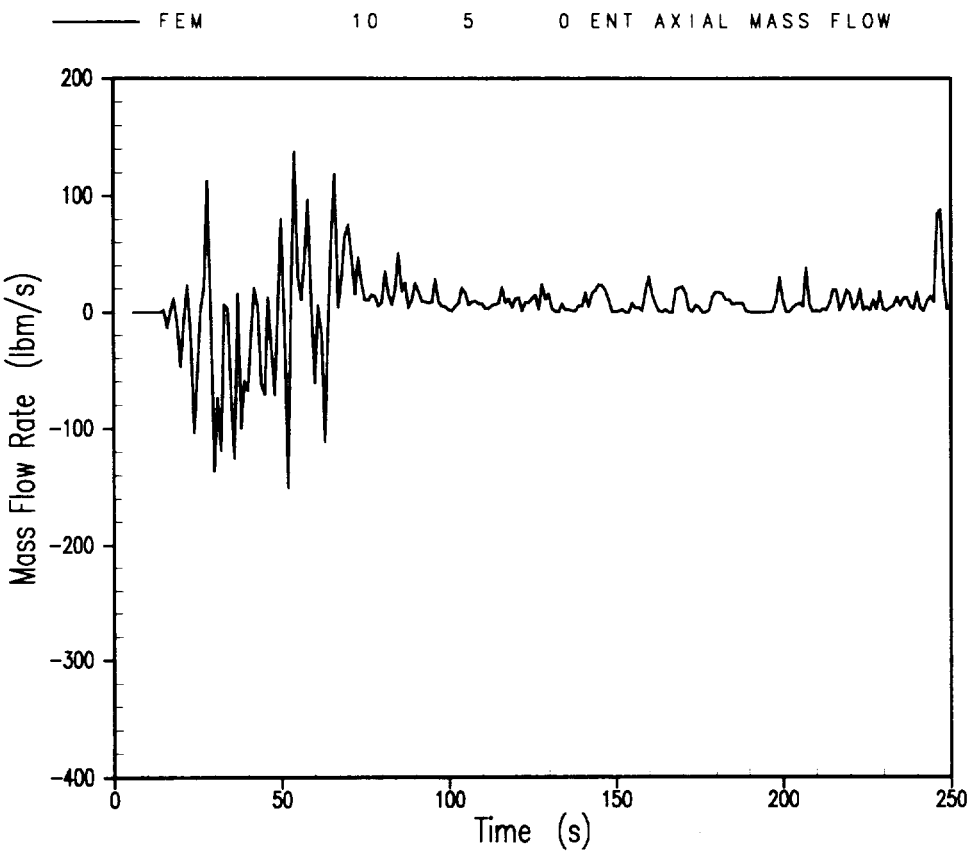


Figure 440.173-3: Core Exit Steam Flow Rate. Base Case Upper Plenum Model

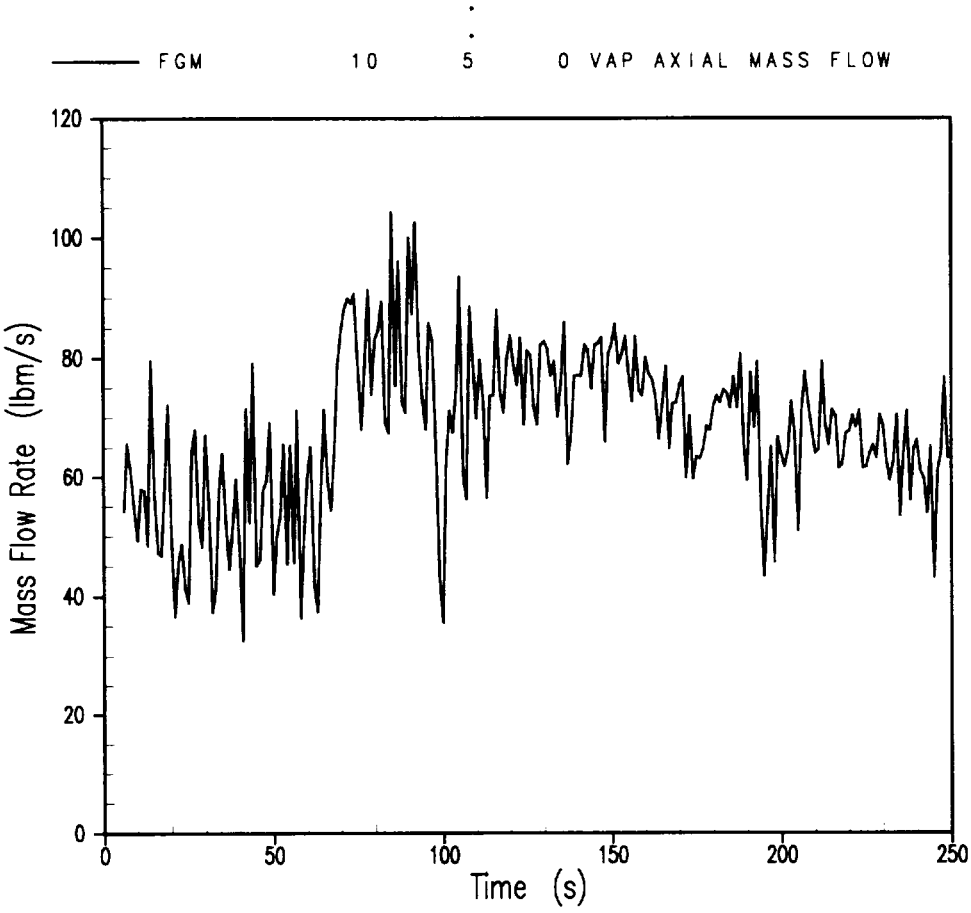


Figure 440.173-4: Upper Plenum Cell Void Fraction,
Base Case Upper Plenum Model

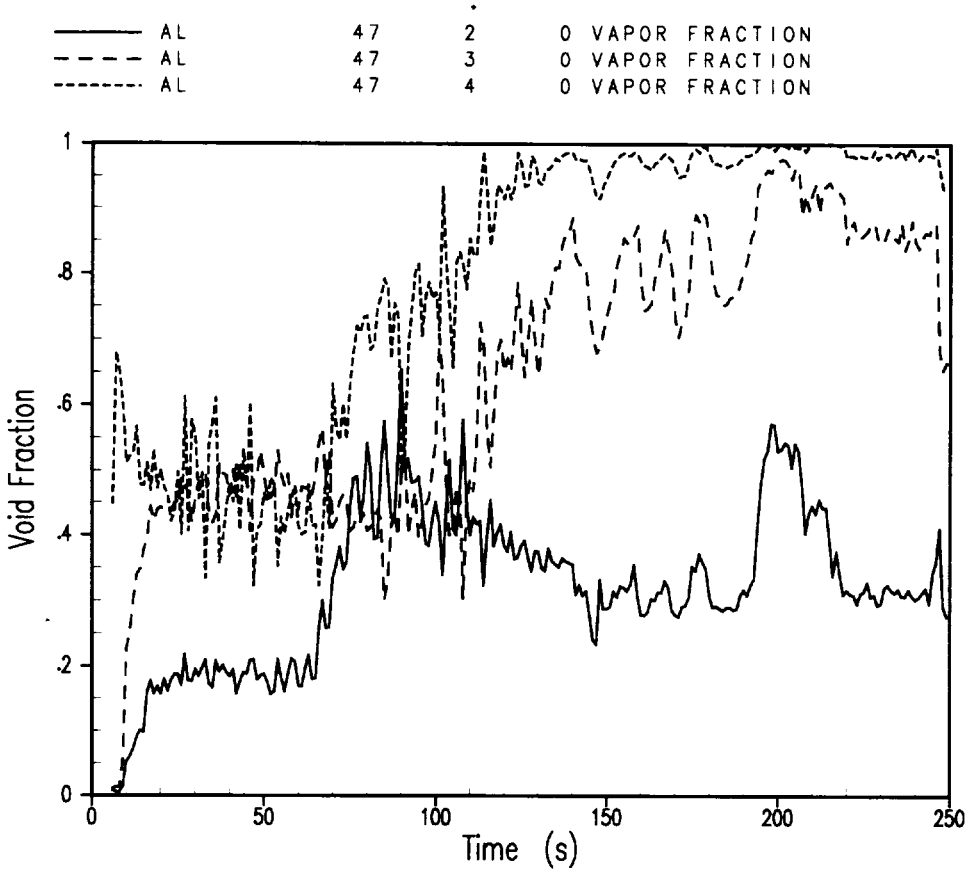


Figure 440.173-5: Upper Plenum Cell Void Fraction.
Base Case Upper Plenum Model

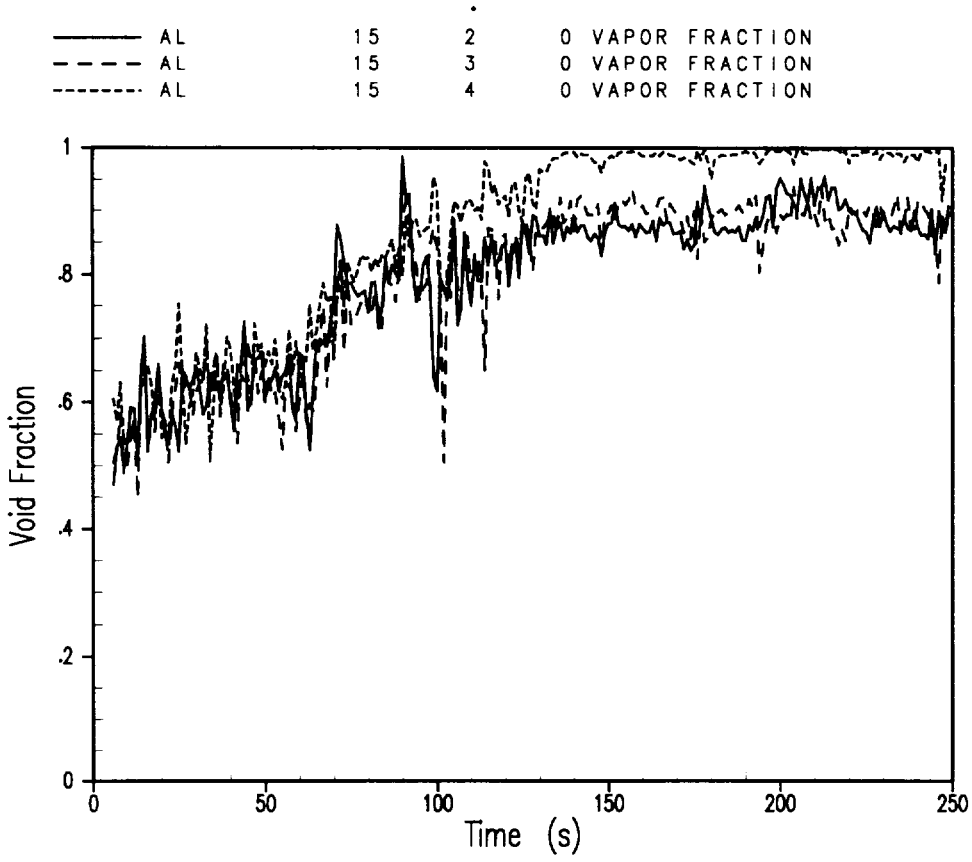


Figure 440.173-6: Upper Plenum Cell Void Fraction, at Hot Leg, Base Case Upper Plenum Model

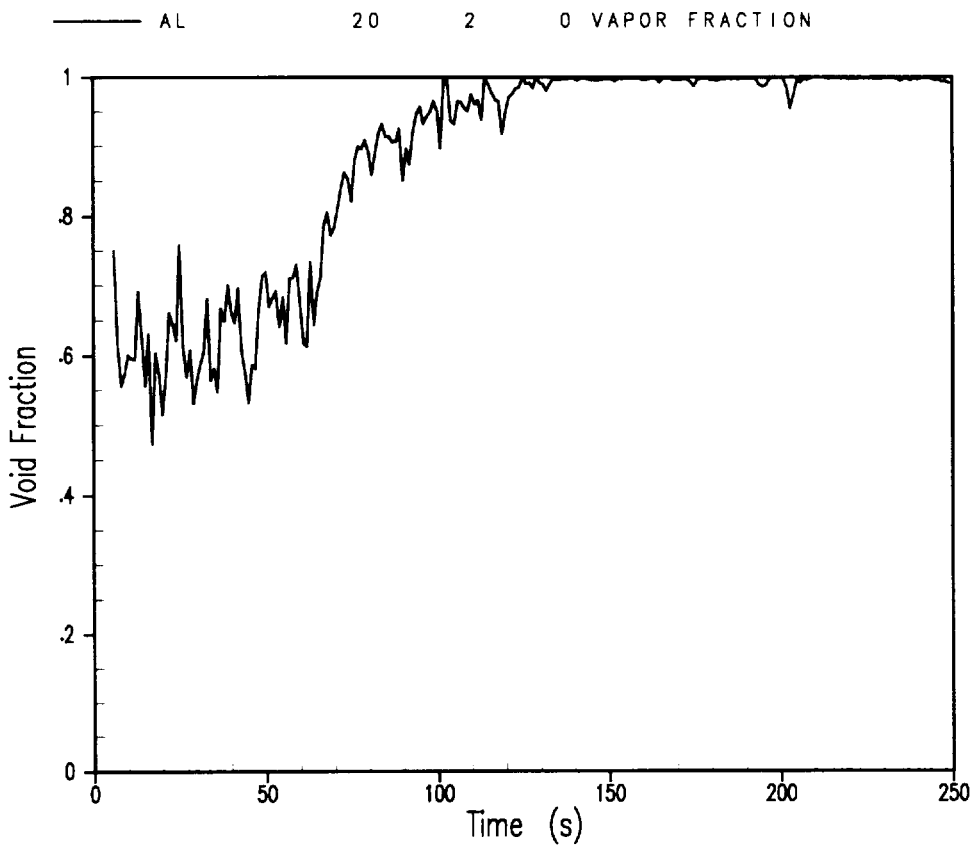


Figure 440.173-7: Upper Plenum Collapsed Liquid Level.

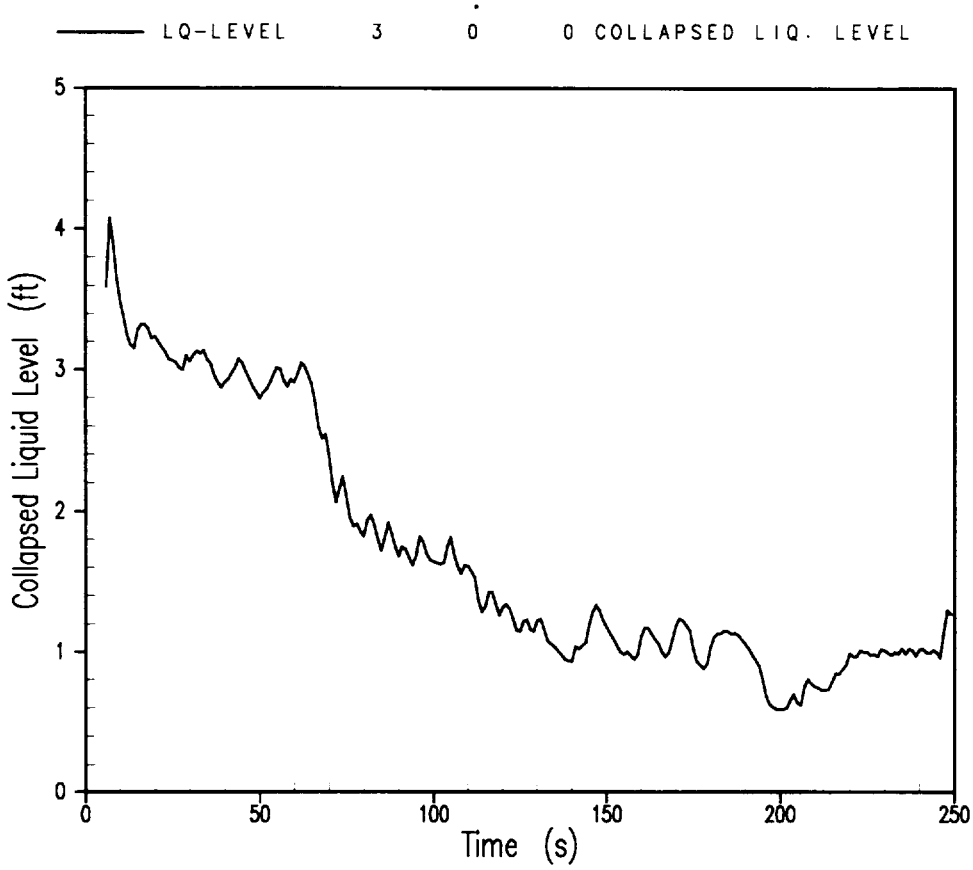


Figure 440.173-8: Continuous Liquid Flow Entering the Pressurizer Loop Hot Leg. Base Case Upper Plenum Model

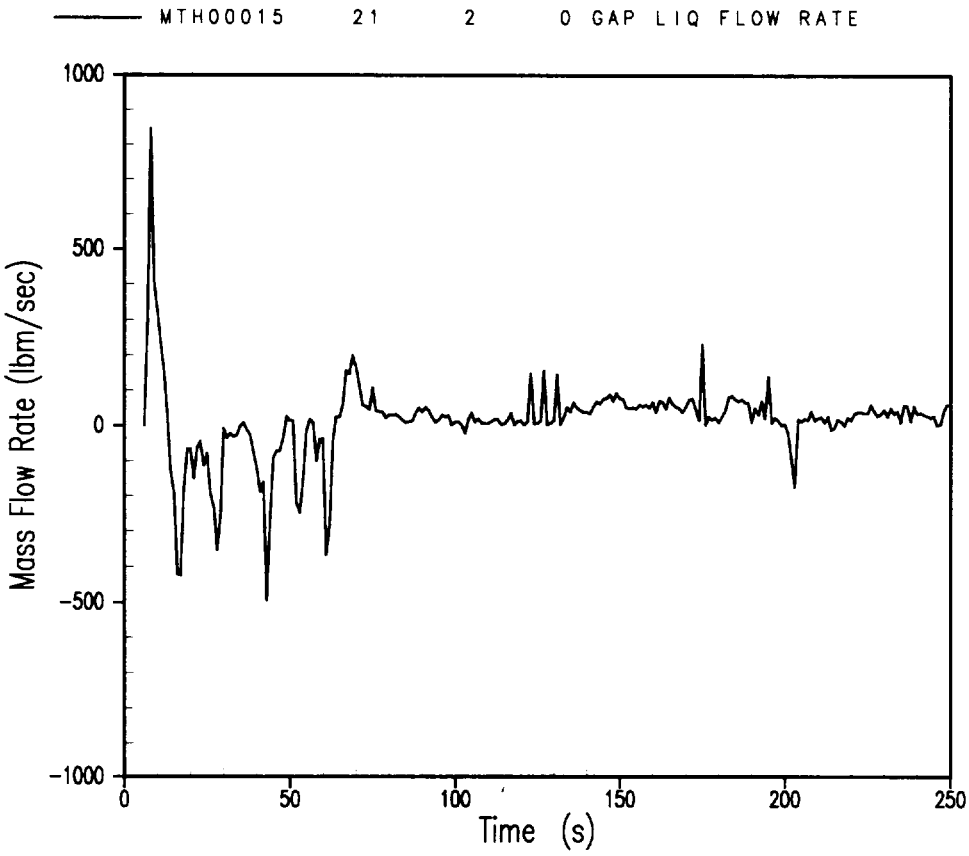


Figure 440.173-9: Entrained Liquid Flow Entering the Pressurizer Loop Hot Leg, Base Case Upper Plenum Model

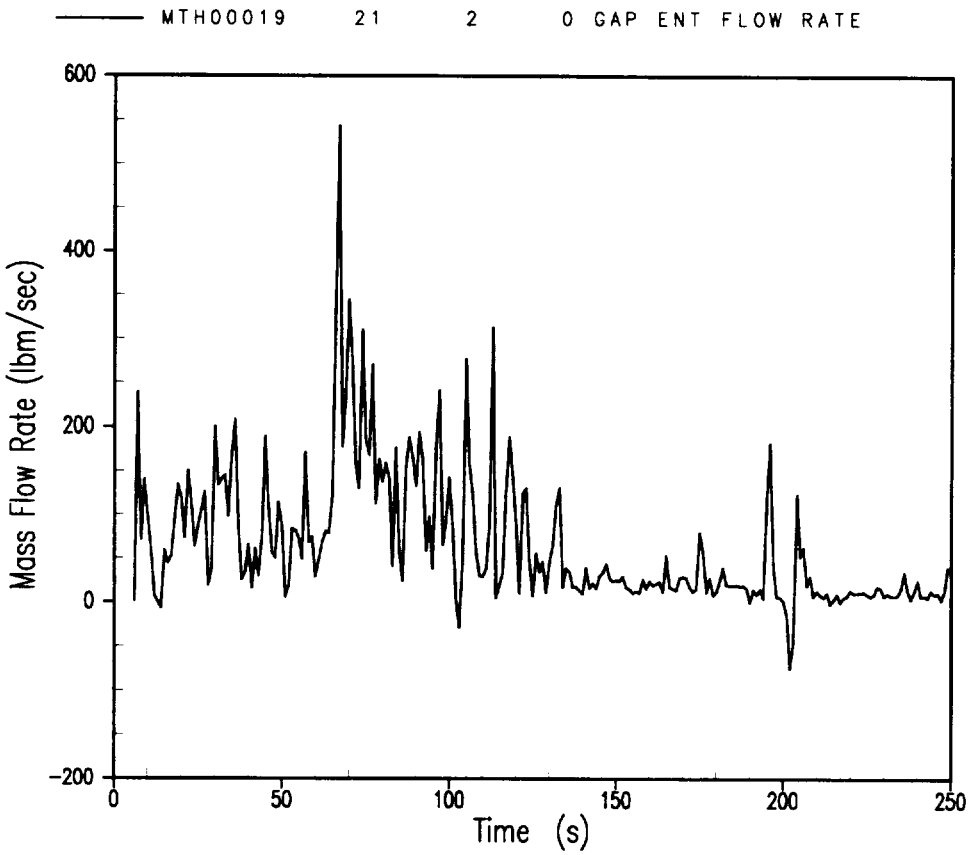


Figure 440.173-10: Vapor Flow Entering the HL Pressurizer Loop
Hot Leg, Base Case Upper Plenum Model

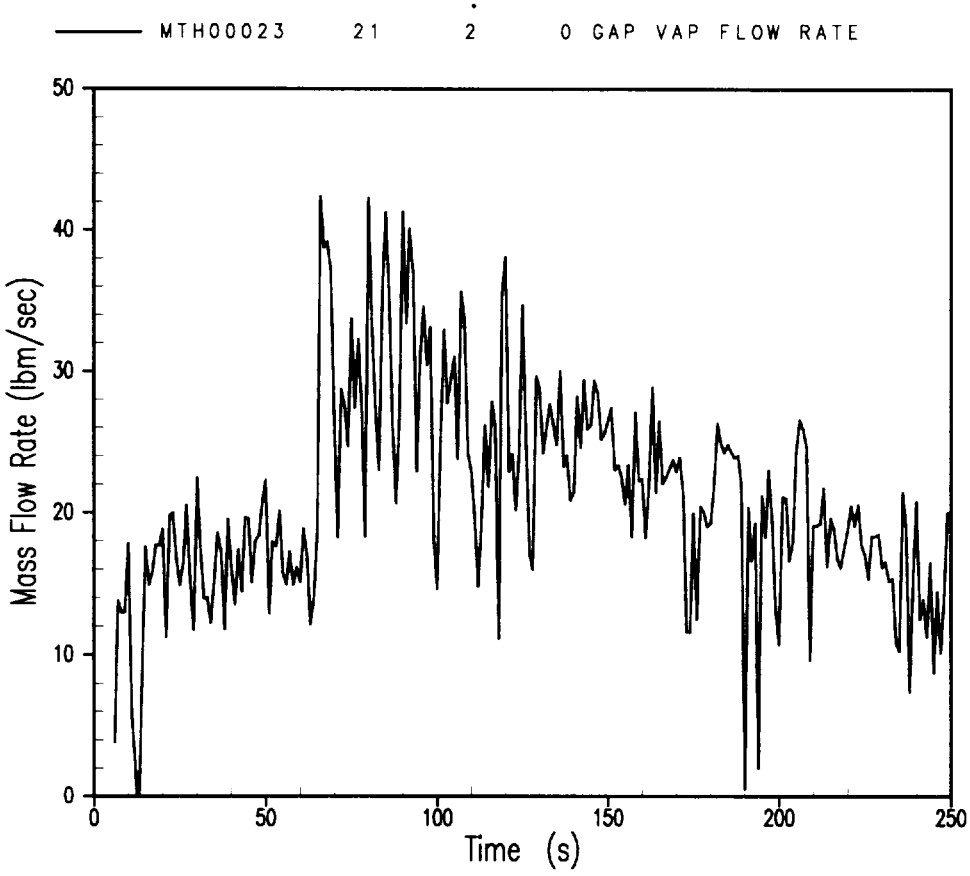


Figure 440.173-11: Continuous Liquid Flow Entering the 2*ADS-4 Flow Path Hot Leg. Base Case Upper Plenum Model

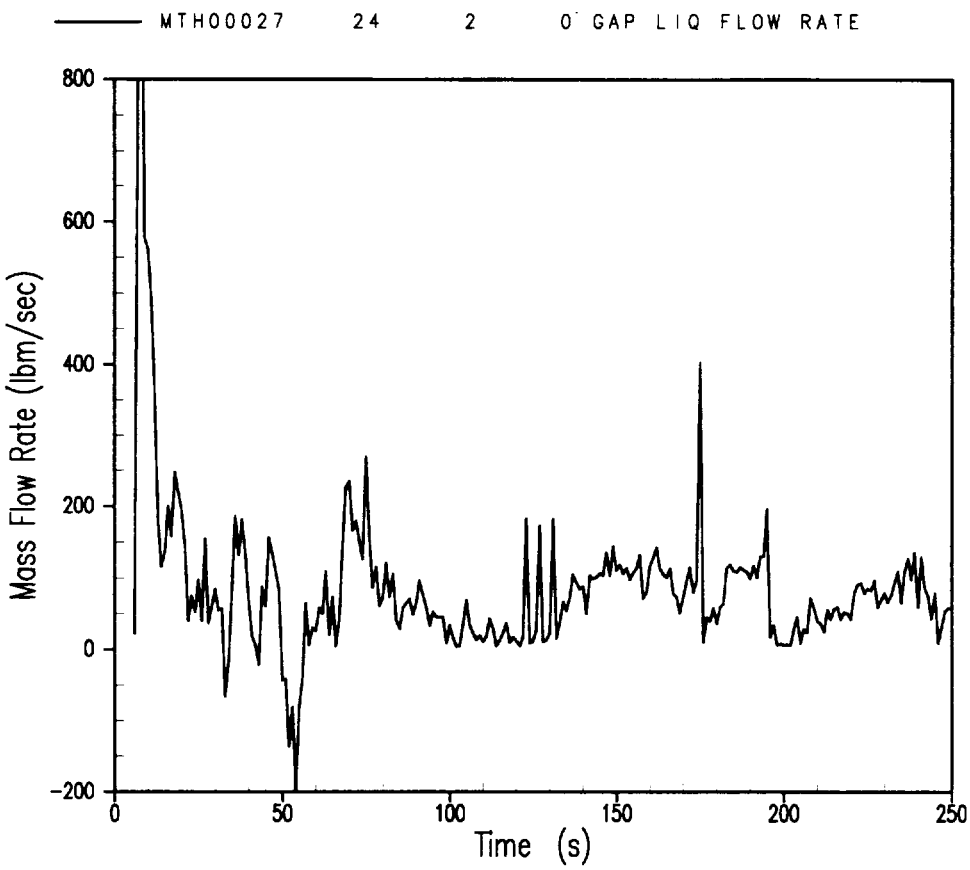


Figure 440.173-12: Entrained Liquid Flow Entering the 2*ADS-4 Flow Path Hot Leg, Base Case Upper Plenum Model

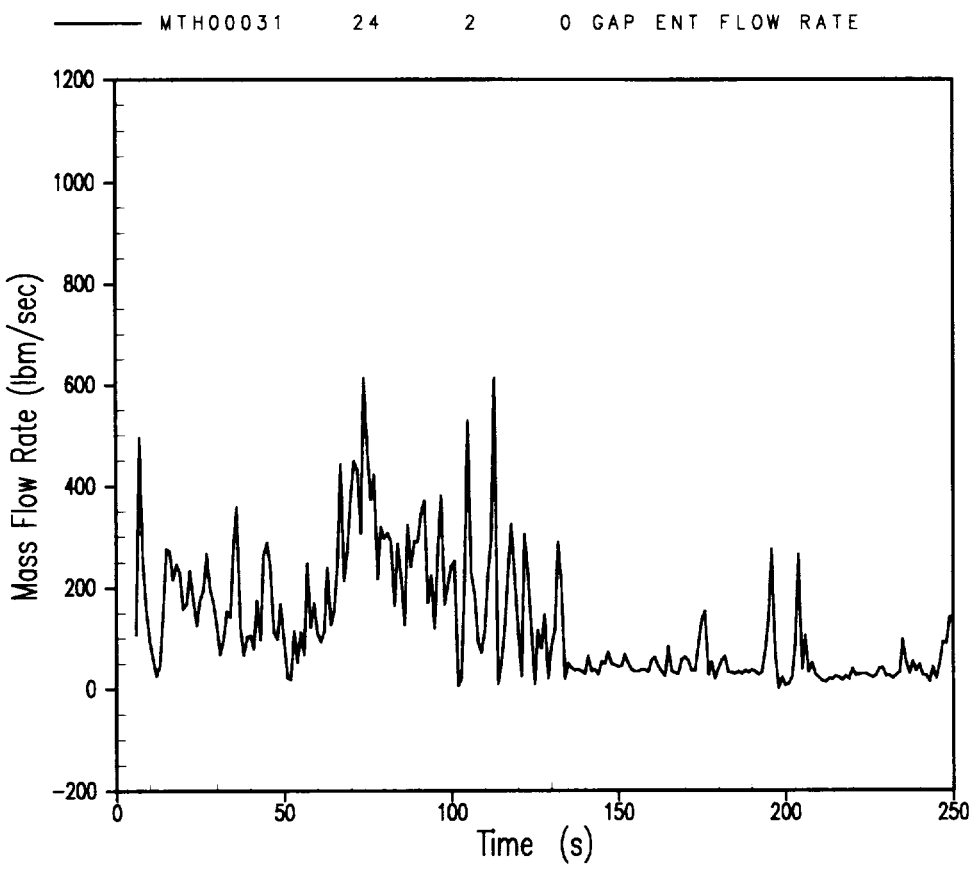


Figure 440.173-13: Vapor Flow Entering the 2*ADS-4 Flow Path
Hot Leg, Base Case Upper Plenum Model

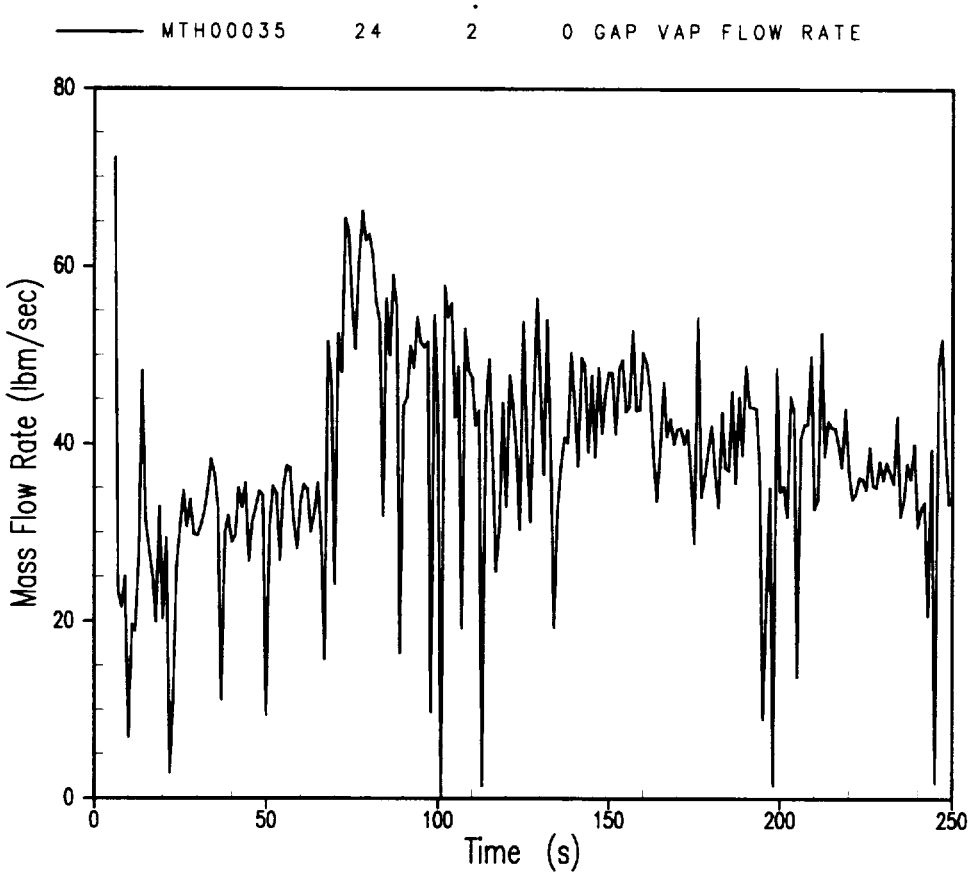


Figure 440.173-14: Upper Plenum Continuous Liquid Flow Rate, Base Case Upper Plenum Model

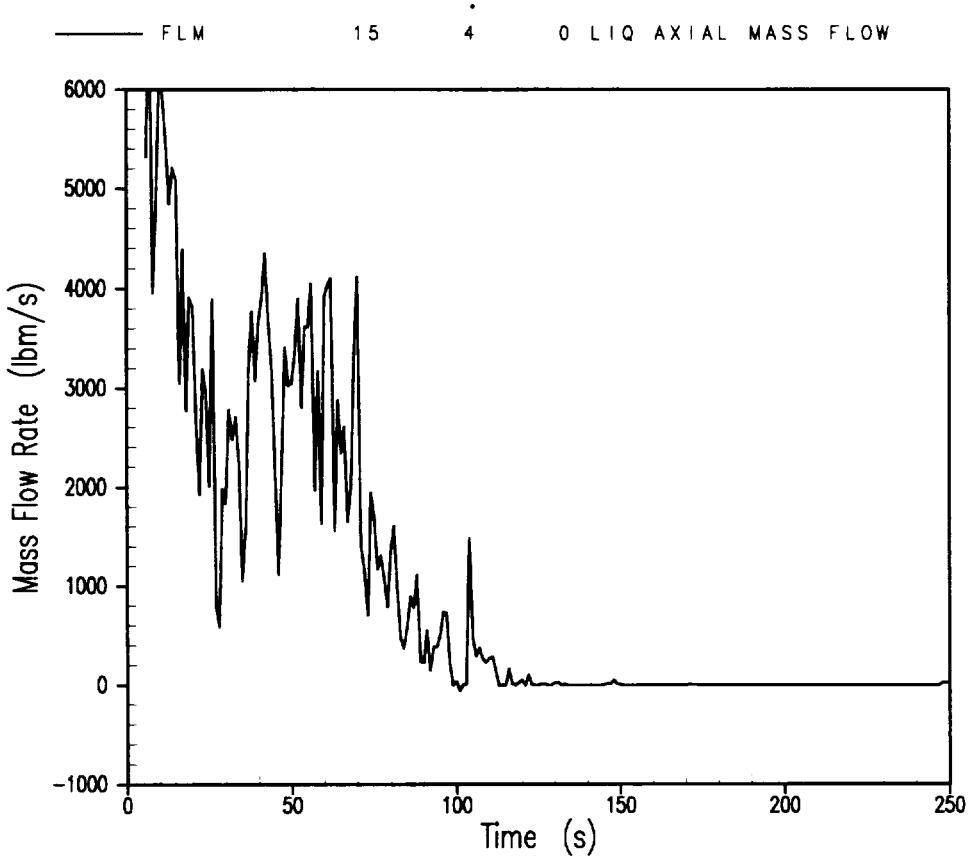


Figure 440.173-15: Upper Plenum Entrained Liquid Flow Rate.
Base Case Upper Plenum Model

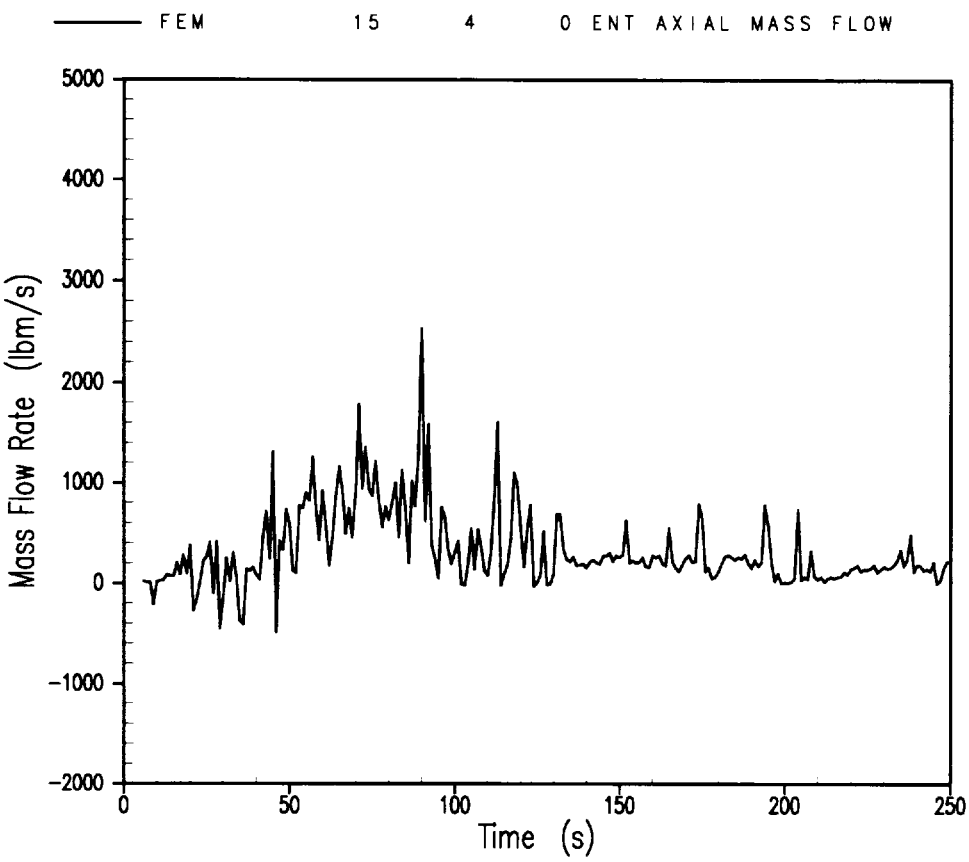


Figure 440.173-16: Upper Plenum Steam Flow Rate, Base Case Upper Plenum Model

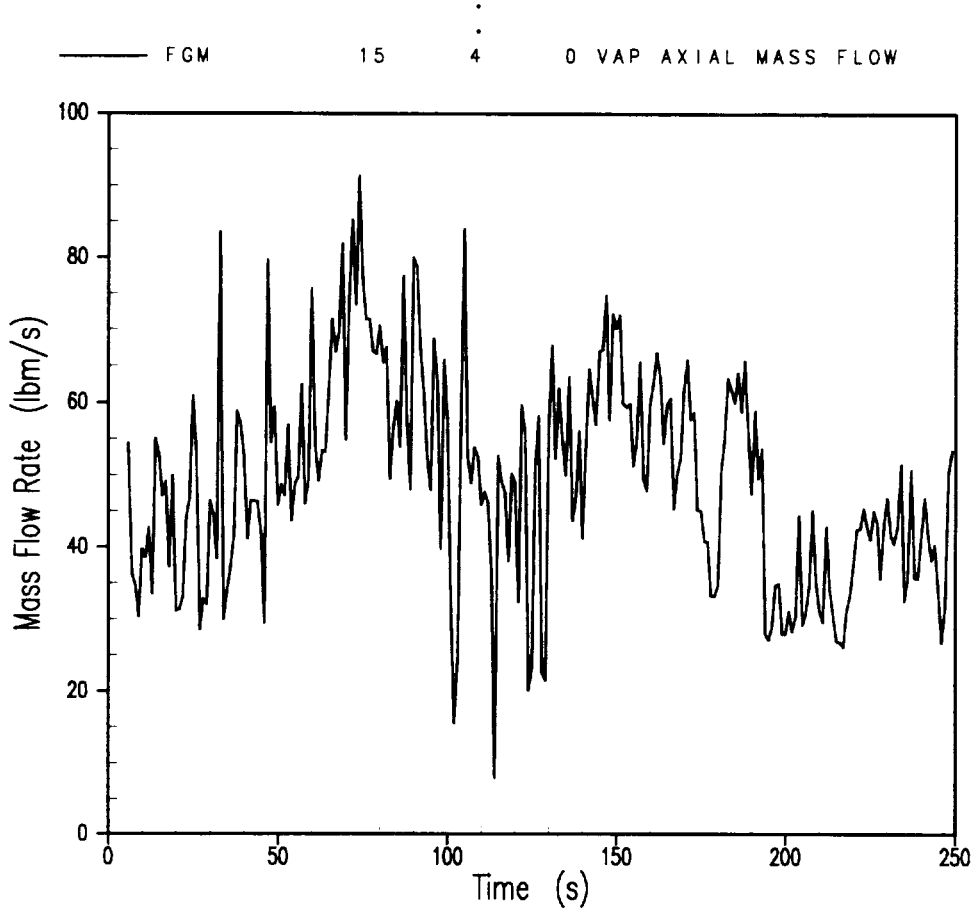


Figure 440.173-17: Upper Plenum Continuous Liquid Flow Rate, Base Case Upper Plenum Model

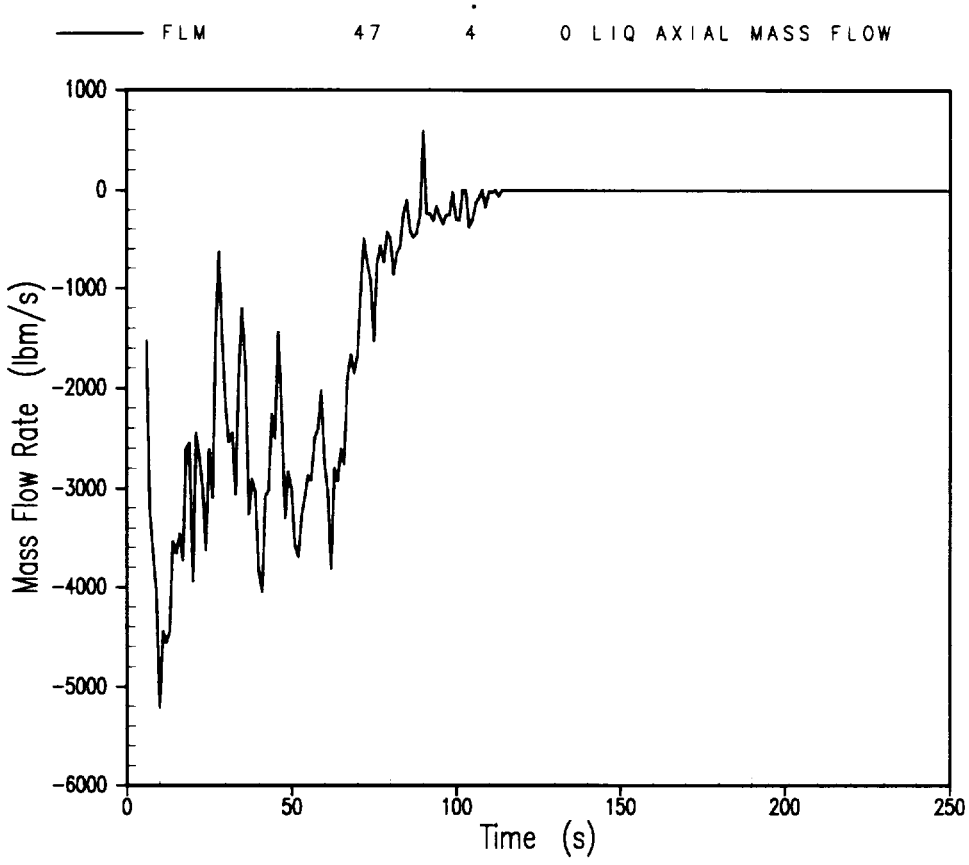


Figure 440.173-18: Upper Plenum Entrained Liquid Flow Rate, Base Case Upper Plenum Model

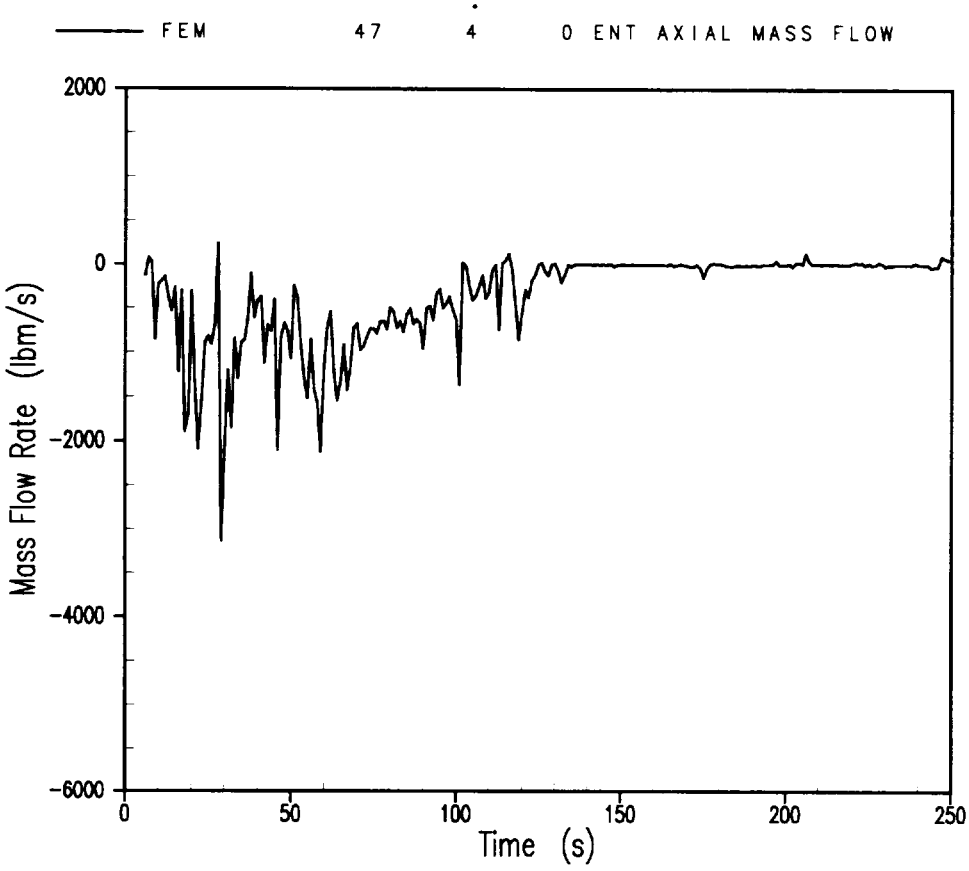


Figure 440.173-19: Upper Plenum Steam Flow Rate. Base Case Upper Plenum Model

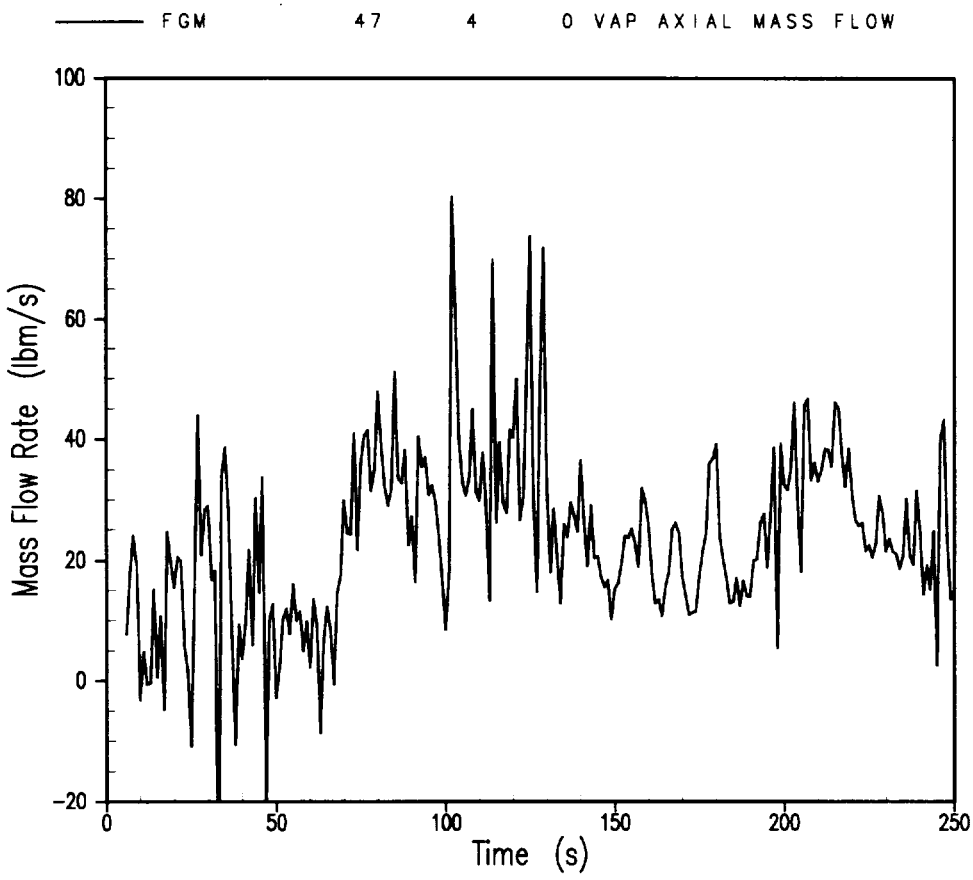
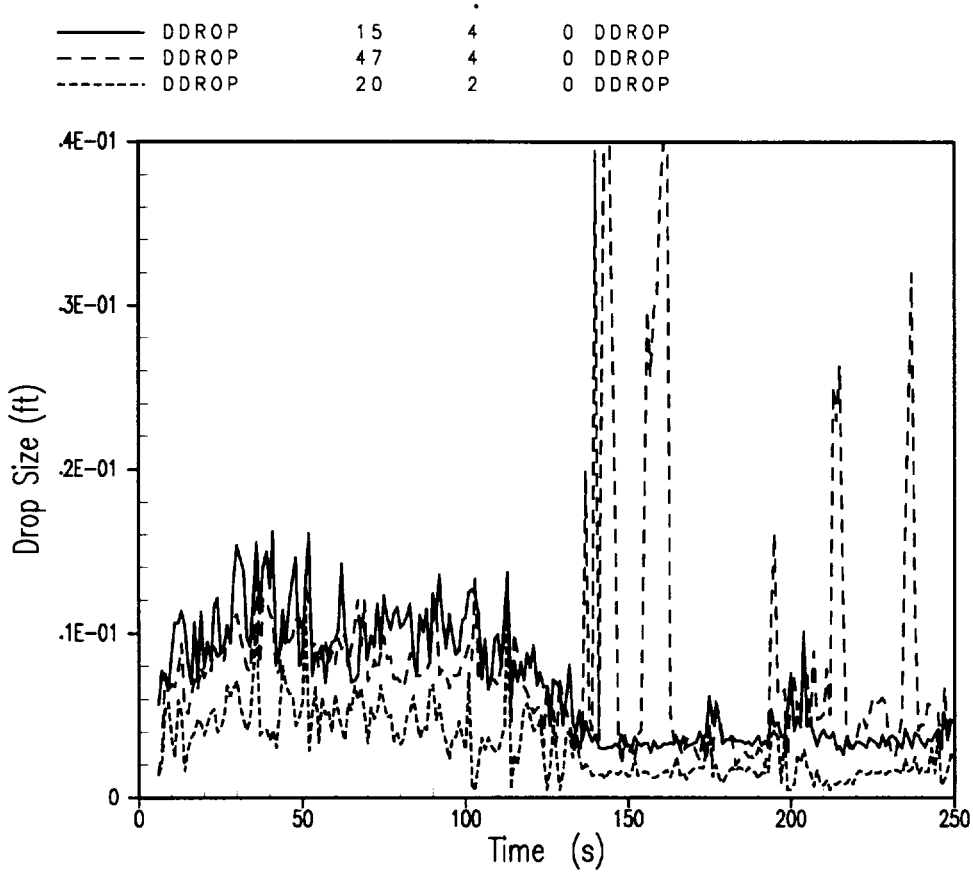


Figure 440.173-20: Drop Size in Upper Plenum at Hot Leg Elevation. Base Case Upper Plenum Model



AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.179

Question:

As discussed in Section A.4.5, when the void fraction in WCOBRA/TRAC cell is less than 0.8 the cell is assumed to be liquid-continuous and when the cell void fraction is greater than 0.8 the cell is assumed to be vapor-continuous. Please justify this assumption by comparison to experimental level swell data for conditions corresponding to those calculated for AP1000.

Westinghouse Response:

In WCOBRA/TRAC the transition from churn-turbulent flow regime to annular flow regime occurs when the void fraction reach a critical value α_{crit}. The critical void fraction α_{crit}. is determined from a force balance between the disruptive force of the pressure gradient over the crest waves on the film and the restraining force of surface tension. The critical void fraction is then limited to a value no less than 0.8. The critical void fraction is determined as follows:

$$\alpha_{crit} = \text{maximum}\begin{cases} 1 - \dfrac{2\sigma}{\rho_v |u_{vl}|^2 D_h} - \alpha_e \\ 0.8 \end{cases}$$

The following plots show some of the most widely used vertical flow regime maps. Figure 1 shows the flow regime map by Taitel-Dukler in terms of the phase superficial velocities Jg and Jl. Figure 2 shows the map from Mishima-Ishii defined in void fraction and gas superficial velocity. The same flow regime map is 'translated' in a equivalent flow regime map defined in term of Jg and Jl in Figure 3.

The comparison of Figure 1 and Figure 3 shows that the transition from slug/churn flow regime to annular flow regime is equivalent in both Mishima-Ishii and Taitel Dukler. Figure 4 shows the flow regime map implemented in WCOBRA/TRAC. The comparison between Figure 4 and Figure 2 shows that the flow regime map coded in WCOBRA/TRAC is consistent with the most widely used flow regime maps.

Figure 1 - Flow Regime Map by Taitel-Dukler:

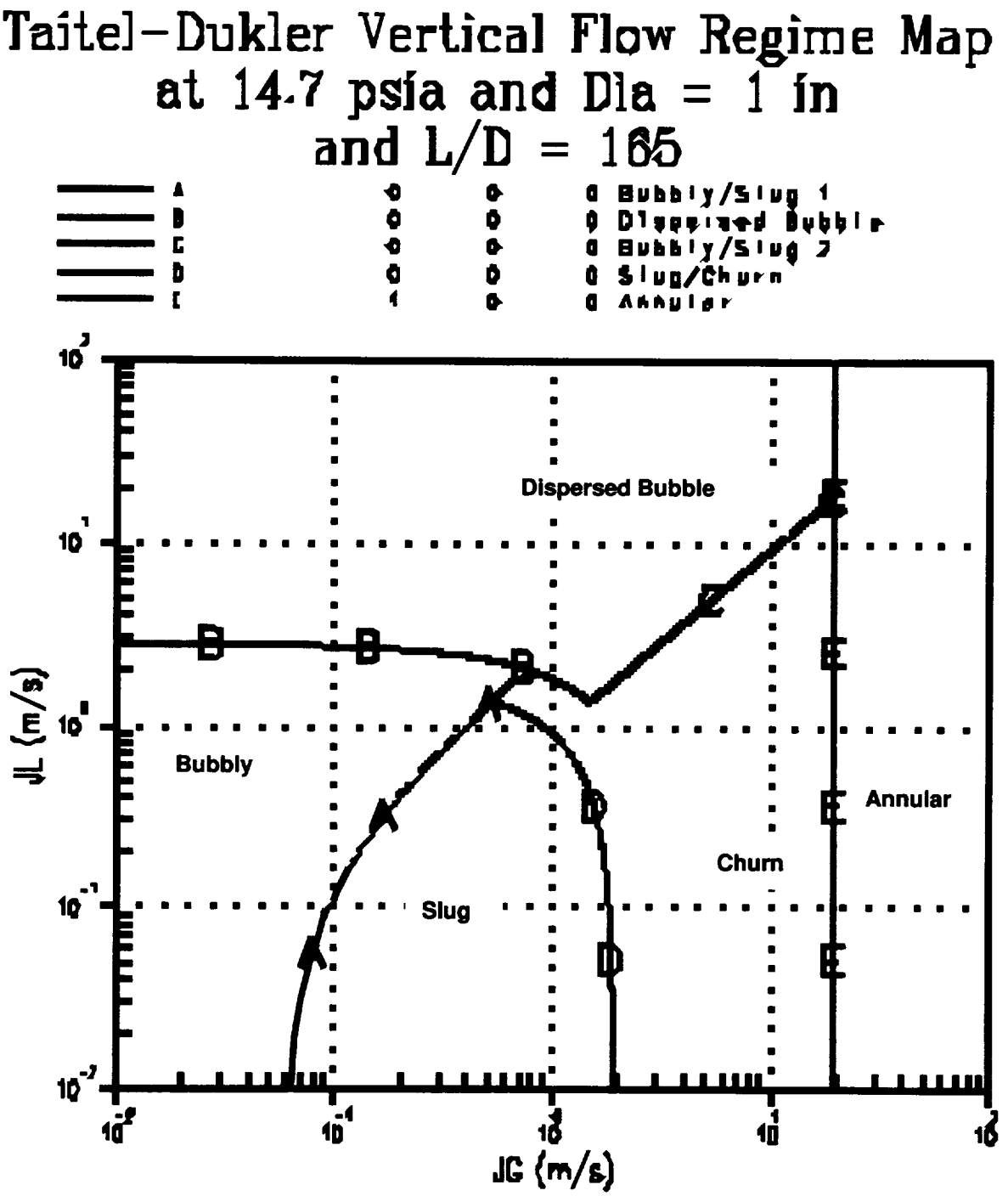
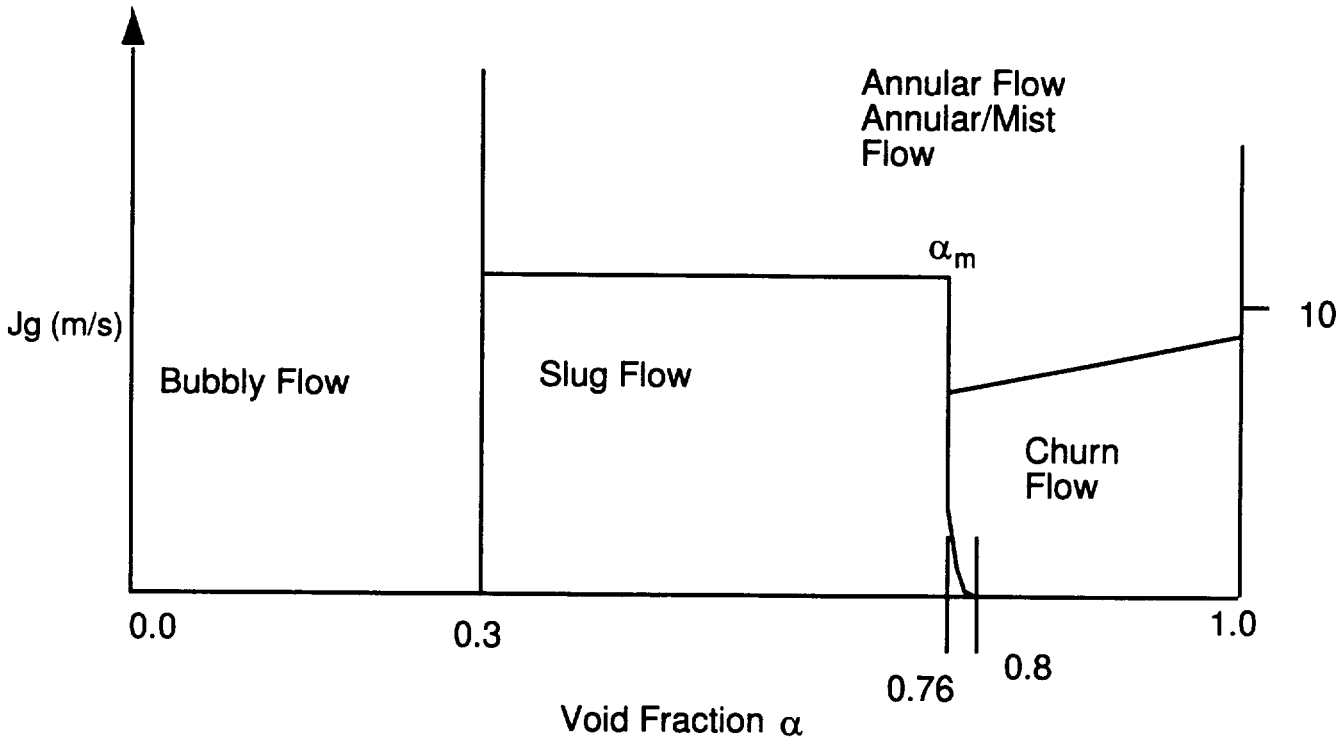


Figure 2 - Mishima-Ishii defined in Void Fraction:



Mishima/Ishii Vertical Flow Regime

Figure 3 - Mishima-Ishii Map in J_f-J_g Plane:

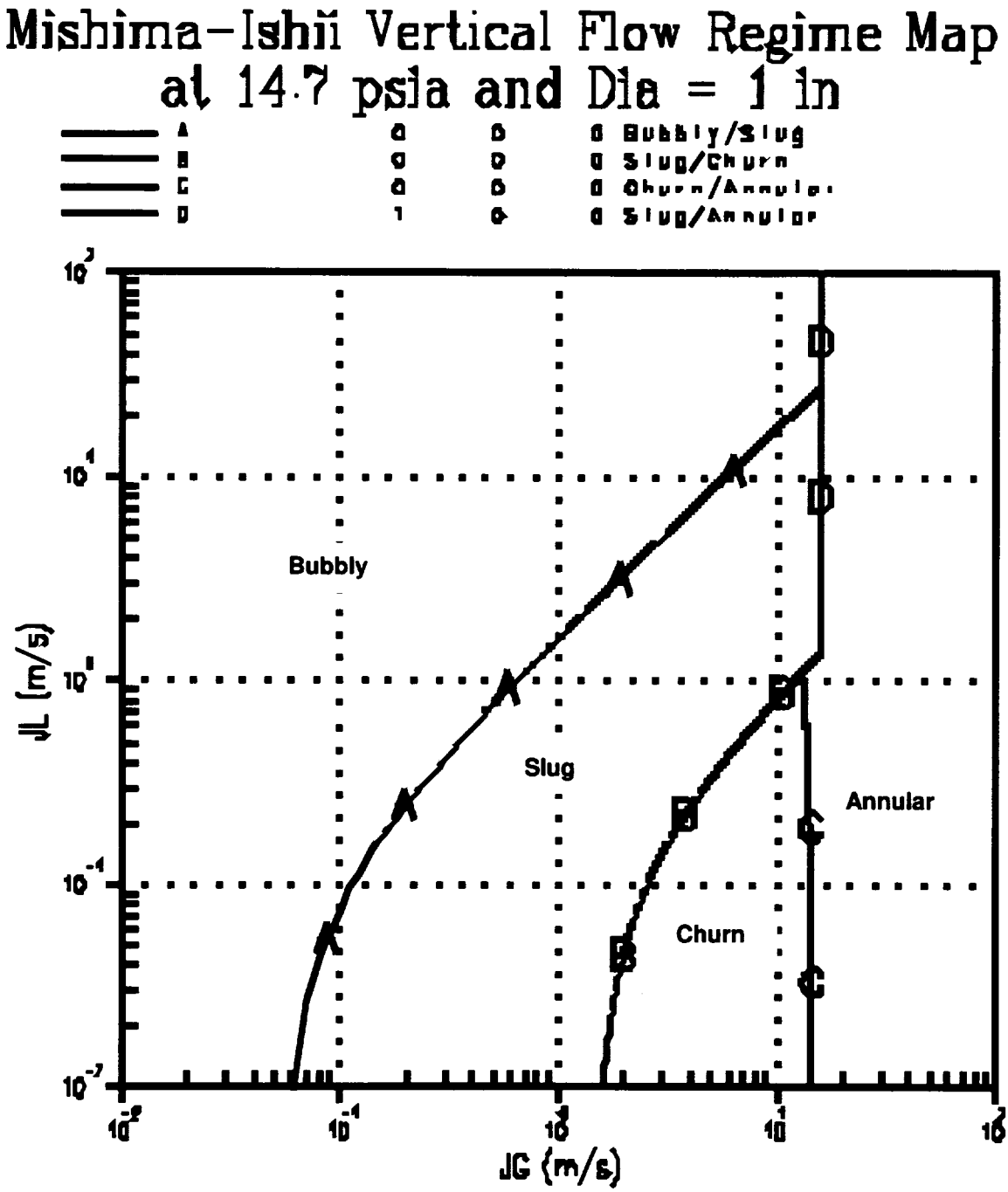
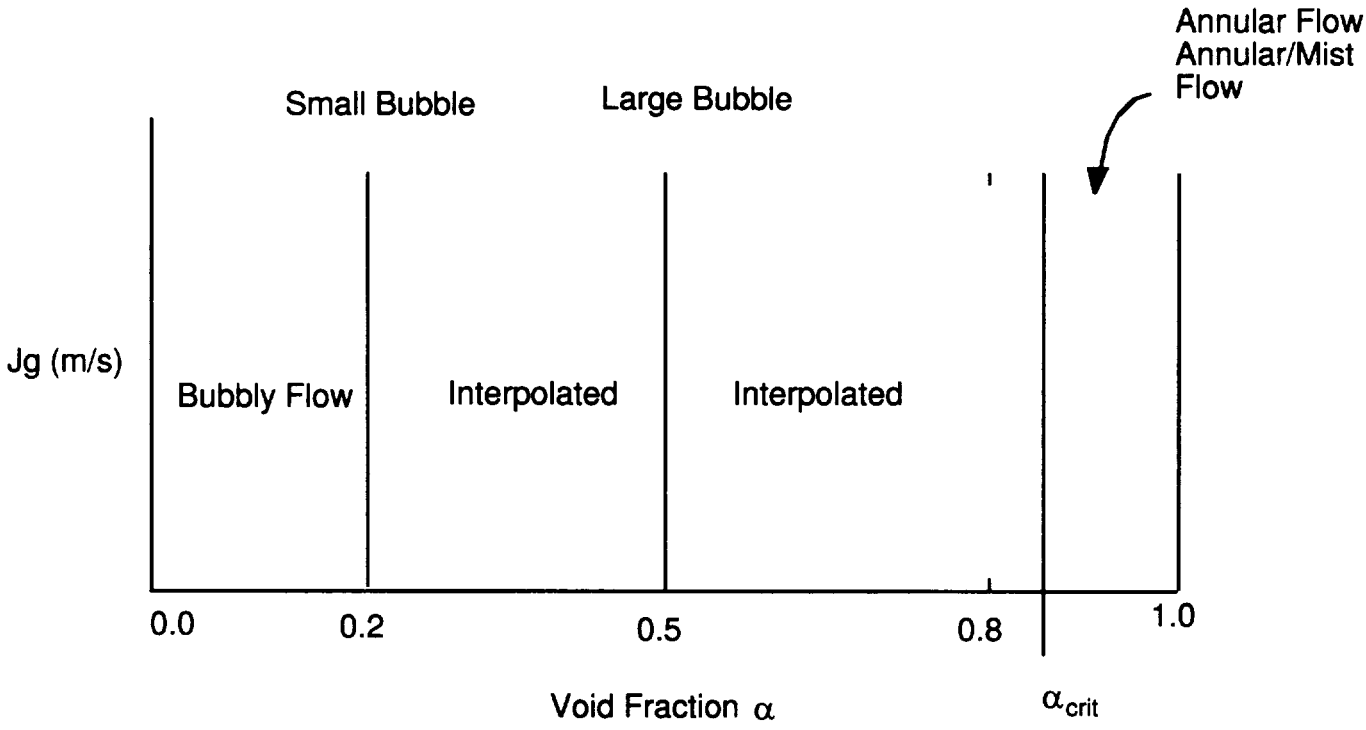


Figure 4 - WCOBRA/TRAC:



WCOBRA/TRAC Vessel Vertical Flow Regime

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.180

Question:

In Section A.4.5, it is stated that in WCOBRA/TRAC if the difference in void fraction between a cell center and the lower boundary is greater than 0.05 then a two phase level exists in the bottom of the cell directly above; whereas if the difference in void fraction between a cell center and the upper boundary is greater than 0.05 then a two phase level exists in the top of the cell directly above. Please justify these assumptions by comparison to experimental level swell data for conditions corresponding to those calculated for AP1000.

Westinghouse Response:

The purpose of Section A.4.5 is to compare the Kataoka-Ishii model predictions for regions 2 and 3 with the WCOBRA/TRAC entrainment prediction. In order to do this, it is necessary to infer an upper plenum pool liquid level for the WCOBRA/TRAC case. The WCOBRA/TRAC computer code does not contain an explicit mixture level tracking model. Mixture level is therefore determined by nodalization, and by the prediction of the axial void gradient between hydraulic cells. Figure A.4.5-2 presents the upper plenum void fraction profile at seven points in time during the ADS-4 IRWST initiation transient. One possible approach to define a pool level suggested by Figure A.4.5-2 is to simply declare that shortly after 104 seconds, when both Channel 20 Cell 2 and Channel 47 Cell 4 exhibit void fractions > 0.8, that the upper plenum pool level is below the bottom of the hot leg because both of these cells are within the vapor-continuous flow regime.

This linear interpolation of void fractions does not identify the location of an interface. The WCOBRA/TRAC channel methodology permits an alternate means to infer a pool liquid level, according to the method identified in Section A.4.5. This approach predicts a more specific location of the pool level based on the gradient in predicted void fraction of the adjacent cells. At 104 seconds, the Channel 20 Cell 2 void fraction is approximately 0.9, while the Channel 47 Cell 4 void fraction is below 0.8; due to the change in flow regime from liquid-continuous to vapor-continuous at 0.8 void fraction, a pool level clearly does exist between the centers of these adjacent cells as long as the Channel 47 Cell 4 void fraction remains at or below 0.8. Therefore, the gradient of [0.9 – 0.8]* 0.5, or 0.05, between cell center and cell boundary is selected as the criterion for establishing an interface location. Please note that this approach is merely a means by which to identify a level from the WCOBRA/TRAC prediction for use in the Kataoka-Ishii correlation – it is not intended to accomplish anything beyond that, and it is not based on an experimental result. It accomplishes the goal of identifying a pool level from the WCOBRA/TRAC prediction to permit comparison with the Kataoka-Ishii correlation regions 2 and 3 entrainment values.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number 440.182

Question:

A. AP1000 RAI 440.009 requested information on the steady-state pressure drops through the vessel and various regions and components in the primary loops of the AP1000 design at best estimate flow conditions for a 10 percent steam generator tube plugging level.

As a part of AP1000 design certification, please specify in Tier 1 ITAAC Table 2.1.2-4 the acceptable ranges of the resistance factors for the regions and components described in RAI 440.009. Describe the bases for the acceptance ranges.

B. Figure 2.1.2-1 in Tier 1 information provides the general layout of the reactor coolant system. Please specify in Table 2.1.2-4 the acceptable ranges of the pressurizer volume, the horizontal lengths of the hot legs and cold legs, the relative locations of the pressurizer and the ADS-4 off-take pipes from the reactor vessel, and the elevation of steam generator relative to the reactor vessel. Describe the bases for the acceptance ranges.

C. What is the acceptable range, and the basis, of the reactor coolant pump head-capacity characteristics for the AP1000 design? Why is this not included in the Tier 1 ITAAC?

Westinghouse Response:

A & C. The RCS pressure drop and reactor coolant pump head/flow characteristics are related directly to the RCS flow rate at normal power operation. A commitment to verify the minimum reactor flow rate is included in the AP1000 Tier 1 Table 2.1.2-4, Design Commitment 9a. Rather than acceptable ranges, the parameters committed to verification in the AP1000 reactor coolant system ITAACs are the minimum or maximum acceptable values necessary to ensure that the given design commitment will be met. Therefore, the parameters specified in the reactor coolant system ITAACs take into consideration design and analysis uncertainties.

B. The key reactor coolant system related dimensions identified for the AP1000 ITAACs are those for the reactor vessel. The design commitments for these dimensions are given in Tier 1 Table 2.1.3-2, Items 2.b and 2.c.

Generally the AP1000 ITAACs have been developed to include only those key parameters that are important to safety. This is the same approach as used in the development of the ITAACs for AP600 and System 80+. In previous NRC reviews of these new plant designs, reactor coolant system ITAACs similar to those provided for the AP1000 were found acceptable. Both plants received design certification with ITAACs that do not include design commitments for the parameters requested above.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 451.001

Question:

What are the references for the meteorological data used in the analyses resulting in selection of the site parameter values presented in Table 2-1?

Westinghouse Response:

The atmospheric dispersion values (X/Q), reported in DCD Chapter 2, Table 2-1, were not determined through analysis using a meteorological data set. Instead, the X/Q values were initially developed for the AP600 as the result of the review of 44 plant sites. Values for X/Q were selected to bound 80% of the sites. For the AP1000, the X/Q values for the Low Population Zone were maintained without modification but the value for the Exclusion Area Boundary X/Q was revised from 1.0E-3 sec/m^3 to 6.0E-4 sec/m^3 (this value is representative of a 60-70th percentile U.S. site). See the response to RAI 451.003 for additional discussion.

The site parameters for air temperature, wind speed and precipitation are based on those specified in the Utility Requirements Document and were used for the AP600. The wind speed is adjusted to the ASCE 7-98 Code as described in DCD subsection 3.3.1.1.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 451.003

Question:

What factors contributed to the reduction in the 0 - 2 hour exclusion area boundary (EAB) X/Q to 6.0E-4 sec/m^3 (seconds-per-cubic meter) proposed for the AP1000 from the value of 1.0E-3 sec/m^3 for the AP600?

Westinghouse Response:

The EAB X/Q is a site interface parameter. Because of the increase in core power for the AP1000 there is an associated increase in the core fission product source term and this resulted in the change to X/Q. The AP1000 X/Q was specified to obtain a dose just below the acceptance limit of 25 rem TEDE at the EAB for the postulated Loss-of-Coolant Accident (LOCA) with degraded core.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 451.006

Question:

Section 15A.3.3 of the AP1000 DCD discusses calculation of the bounding X/Q values for use in the control room design-basis accident dose assessments. Diagrams showing the site plan with release and intake locations are included. Was this description provided for information only and it is expected that the methodology and all inputs, and assumptions selected by the COL applicant will be evaluated at the time of the COL review? If the methodology, and all inputs and assumptions will be evaluated during the COL review, that requirement should be explicitly stated in Section 15A.

If a commitment will not be made that the methodology and all inputs, and assumptions selected by the COL applicant will be evaluated at the time of the COL review, please address the following:

A. What specific methodology, inputs, and assumptions are proposed as part of the AP1000 Design Certification application? Other than the site meteorological data, what will be provided as part of the COL application?

B. Do Notes 4 through 7 on page 15A-15 simply list the design-basis accidents to which the X/Q values should be applied or are the Notes stating that some other X/Q values have not been listed because they are less than, and therefore bounded by, the listed X/Q values? If only bounding values were provided, provide inputs and assumptions for the values that were not submitted in the AP1000 DCD.

C. Because distances, heights, building dimensions and assumptions have not been provided, it is difficult to judge these X/Q values. However, on page 15A-17 the main equipment hatch release location appears to be quite close to the control room heating, ventilation and air conditioning (HVAC) intake, yet the 0 - 2 hour X/Q value listed is, or is bounded by, 1.2E-3 sec/m^3. This value seems low given the short apparent distance relative to other postulated release locations. Further, it is the same value as for the equipment hatch at the staging area which is further away. This appears inconsistent. Please explain.

D. In the Table 15A-5 listing of X/Q values, some of the X/Q values are the same for two release locations and/or consecutive time periods. To what is this attributed? Are these occurrences the result of selecting bounding values or a result of other assumptions? Further, in the case of the ground level containment release points, the proposed values are different than those for the AP600. What factors contribute to the differences?

Westinghouse Response:

Figures 15A-1 & 2 are provided for information only. DCD Section 15A.3.3 provides bounding X/Q values for use in the AP1000 control room design-basis accident dose assessments. The selection of X/Q values was made with the expectation that the values would exceed those calculated for specific sites. The COL item associated with determining X/Q is contained in DCD Section 2.3.6.4.

It is necessary for the COL applicant to collect site-specific meteorological data and perform the appropriate analysis to calculate the atmospheric dispersion factors. As required for all Combined License information items, the specific methodology, inputs and assumptions used in this analysis will be provided as part of the COL application.

A. The methodology that was used in developing the control room X/Q values reported in Table 15A-5 of the DCD is discussed in DCD Section 15A.3.3 and utilizes the ARCON96 computer code. It is expected that the COL applicant will also utilize the ARCON96 code if the existing calculation is not bounding for the selected site.

B. As stated in Note 4, the listed X/Q values bound the values that were calculated for the main equipment hatch and the staging area hatch. The stated values are those calculated for the staging area hatch; the values that were calculated for the main equipment hatch are substantially lower.

Per Note 5, the stated secondary side X/Q values bound the values that were calculated for the release points for the steam vents, the steam line safety & power-operated relief valves, and the condenser air removal stack. Additionally, the bounding calculated values were adjusted upward to provide additional conservatism beyond that included in the calculation.

Notes 6 and 7 describe the stated X/Q values as being bounding values. In these instances, the description of the values as being bounding reflects the fact that the calculated X/Q values were adjusted upward to provide additional conservatism beyond that included in the calculation.

C. As stated in the response to Item B above, the X/Q values for the staging area hatch release point were calculated as bounding those for the equipment hatch. The staging area hatch releases were modeled as being released into the auxiliary building at elevation 100'-0" and assumed to leak to the annex building by way of the roll-up doors leading to the annex building. The cloud of activity in the Annex building was then modeled as the source for releases to the environment by way of the sliding door at elevation 107'-2".

The release pathway for activity postulated to leak past the main equipment hatch is considerably more complex than that for the staging area hatch. The calculations of the X/Q values have taken this complexity of the pathways into account. Leakage from the main equipment hatch would build up in the auxiliary building at elevation 135'-3" and potentially leak through the doors to the annex building. Prior to reaching the environment, the activity would first have to pass through the auxiliary and annex buildings through several pathways, as illustrated in the following:

A. Leakage first enters an enclosed penetration room having only one egress pathway through the normally-closed sliding access door to the annex building.

B. From the door to the annex building, the leakage would enter the large staging and storage area (elevation 135'-3"). Activity then would have to pass through the door leading to the enclosed stairwell.

C. Activity passing through the stairwell would exit at an elevation of 107'-2". From this point there are two parallel pathways for potential release:

1. Pass down the short flight of stairs to the access area on elevation 100'-0", through a set of double doors and into a corridor. At this point there are several other parallel pathways leading to the environment:

   (a) Pass into an enclosed stairwell and out to the environment via the emergency exit door (elevation 100'-0").
   (b) Through the men's changing area and out via an emergency exit door.
   (c) Into the office corridor and out via an emergency exit door.
   (d) Down to the north end of the corridor and out via double doors.

2. Out through double doors into the machine shop access area and out to the environment via the controlled access emergency exit door.

The release point C.1(a) was chosen as the assumed location for the release of activity to the environment for the purposes of calculating X/Q because the emergency exit door is approximately in-line with the main equipment hatch and because it is located at the approximate midplane of the potential release points. To take into account for the tortuous pathway to reach the release point, a diffuse source was modeled.

D. For the various release locations, there was little difference between the calculated values for the 8-24 hour period and the 1-4 day period (this also indicated by the fact that where the stated X/Q values for these two intervals does differ, the difference is small). As a result, when the values were rounded up, often the same value was applied to both time intervals.

The X/Q values for the ground level containment release points differ from those defined for the AP600 dose analyses. The values reported in the AP600 DCD (and which were used in the AP600 analyses) include arbitrary increases to provide additional conservatism beyond that included in the calculation of the X/Q values. The values listed in Table 15A-5 of the AP1000 DCD are the values as conservatively calculated but without the additional margin that was included for AP600.

Design Control Document (DCD) Revision:

Sheet 2 of Table 15A-5 is revised to correct an entry error. The value for the secondary side release points X/Q at the control room door is changed from 1.0E-4 sec/m^3 to 1.0E-3 sec/m^3. The radiological consequences analyses use the latter value and are thus not affected.

From DCD Chapter 15, Appendix 15A, Table 15A-5 (Sheet 2 of 2):

Table 15A-5 (Sheet 2 of 2)

ATMOSPHERIC DISPERSION FACTORS (χ/Q)
FORACCIDENT DOSE ANALYSIS

Main control room

χ/Q (s/m^3) at HVAC Intake for the Identified Release Points[1]

	Elevated Containment Release[3]	Ground Level Containment Release Points[4]	Secondary Side Release Points[5]	Fuel Handling Area[6]	Fuel Building Relief Panel[7]
0 - 2 hours	1.2E-3	1.2E-3	2.0E-2	2.0E-3	3.0E-3
2 - 8 hours	8.0E-4	6.3E-4	1.8E-2	1.5E-3	2.0E-3
8 - 24 hours	4.0E-4	3.0E-4	8.0E-3	8.0E-4	1.0E-3
1 - 4 days	4.0E-4	3.0E-4	7.0E-3	8.0E-4	1.0E-3
4 - 30 days	3.0E-4	2.6E-4	6.0E-3	7.0E-4	9.0E-4

χ/Q (s/m^3) at Control Room Door for the Identified Release Points[2]

	Elevated Containment Release[3]	Ground Level Containment Release Points[4]	Secondary Side Release Points[5]	Fuel Handling Area[6]	Fuel Building Relief Panel[7]
0 - 2 hours	4.0E-4	6.6E-4	2.5E-3	1.0E-3	1.0E-3
2 - 8 hours	2.0E-4	3.8E-4	2.0E-3	6.0E-4	6.0E-4
8 - 24 hours	1.0E-4	1.9E-4	~~1.0E-4~~1.03E-3	3.0E-4	3.0E-4
1 - 4 days	9.0E-5	1.8E-4	9.0E-4	3.0E-4	3.0E-4
4 - 30 days	8.0E-5	1.6E-4	8.0E-4	2.5E-4	2.5E-4

Notes:

1. These dispersion factors are to be used 1) for the time period preceding the isolation of the main control room and actuation of the emergency habitability system, 2) for the time after 72 hours when the compressed air supply in the emergency habitability system would be exhausted and outside air would be drawn into the main control room, and 3) for the determination of control room doses when the non-safety ventilation system is assumed to remain operable such that the emergency habitability system is not actuated.
2. These dispersion factors are to be used when the emergency habitability system is in operation and the only path for outside air to enter the main control room is that due to ingress/egress.
3. These dispersion factors apply to releases from the plant vent.
4. The listed values bound the dispersion factors for releases from the main equipment hatch and the staging area hatch. These dispersion factors would be used for evaluating the doses in the main control room for a loss-of-coolant accident, for the containment leakage of activity following a rod ejection accident, and for a fuel handling accident occurring inside the containment.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number 460.002

Question:

(Section 11.1) Demonstrate that the AP1000 design meets the following criteria listed in SRP Section 11.1, Paragraph II, "Acceptance Criteria." Please identify where each of the following criteria are addressed in the DCD. If any criteria is not addressed, please provide that information and revise the DCD accordingly. Providing a table that identifies the applicable criteria and DCD section would be adequate.

A. The parameters used to calculate primary and secondary coolant concentrations are consistent with those given in NUREG-0017, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors."

B. All normal and potential sources of radioactive effluent delineated in Subsection I of SRP Section 11.1 are considered.

C. For each source of liquid and gaseous waste considered in subsection I.1 of SRP 11.1, the volumes and concentrations of radioactive material given for normal operation and anticipated operational occurrences are consistent with those given in NUREG-0017.

D. Decontamination factors for in-plant control measures used to reduce gaseous effluent releases to the environment, such as iodine removal systems and high-efficiency particulate air (HEPA) filters for building ventilation exhaust systems and containment internal cleanup systems are consistent with those given in RG 1.140. The building mixing efficiency for containment internal cleanup is consistent with NUREG-0017.

E. Decontamination factors for in-plant control measures used to reduce liquid effluent releases to the environment, such as filters, demineralizers, and evaporators are consistent with those given in NUREG-0017.

F. Radwaste augments used in the calculation of effluent releases to the environment are consistent with the findings of a cost-benefit analysis, and are performed using the guidance of RG 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Plants." The provisions that require a cost-benefit analysis are stated in Section II.D of Appendix I to 10 CFR Part 50.

G. Effluent concentration limits at the boundary of the unrestricted area do not exceed the values specified in Table 2 of Appendix B to 10 CFR Part 20.

H. The source terms result in meeting the design objectives for doses in an unrestricted area as set forth in Appendix I to 10 CFR Part 50.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

I. For evaluating the source terms, the applicant should provide the relevant information in the DCD as required by 10 CFR Part 50, Section 50.34a. This technical information should include all the basic data listed in Appendix B to RG 1.112 required in calculating the releases of radioactive material in liquid and gaseous effluents. An acceptable method for satisfying the criteria given in items 1 through 6 consists of using the Gaseous and Liquid Effluent (GALE) Computer Code and the source term parameters given in NUREG-0017.

J. If the calculational technique or any source term parameter differs from that given in NUREG-0017, please describe these differences in detail and the bases for the methods and parameters used.

Westinghouse Response:

The acceptance criteria of SRP section 11.1, Paragraph II as stated above are addressed in the AP1000 DCD as shown in the following table:

Criteria		AP1000 DCD Section	Comments
A	The parameters used to calculate primary and secondary coolant concentrations are consistent with those given in NUREG-0017, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors."	11.1.3, Table 11.1-1, Table 11.1-7	
B	All normal and potential sources of radioactive effluent delineated in Subsection I of SRP Section 11.1 are considered.	Table 11.2-6	
C	For each source of liquid and gaseous waste considered in subsection I.1 of SRP 11.1, the volumes and concentrations of radioactive material given for normal operation and anticipated operational occurrences are consistent with those given in NUREG-0017.	Table 11.2-6	
D	Decontamination factors for in-plant control measures used to reduce gaseous effluent releases to the environment, such as iodine removal systems and high-efficiency particulate air (HEPA) filters for building ventilation exhaust systems and containment internal cleanup systems are consistent with those given in RG 1.140. The building mixing efficiency for containment internal cleanup is consistent with NUREG-0017.	Table 11.2-6	

Criteria		AP1000 DCD Section	Comments
E	Decontamination factors for in-plant control measures used to reduce liquid effluent releases to the environment, such as filters, demineralizers, and evaporators are consistent with those given in NUREG-0017.	Table 11.2-5, Table 11.2-6	
F	Radwaste augments used in the calculation of effluent releases to the environment are consistent with the findings of a cost-benefit analysis, and are performed using the guidance of RG 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Plants." The provisions that require a cost-benefit analysis are stated in Section II.D of Appendix I to 10 CFR Part 50.	Not applicable.	Augments are not assumed as part of the licensing basis for the AP1000 radwaste systems.
G	Effluent concentration limits at the boundary of the unrestricted area do not exceed the values specified in Table 2 of Appendix B to 10 CFR Part 20.	Table 11.2-8 (liquid releases), Table 11.3-4 (gaseous releases)	
H	The source terms result in meeting the design objectives for doses in an unrestricted area as set forth in Appendix I to 10 CFR Part 50.	11.3.3.1, 11.3.3.4	
I	For evaluating the source terms, the applicant should provide the relevant information in the DCD as required by 10 CFR Part 50, Section 50.34a. This technical information should include all the basic data listed in Appendix B to RG 1.112 required in calculating the releases of radioactive material in liquid and gaseous effluents. An acceptable method for satisfying the criteria given in items 1 through 6 consists of using the Gaseous and Liquid Effluent (GALE) Computer Code and the source term parameters given in NUREG-0017.	11.1	PWR-GALE code is used; thus RG 1.112 Appendix B is satisfied.
J	If the calculational technique or any source term parameter differs from that given in NUREG-0017, please describe these differences in detail and the bases for the methods and parameters used.	Not applicable.	NUREG-0017 is used for AP1000.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 460.004

Question:

(Section 11.2) AP1000 DCD, Section 11.2.1.2.1, "Capacity," states that the liquid waste system provides adequate capacity to meet the anticipated processing requirements of the plants. The tables being referenced in Section 11.2.1.2.1 do not have sufficient explanation regarding the capacity of the liquid waste system. The projected flows of various liquid waste systems under normal conditions are identified in DCD Table 11.2-1. Please provide additional information that demonstrates the adequacy of the liquid waste system to handle the normal input specified in Table 11.2-1.

DCD Table 11.2-4 contains information on the surge capacity of individual tanks. Where is the information about the surge input rates? Demonstrate the adequacy of the capacity to handle the surge input.

Revise DCD Section 11.2.1.2.1 to specifically explain the capacity of the liquid waste management system.

Westinghouse Response:

The installed ion exchange / filtration equipment has a processing flow capability of 75 gpm. This capacity is not explicitly stated in the AP1000 DCD and will be added to Section 11.2.1.2.1.

Wastes which enter the liquid radwaste system (WLS) are processed either through the permanently installed ion exchange / filtration train, shown on Figure 11.2-2 Sheet 4, or with mobile equipment. Referring to Table 11.2-1, it would typically be expected that:

- Fluid that enters the effluent holdup tanks or waste holdup tanks will be processed with the permanently installed ion exchange / filtration train.
- Fluid that enters the chemical waste tank will be analyzed and chemically treated as necessary, and then either processed with the permanently installed ion exchange / filtration train or processed with mobile equipment, depending upon the makeup of the tank contents.
- Fluid classified as "detergent waste" in Table 11.2-1 will be routed to either the waste holdup tanks or the chemical waste tank, depending upon its composition.

From an equipment capacity viewpoint it is conservative to assume that all liquid wastes are processed with the permanently installed equipment.

Table 11.2-1 provides expected generation rates for inputs. As noted in that table, the input from the chemical and volume control system (CVS) letdown, which is calculated specifically based upon AP1000 conditions, assuming startup from refueling, normal dilution for burnup, and shutdown and boration to refueling concentration over an 18 month fuel cycle.

Other inputs are not specifically dependent on AP1000 design features and therefore are taken from ANSI/ANS-55.6-1993, which is referenced in Table 11.2-1. That standard provides more definition of the various inputs to the WLS, and considers both limiting event (which are assumed to occur infrequently) and normal production. The standard identifies design bases for various subsystems based upon various infrequent events, which are not assumed to occur simultaneously. However, since AP1000 processes most liquid radioactive waste through a single train, it is instructive to conservatively assume that all of the peak events from ANSI/ANS-55.6-1993 occur in a single year, as shown in Table RAI-460.004-1, attached.

Using these conservatively high assumptions for volumes gives total annual processing of about 1,300,000 gallons. With the on-site processing capability of 75 gpm this will require less than one hour per day. Moreover, if the items noted as peaking during shutdown are considered, they total about 385,000 gallons (this is over-estimated, since much of the CVS letdown is generated throughout the core cycle by burnup dilution). Processing that 385,000 gallon generation during a 17 day outage will require processing time of five hours per day. This still allows adequate time for sampling, maintenance of the processing equipment, and so forth.

Design Control Document (DCD) Revision:

Revise section 12.2.1.2.1 as shown in the attached markup.

PRA Revision:

None

Table RAI-460.004-1 Conservative Annual Volumetric Inputs to AP1000 WLS				
Source	**Maximum Input Rate**	**Reference for Input Rate**	**Maximum Anticipated Annual Input, Total**	**Reference for Annual Input Total**
CVS letdown[1]	100 gpm	AP1000 specific / DCD Section 9.6.3	159,000 gallons	AP1000 calculations / DCD Table 11.2-1
Leakage inside containment, containment cooling	500 gpd plus one annual occurrence of 4 gpm	ANSI/ANS-55.6-1993	188,100 gallons	ANSI/ANS-55.6-1993 (500 gpd * 365 + 5,600 gal once)
Leakage inside containment, other	10 gpd plus one annual occurrence of 6 gpm	ANSI/ANS-55.6-1993	12,650 gallons	ANSI/ANS-55.6-1993 (10 gpd * 365 + 9,000 gal once)
Leakage outside containment, pumps and valves	80 gpd plus one annual occurrence of 20 gpm	ANSI/ANS-55.6-1993	39,200 gallons	ANSI/ANS-55.6-1993 (80 gpd * 365 + 10000 gal once)
Leakage outside containment, misc and SFP liner leakage	700 gpd plus one lifetime occurrence of 4 gpm	ANSI/ANS-55.6-1993	595,500 gallons	ANSI/ANS-55.6-1993 (700 gpd * 365 + 2 * 170,000 gal once)
Reactor systems sampling	200 gpd plus 3000 gal once	ANSI/ANS-55.6-1993	76,000 gallons	ANSI/ANS-55.6-1993 (200 gpd * 365 + 3,000 gal once)
Hot shower[1]	0 gpd normal, 400 gpd shutdown	ANSI/ANS-55.6-1993	1,200 gallons	ANSI/ANS-55.6-1993
Hand wash[1]	200 gpd normal, 1500 gpd shutdown	ANSI/ANS-55.6-1993	118,000 gallons	ANSI/ANS-55.6-1993 (200 gpd * 365 + 45,000 gal once)
Equipment and area decon[1]	40 gpd normal, 3000 gpd shutdown	ANSI/ANS-55.6-1993	104,600 gallons	ANSI/ANS-55.6-1993 (40 gpd * 365 + 90,000 gal once)
[1] Peaks during plant shutdown.				

11.2 Liquid Waste Management Systems

The liquid waste management systems include the systems that may be used to process and dispose of liquids containing radioactive material. These include the following:

- Steam generator blowdown processing system (subsection 10.4.8);
- Radioactive waste drain system (subsection 9.3.5);
- Liquid radwaste system (WLS) (Section 11.2).

This section primarily addresses the liquid radwaste system. The other systems are also addressed in subsection 11.2.3, which discusses the expected releases from the liquid waste management systems.

The liquid radwaste system is designed to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences.

11.2.1 Design Basis

Subsection 1.9.1 discusses the conformance of the liquid radwaste system design with the criteria of Regulatory Guide 1.143.

11.2.1.1 Safety Design Basis

The liquid radwaste system serves no safety-related functions except for:

- Containment isolation; see subsection 6.2.3.

- Draining the passive core cooling system compartments to the containment sump to prevent flooding of these compartments and possible immersion of safety-related components.

- Back flow prevention check valves in the drain lines from the chemical and volume control system compartment and the passive core cooling system compartments to the containment sump, which prevent cross flooding of these compartments. Each drain line has two check valves in series so that a single failure does not compromise the back flow prevention safety function. See subsection 6.3.3.3.2 for a discussion of containment flooding.

11.2.1.2 Power Generation Design Basis

11.2.1.2.1 Capacity

The liquid radwaste system provides holdup capacity as shown in Table 11.2-2, and permanently installed processing capacity of 75 gpm through the ion exchange / filtration train. This is adequate capacity to meet the anticipated processing requirements of the plant. The projected flows of various liquid waste streams to the liquid radwaste system under normal conditions are identified in Table 11.2-1.

The liquid radwaste system design can accept equipment malfunctions without affecting the capability of the system to handle both anticipated liquid waste flows and possible surge load due to excessive leakage. Table 11.2-4 contains information on the surge capacity of individual tanks.

Portions of the liquid radwaste system may become unavailable as a result of the malfunctions listed in subsection 11.2.1.2.2.

Ample surge capacity of the system, provisions for using mobile processing equipment and the low load factor of the processing equipment permits the system to accommodate waste until failures can be repaired and normal plant operation resumed. In addition, the liquid radwaste system is designed to accommodate the anticipated operational occurrences described in subsection 11.2.1.2.3.

11.2.1.2.2 Failure Tolerance

11.2.1.2.2.1 Pump Failure

Where operation is not essential and surge capacity is available, a single pump is provided. This applies to most applications in the liquid radwaste system. Two reactor coolant drain tank pumps and two containment sump pumps are provided because the relative inaccessibility of the containment during power operation would hinder maintenance. The containment sump pumps are submersible pumps with permanently lubricated bearings and mechanical seals. To protect them from damage due to loss of suction, each pump is interlocked to stop on a low level condition in the sump. The reactor coolant drain tank pumps are vertical sump type pumps with motors above the reactor coolant drain tank shaft coupled to pumps submersed in the liquid within the reactor coolant drain tank. This arrangement minimizes contamination of the motors and permits removal and maintenance of the motors outside of the radiation area.

Process pumps located outside containment are air-operated, double diaphragm type. These pumps are capable of significant suction lifts, and can thus be located on or near the top of the associated waste tank, with internal suction piping. They can pump slurries with high solids fractions, run deadheaded, and run dry without damage. In addition, they can operate over a wide range of hydraulic conditions by varying the driving air input. This makes it possible to fulfill many different applications with a single pump model, thereby facilitating maintenance and reducing the inventory of spare parts.

11.2.1.2.2.2 Filter or Ion Exchanger Plugging

Instrumentation is provided to give indication of the pressure drop across filters and ion exchangers. Periodic checks of the pressure drops provide indication of equipment fouling, thus permitting corrective action to be taken before an excessive pressure drop is reached. Change of filter cartridges and ion exchange beds is expected to occur based upon radiation survey.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 460.006

Question:

(Section 11.4) Discuss the specific design features provided in the system design to comply with GDCs 60, 63, "Monitoring Fuel and Waste Storage," and 64, "Monitoring Radioactivity Releases," as they relate to (1) Acceptance Criteria in SRP Section 11.4, Paragraph II.C, regarding control of release of radioactive materials to the environment from the plant areas where the solid radwastes are processed, and (2) Acceptance Criteria in SRP Section 11.4, Paragraph II.D, regarding the radioactive waste system being designed for monitoring radiation levels.

Westinghouse Response:

The higher radioactivity solid wastes, such as primary resins and filters, are handled and packaged in the auxiliary building, rooms 12371, 12372, 12373, and 12374. As a part of the auxiliary building, the HVAC for these rooms is supplied by the radiologically controlled area ventilation system (VAS). The VAS is continuously monitored for radioactivity, and if activity above a predetermined setpoint is detected the discharge is automatically diverted to the containment air filtration system (VFS). The VFS provides filtration and additional monitoring before discharge via the plant vent. Liquid drains from the auxiliary building are collected in the auxiliary building sump, and from there routed to the liquid radwaste system (WLS) for processing and monitored discharge. As shown in AP1000 DCD Table 11.5-2, an area radiation monitor (RMS-JE-RE013) is provided in this area.

Lower activity solid wastes are processed and packaged in the radwaste building, rooms 50350, 50351, and 50352. HVAC for these rooms is supplied by the radwaste building HVAC system (VRS), which includes a radiation monitor and alarm before discharge to the plant vent. The floor of the radwaste building is curbed and sloped to ensure all drainage is collected in sumps, which are in turn routed to the auxiliary building sump for processing by the WLS. As shown in AP1000 DCD Table 11.5-2, an area radiation monitor (RMS-JE-RE017) is provided in this area.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.004

Question:

With regard to the radiological consequences analysis of the design basis Small Line Break Outside Containment as discussed in Chapter 15.6.2 and Table 15.6.2-1 of the AP1000 DCD. What is the basis for the assumed leak flashing fraction of 0.41?

Westinghouse Response:

The basis for the flashing fraction of 0.41 is that the primary coolant leakage is assumed to be at the Reactor Coolant System average temperature of 573.6°F (no credit taken for any reduction in temperature during transit to the point of release). Assuming an isenthalpic reduction in pressure results in 41% of the leakage flashing to steam.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.006

Question:

Please provide the following information in regard to the radiological consequences analysis of the design-basis loss-of-coolant accident (LOCA) as discussed in Chapter 15.6.5.3 and Table 15.6.5-2 of the AP1000 design control document (DCD):

A. How was the main control room activity level (2.0E-6 Ci/m^3 [Curies-per-cubic meter] of dose equivalent I-131) and time (0.2622 hours) at which the emergency habitability system is actuated determined? What assumptions were made in the determination of these values?

B. What is the basis for the control room unfiltered in-leakage assumption of 5.0 cfm (cubic feet-per-minute)?

C. What assumptions and inputs were used to calculate the LOCA doses in the control room due to radiation from adjacent structures and sky-shine?

D. General Design Criterion 19, in 10 CFR Part 50, Appendix A, requires that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposure in excess of 5 rem total effective dose equivalent (TEDE), for the duration of the accident. In Section 6.4.4 of the DCD you state that the doses to main control room personnel associated with the LOCA are bounding for all other design basis accidents. The only documented control room dose analysis is for the LOCA, found in Section 15.6.5. Please provide analyses of the doses to main control room personnel associated with the remaining design basis accident radiological analyses, or alternatively provide a discussion why the LOCA control room dose analysis is bounding for each remaining Chapter 15 radiological analysis.

Westinghouse Response:

A. The actuation setpoint of 2.0E-6 Ci/m^3 of dose equivalent I-131 is the same value that was used for the AP600. The setpoint was selected for the AP600 as a concentration which, if initially present in the control room, would give a dose to control room operators of no more than 0.5 rem TEDE over an exposure duration of 30 days (assuming a 50% occupancy factor and an ongoing clean air purge from the emergency habitability system). No credit was taken for the large air exchange when the control room is opened after three days and which would rapidly deplete the activity remaining in the control room.

The time of 0.2622 hr for having the emergency habitability system in service was determined by first determining the point in time that the concentration of I-131 in the control room would reach the actuation setpoint and then adding 30 seconds for system actuation time. The concentration in the control room was modeled using the assumptions and inputs detailed in DCD Table 15.6.5-1 to determine the rate of release of activity due to containment leakage and the introduction of activity into the control room. Accidents involving an earlier release of significant levels of activity will have a shorter time to reach the actuation setpoint.

B. For the AP1000 design (as for the AP600) there is essentially no unfiltered inleakage to the control room during operation of the safety-related emergency control room habitability system (VES) except for that due to ingress and egress. Reference 470.006-1 specifies an ingress/egress inleakage of 10 cfm for a design lacking a vestibule and then goes on to say that with a vestibule the inleakage would be reduced or eliminated. The AP1000 control room design is the same as the AP600, and does contain a vestibule that will significantly reduce ingress/egress. The model for specifying the amount of air leaking into the control room due to ingress/egress is based on defining what is meant by unlimited access to the control room. Truly unlimited access would involve having the control door constantly open which is not consistent with prudent operation nor is it consistent with any projected situation. To be credible, "unlimited access" must be defined by a limited number of ingress/egress operations. It was conservatively assumed that for unlimited access there would be one ingress or egress occurring every 30 minutes (i.e., 48 times in a 24-hour period).

The air volume of the vestibule to the control room is 365 cubic feet but only a portion of that volume would exchange with the air in the control room during ingress or egress. It is assumed that 150 cubic feet of air in the vestibule would exchange (this is approximately 40 percent of the activity in the vestibule). This results in an equivalent of 5.0 cfm of unfiltered outside air entering the control room. It is noted that the number of personnel passing into or out of the control room at any time is variable. There would be a minimum of one to a reasonable maximum of six to eight (the vestibule can accommodate more than eight but it is not reasonable to assume that many more would be sharing the same ingress or egress unless administrative controls forced them to do so). If only one or two people pass through the door, there would be a smaller air transfer and, if there are eight or more people, it would be expected that more than half of the air in the vestibule might exchange with the air in the MCR. The assumption of 150 cubic feet per ingress/egress operation is believed to be conservative.

Reference 470.006-1: K. G. Murphy and K. W. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," published in *Proceedings of 13th AEC Air Cleaning Conference,* U. S. Atomic Energy Commission, August 1974.

C. The values for sky-shine dose and dose due to direct radiation from adjacent structures were determined for the AP1000 using the detailed analysis that was performed for the AP600 and conservatively adjusting those doses to take into account the increase in core activity associated with the AP1000. The nuclides of concern for calculating doses increased by less than a factor of two and a factor of two was thus conservatively applied. This adjustment resulted in:

Sky-shine dose	2(0.0612) = 0.1224 rem	(this value was increased to 0.15 rem for reporting in the DCD)
Dose from adjacent structures	2(0.002) = 0.004 rem	(this value was increased to 0.01 rem for reporting in the DCD – thus, there was no change from the AP600 reported value)

D. Control room doses were calculated for all of the design basis accidents for which radiological consequences were determined.

Large Break LOCA	4.6 rem TEDE
Fuel Handling Accident	0.8 rem TEDE
Steam Generator Tube Rupture	
(Pre-existing iodine spike)	4.7 rem TEDE
(Accident-initiated iodine spike)	2.0 rem TEDE
Steam Line Break	
(Pre-existing iodine spike)	3.1 rem TEDE
(Accident-initiated iodine spike)	4.1 rem TEDE
Rod Ejection Accident	3.0 rem TEDE
Locked Rotor Accident	2.9 rem TEDE
Small Line Break Outside Containment	0.6 rem TEDE

The statement in DCD Section 6.4.4 that the dose to control room operators for the large break LOCA is seen to be in error. The control room dose calculated for the SGTR is slightly greater. For all events the calculated control room doses are within the dose acceptance limit of 5.0 rem TEDE.

Design Control Document (DCD) Revision:

From DCD Section 6.4.4:

6.4.4 System Safety Evaluation

~~Doses to main control room personnel resulting from a postulated LOCA are presented in subsection 15.6.5.3. The doses associated with the LOCA are bounding for all other design basis accidents. No radioactive materials are stored or transported near the main control room pressure boundary.~~

Doses to main control room personnel were calculated for the following accidents:

Large Break LOCA	4.6 rem TEDE
Fuel Handling Accident	0.8 rem TEDE
Steam Generator Tube Rupture	
(Pre-existing iodine spike)	4.7 rem TEDE
(Accident-initiated iodine spike)	2.0 rem TEDE
Steam Line Break	
(Pre-existing iodine spike)	3.1 rem TEDE
(Accident-initiated iodine spike)	4.1 rem TEDE
Rod Ejection Accident	3.0 rem TEDE
Locked Rotor Accident	2.9 rem TEDE
Small Line Break Outside Containment	0.6 rem TEDE

For all events the dose is within the dose acceptance limit of 5.0 rem TEDE. The details of analysis assumptions for modeling the doses to the main control room personnel are delineated in the LOCA dose analysis discussion in subsection 15.6.5.3.

No radioactive materials are stored or transported near the main control room pressure boundary.

As discussed and evaluated in subsection 9.5.1, the use of noncombustible construction and heat and flame resistant materials throughout the plant reduces the likelihood of fire and consequential impact on the main control room atmosphere. Operation of the nuclear island nonradioactive ventilation system in the event of a fire is discussed in subsection 9.4.1.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: **470.007**

Question:

All Chapter 15 design-basis accident radiological analyses include a discussion of additional radiological consequences of spent fuel pool boiling that may occur coincident with the accident. What assumptions and inputs were used to calculate the radiological consequences as a result of spent fuel pool boiling?

Westinghouse Response:

The following assumptions and inputs specific to modeling the activity release from spent fuel boiling were used:

Parameter	Value	Notes
Initial activity in spent fuel pool I-131 Other nuclides	 3.18 Ci None modeled	It is conservatively assumed that all activity is I-131. The activity is based on the concentration that will result in a radiation field of 2.5 mrem/hr at the pool surface.
Fuel stored in the spent fuel pool	See note	It is assumed that the spent fuel from ten years of operation is in the spent fuel racks, including a region (68 fuel assemblies) from a recent refueling. Based on a nominal 18 month fuel cycle, there are regions that have the following decay intervals: 399 hours, 1.5 years, 3 years, 4.5 years, 6 years, 7.5 years, and 9 years.
Amount of I-131entering the pool over a 30-day period due to diffusion from fuel rods containing cladding defects	1.94 Ci	Although the release of activity to the water pool takes place over the 30 day interval, the activity is conservatively assumed to all be present in the pool at the onset of pool boiling.
Initial pool water temperature	120 °F	
Time to initiate pool boiling	8.8 hr	

Parameter	Value	Notes
Steaming rate 8.8 hr 24 hr 48 hr 72 hr 88 hr 120 hr ≥168	 16,200 lb/hr 16,000 15,700 15,420 15,250 14,930 14,500	Continued reduction in decay heat generation after 168 hours (7 days) was not modeled. The steaming rates conservatively assume that all heat loss is through steaming. Credit was not taken for termination of steaming after 7 days although support from offsite could be credited at that point.
Partition coefficient for iodine	0.01	
Spent fuel pool water mass 8.8 hr 24 hr 48 hr 72 hr ≥88 hr	 1.19E6 lb 1.05E6 lb 8.50E5 lb 6.16E5 lb 4.40E5 lb	Although makeup water would be available earlier, no credit is taken until 88 hours when the water level has dropped to the top of the stored fuel assemblies. The water level in the pool is assumed to be maintained at that level with no credit for the makeup water increasing the level.
Atmospheric dispersion factors at LPZ boundary 8 – 24 hr 24 – 96 hr >96 hr	 1.0E-4 sec/m^3 5.4E-5 sec/m^3 2.2E-5 sec/m^3	There are no releases modeled prior to 8.8 hours when boiling starts.
Atmospheric dispersion factors at CR intake 8 – 24 hr 24 – 72 hr 72 – 96 hr >96 hr	 3.0E-4 sec/m^3 3.0E-4 sec/m^3 1.0E-3 sec/m^3 9.0E-4 sec/m^3	There are no releases modeled prior to 8.8 hours when boiling starts. The intake point is initially the entrance to the control room. After 72 hours the intake point is the normal air intake.
Nuclide data	See DCD Table 15A-4	
Offsite breathing rate 8 – 24 hr >24 hr	 1.8E-4 m^3/sec 2.3E-4 m^3/sec	

Additional assumptions associated with modeling the control room are as described for the LOCA dose analysis (DCD Table 15.6.5-2).

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.008

Question:

Table 15A-2 lists the iodine appearance rates used in design-basis accident radiological consequences analyses that include iodine spiking. How were the RCS iodine appearance rates calculated?

Westinghouse Response:

The iodine appearance rates are calculated based on the plant operating in an equilibrium condition in which the appearance of iodine activity in the coolant (from fuel rods with cladding defects) is balanced by the removal of activity through radioactive decay, cleanup by processing through the letdown demineralizer, and postulated leakage. It is assumed that the plant is operating at the Technical Specification limit of 1.0 μCi/g Dose Equivalent I-131. At this condition, the distribution of iodine isotopes has been defined for the AP1000 to be:

I-130	1.097E-2 μCi/gm
I-131	7.241E-1
I-132	9.589E-1
I-133	1.313E0
I-134	2.236E-1
I-135	7.959E-1

The RCS activity (A) is related to the appearance rate by:

A = Iodine Appearance Rate (P) / Removal rate (λ)

where $\lambda = \lambda_{purification} + \lambda_{decay}$

with $\lambda_{purification} = [(1 - 1/DF)(F) + L] / V$

Where DF = Maximum Decontamination Factor provided for iodine
DF = infinite (i.e., complete removal)
Thus 1/DF = 0

F = Letdown flow rate
This is 100 gpm but and this is conservatively increased by 10% to 110 gpm (cold conditions are assumed so density = 62.4 lb/ft^3)

F = (110 gpm)(0.13368 ft^3/gal)(62.4 lb/ft^3) = 917.6 lb/min
= 55,056 lb/hr

L = Leakage from the primary coolant system
= 10.5 gpm at cold conditions from Tech Spec 3.4.8 - assume cold water density of 62.4 lb/ft^3
= (10.5 gpm)(0.13368 ft^3/gal)(62.4 lb/ft^3) = 87.59 lb/min
= 5255 lb/hr

V = RCS water mass

Value is based on nominal cold volume of 10,621 ft^3 of which 2100 ft^3 is the pressurizer and is assumed to be half full. Using the vessel average temperature of 573.6°F and system pressure of 2250 psia, the density is 45.25 lb/ft^3.

$$10{,}621 - 0.5(2100) = 9571 \text{ ft}^3$$

V = (9571)(45.25) = 433,088 lb

Thus, $\lambda_{purification}$ = (55,056 + 5255) / 433,088 = 0.13926 hr^{-1}

Values for λ_{decay} are:

I-130	0.0561 hr^{-1}
I-131	0.00359 hr^{-1}
I-132	0.301 hr^{-1}
I-133	0.0333 hr^{-1}
I-134	0.791 hr^{-1}
I-135	0.105 hr^{-1}

Combining the decay term with the purification term to create a total removal term for each isotope gives the total removal term:

$\lambda_{I\text{-}130}$ = 0.13926 + 0.0561 = 0.19536 hr^{-1}
$\lambda_{I\text{-}131}$ = 0.13926 + 0.00359 = 0.14285 hr^{-1}
$\lambda_{I\text{-}132}$ = 0.13926 + 0.301 = 0.44026 hr^{-1}
$\lambda_{I\text{-}133}$ = 0.13926 + 0.0333 = 0.17256 hr^{-1}
$\lambda_{I\text{-}134}$ = 0.13926 + 0.791 = 0.93026 hr^{-1}
$\lambda_{I\text{-}135}$ = 0.13926 + 0.105 = 0.24426 hr^{-1}

The RCS Inventory is:

$A_{I\text{-}130}$ = (0.01097 μCi/g)(433,088 lb)(453.6 g/lb) = 2.16 Ci
$A_{I\text{-}131}$ = (0.7241 μCi/g)(433,088 lb)(453.6 g/lb) = 142.2 Ci
$A_{I\text{-}132}$ = (0.9589 μCi/g)(433,088 lb)(453.6 g/lb) = 188.4 Ci
$A_{I\text{-}133}$ = (1.313 μCi/g)(433,088 lb)(453.6 g/lb) = 257.9 Ci
$A_{I\text{-}134}$ = (0.2236 μCi/g)(433,088 lb)(453.6 g/lb) = 43.9 Ci
$A_{I\text{-}135}$ = (0.7959 μCi/g)(433,088 lb)(453.6 g/lb) = 156.4 Ci

Equilibrium appearance rate is then calculated by $P = A\lambda$

$P_{I\text{-}130}$ = (2.16 Ci)(0.19536 hr^{-1}) / (60 min/hr) = 7.03E-3 Ci/min
$P_{I\text{-}131}$ = (142.2 Ci)(0.14285 hr^{-1}) / (60 min/hr) = 0.339 Ci/min
$P_{I\text{-}132}$ = (188.4 Ci)(0.44026 hr^{-1}) / (60 min/hr) = 1.38 Ci/min
$P_{I\text{-}133}$ = (257.9 Ci)(0.17256 hr^{-1}) / (60 min/hr) = 0.742 Ci/min
$P_{I\text{-}134}$ = (43.9 Ci)(0.93026 hr^{-1}) / (60 min/hr) = 0.681 Ci/min
$P_{I\text{-}135}$ = (156.4 Ci)(0.24426 hr^{-1}) / (60 min/hr) = 0.637 Ci/min

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.009

Question:

(Appendix 15B of the AP1000 DCD) The staff plans to perform an independent Monte Carlo analysis of the uncertainties in prediction of the aerosol removal coefficients for the AP1000 design, to determine if use of the previously determined AP600 values is acceptable. For use in this analysis, please provide:

A. containment geometry inputs for the AP1000 design (volume, upward facing surface area, etc),

B. upper and lower values of the thermal-hydraulic containment parameters relevant to the aerosol removal, i.e., pressure, temperature, relative humidity, steam/air ratio, and steam condensation rate on the heat sinks, and

C. ranges of non-safety related containment spray flow rate and spray droplets distribution.

Westinghouse Response:

A. The upward-facing area inside containment that is available for sedimentation removal is the same as for the AP600 (2900 m^2). The specified surface area does not include any surfaces that are in dead-ended volumes (e.g., accumulator rooms or CVS room) or flooded areas (e.g., the reactor cavity). Also the surfaces directly above the steam generator compartments were excluded due to concerns about potential resuspension of deposited aerosols by the super-heated steam plume.

   The available volume for aerosol mixing is 55,250 m^3. Dead-ended volumes and flooded volumes are not included.

B. Releases occur during LOCA events. For the AP1000 design basis containment analysis, the following conditions were observed for the double-ended cold leg and double-ended hot leg breaks.

Containment temperature	120F – 267F
Containment pressure	14.7 psia – 70 psia
Containment Steam Pressure Ratio	0 – 0.7
Containment Relative Humidity	22% - 100%
Condensation Heat Transfer Coeff.	0 – 140 Btu/hr-ft2-F

C. The AP1000 containment spray is a nonsafety-related function that is performed by the fire protection system. Containment spray has not been credited in the calculation of the aerosal removal coefficients for AP1000, and is not credited in the Chapter 15 accident analyses. Westinghouse does not have a calculated range of flow rates for the nonsafety-related containment spray system. A description of the performance of the nonsafety-related containment spray system is included in DCD Section 6.5.2.2. The nominal containment spray flow rate and spray droplet distribution are provided in the DCD and are summarized below:

Nominal spray flow rate:	1034 gpm
Number of spray nozzles:	68
Nominal spray flow rate per nozzle:	15.2 gpm
Nozzle pressure drop at nominal flow:	40 psid
Mass mean droplet size:	1000 microns
Total free volume of the sprayed region:	1.7×10^6
Containment free volume coverage:	84%
Containment pressure at nominal spray flow rate:	20 psig
Developed head of the fire pumps at nominal spray flow:	385 ft

Westinghouse has not performed a calculation of the spray flow rate for a range of containment conditions.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.010

Question:

(Appendix 15B, paragraph 15B2.4.2) Please provide references for the chosen aerosol size distribution parameters (i.e., radius of 0.22 micrometers and standard deviation of 1.81). Also, please justify the statement that "sensitivity of aerosol removal coefficient calculations to these values is small" since, in general, the opposite is true.

Westinghouse Response:

The selected aerosol size was derived from an evaluation of a large number of aerosol distributions measured in a variety of degraded-fuel tests and experiments. This aerosol size reflects a reevaluation of the results presented in Reference 1 (which had identified an aerosol radius of 0.57 μm). The re-interpretation of the data to give a smaller radius is an increase in conservatism since aerosol removal rates decrease as the aerosol size decreases.

Note that the use of a packing fraction of 0.8 (as discussed in DCD Appendix 15B) increases the nominal radius from 0.22 to 0.232 μm.

A sensitivity study performed during the development of the aerosol removal coefficients showed the following relationship of particle size to the total mass of particulates released to the environment due to containment leakage.

Particle Radius	Mass release (relative to the base case of 0.232 μm (Std. Dev. of 1.81)
0.05 μm (Std. Dev. of 1.81)	5% greater
0.1μm (Std. Dev. of 1.81)	6% greater
0.4 μm (Std. Dev. of 1.81)	14% less

The changes in mass released from the containment reflects the sensitivity of the aerosol removal coefficients to aerosol size. Within the range of aerosol size considered, the sensitivity of the removal coefficients is demonstrated to be small.

Reference: 1. Sher, R., M. D. Hoover, G. J. Newton, and F. J. Rahn, "Aerosol Behavior in Nuclear Facilities," Transactions of the American Nuclear Society, Volume 70, p. 260 (June 1994).

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.011

Question:

(Appendix 15B, paragraph 15B.2.6) The paragraph presents a qualitative discussion of the differences between AP600 and AP1000 designs concluding that the use of the AP600 removal coefficients is conservative. Please, provide either a sample calculation, or an analytical justification for this conclusion. Also, one potentially important difference is omitted, i.e., the increased height of the AP1000 containment. It is known that the increased height decreases the rate of aerosol removal, which would be a non-conservative effect. Please, discuss the significance of this issue.

Westinghouse Response:

The analytical justification is as stated in DCD paragraph 15B.2.6:

- An increase in the power level results in an increase in the amount of decay heat that will be removed by condensation onto the containment shell and heat transfer to the containment shell thus enhancing the removal of particulates by diffusiophoresis and thermophoresis respectively.

- An increase in the projected containment pressure results in an increase in the rate of heat transfer by condensation onto the containment shell and heat transfer to the containment shell thus enhancing the removal of particulates by diffusiophoresis and thermophoresis respectively.

The increase in containment volume would potentially reduce the rate of aerosol removal as a result of a reduction in the aerosol concentration. However, this does not affect the aerosol removal coefficients. The removal coefficients are merely operating in an atmosphere having a lower concentration than would exist if the containment volume was less. Additionally, for the AP1000 there is a ~25% increase in core mass and ~20% increase in containment volume; thus, the aerosol concentration is essentially the same as for the AP600. Also, the increase in containment height will not result in a reduction in sedimentation removal. Sedimentation is a phenomenon that takes place in proximity to upfacing surfaces. The air volume in the containment will be highly mixed by convection as heat is transferred through the containment shell. The mixing does not adversely impact the sedimentation process.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.012

Question:

In the AP1000 PRA (as in the AP600 PRA), a decontamination factor (DF) of 100 has been credited for SGTRs due to impaction on the SG tubes, based on results of an unpublished paper by Jun Li, David Leaver, James Metcalf, entitled "Aerosol Retention During an Unisolated Steam Generator Tube Rupture Severe Accident Even." Please provide justification for this seemingly high DF estimate in light of more recent analyses and or experiments, if available. In addition, please provide a copy of this paper. Also, since this release category is also used for inter-system LOCAs (ISLOCAs), please justify the use of such a high DF, as a general rule, for all bypass events.

Westinghouse Response:

The response to RAI 470.013 provides fission product release fractions from AP1000 MAAP4 analyses. A decontamination factor (DF) of 1 is used for SGTRs in these new analyses.

Design Control Document (DCD) Revision:

None

PRA Revision:

See the response to RAI Number 470.013.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.013

Question:

Please justify why the AP600 release fractions, as calculated by MAAP, should be considered applicable to the AP1000. In addition, please justify ZERO release fraction for tellurium in all release classes (in both chemical forms, TeO_2 and Te_2).

Westinghouse Response:

Release fractions for AP1000 have been calculated by Westinghouse. The results from the analyses will be incorporated into the next revision of the AP1000 PRA Chapter 45 as indicated below. The source term MAAP4.04 analysis results will be incorporated into the next revision of AP1000 PRA Chapter 34 (see RAI 720.042 response). These analyses show Tellurium release in chemical form TeO_2.

Design Control Document (DCD) Revision:

None

PRA Revision:

CHAPTER 45

FISSION-PRODUCT SOURCE TERMS

This chapter discusses the fission-product source terms that are used in the offsite dose analysis (Chapter 49) for each of the release categories, or end states, of the containment event tree, as discussed in Chapter 35. The source terms ~~release fractions for the AP1000 Probabilistic Risk Assessment (PRA)~~ are taken from the ~~AP600 Probabilistic Risk Assessment (Reference 45-1)~~**MAAP4 analyses results presented in Chapter 34.** They are used as the input to the offsite dose analysis presented in Chapter 49.

45.1 Summary of AP1000 Release Categories

The release categories group similar fission-product source terms from the Level 2 analysis to bound the offsite consequences and reduce the number of sequences to be analyzed. The AP1000 prevents large releases with design features of the containment that provide redundant, diverse mitigation of challenging phenomena in the unlikely event of a severe accident. These features include the reactor coolant automatic depressurization system (ADS), the ability to flood the reactor vessel cavity, hydrogen igniters in the large dry containment, and the passive containment cooling system (PCS). The design features act to maintain reactor coolant system (RCS) integrity, prevent containment overpressurization from hydrogen detonation or deflagration, and remove heat from the containment. The mitigation features maintain the potential for fission-product release from the AP1000 containment very low.

Given a severe accident, a release of fission products occurs through normal containment leakage, a breach of the containment or a bypass of the containment.

A large pre-existing opening, or containment isolation failure, produces containment leakage beyond the design basis. The failure of a large purge line isolation valve is assumed for containment isolation failure.

A breach of the containment shell is assumed to occur based on the containment pressure and the conditional containment failure probability discussed in Chapter 42. The containment is also assumed to fail if hydrogen detonation occurs.

Containment bypass in the AP1000 is typically caused by steam generator tube rupture initiated events that progress to severe accident or by steam generator tube ruptures induced by high pressure and temperature core damage events.

The containment release categories are described in Section 35.6. For each of these categories, the release fractions are determined over a 72-hour period following the onset of core damage. For all containment failure release categories, the release is assumed to be directly from the containment to the environment at ground level. For the intact containment (IC) ~~and bypass (BP)~~ categor~~ies~~, a decontamination factors (DF) ~~are~~ of three is applied. ~~For the IC category, a DF of 3 is applied~~ (Reference 45-2) due to deposition in the auxiliary building. ~~For the BP category, a DF of 100 is assumed due to impaction on the tubes in the secondary system.~~

For each of the release categories, a representative source term is used to complete the Level 3 analysis. This representative source term was identified as the bounding release for the accident sequences in the specific release category.

The release fractions at 24 and 72 hours for each fission-product group are presented in Tables 45-1 and 45-2, respectively.

~~The release fractions used in the AP1000 PRA were determined in the AP600 PRA source term analysis (Reference 45-1). AP1000 employs the same release categories and release category definitions and timing. The overall AP1000 plant response to severe accidents is very similar to the AP600, and therefore the release fractions and timing for the AP1000 are expected to be approximately the same as the AP600. For the offsite dose analysis in Chapter 49, the release of radiation to the environment is calculated using the release fractions and the AP1000 fission product inventory for each isotope.~~

45.2 Release Category Source Terms

~~For each of the release categories, a representative source term is used to complete the Level 3 analysis. This representative source term was identified in Reference 45-1 as the bounding release for the accident sequences in the specific release category.~~

~~The release fractions at 24 and 72 hours for each fission product group are presented in Tables 45-1 and 45-2, respectively.~~

45.2.1 Release Category IC

Release Category IC represents the release of fission products from an intact containment during a severe accident. Normal containment leakage accounts for the fission-product releases to the environment. The likely normal leakage release pathway is via containment penetration leakage into the auxiliary building.

The fission product release fractions from an accident class 3BE sequence with cavity flooding, hydrogen control, and passive containment cooling are used to represent the IC release. A decontamination factor of 3 is applied to the aerosol-release fractions to model deposition in the auxiliary building. A direct release sensitivity analysis to the decontamination factor is ~~performed~~discussed in section 45.3.

The source term releases for Release Category IC are presented in Figures 45-1 through 45-12.

45.2.2 Release Category BP

Release category BP represents containment bypass releases to the environment. Fission products are released from the reactor coolant system via failed steam generator tubes to the secondary system and to the environment through a stuck-open safety valve. Release category BP contributes to the large, early release frequency (LERF) of the AP1000. The fission product release fractions from a steam generator tube rupture induced by high pressure core damage in accident class 1A are used to represent the BP release. ~~As discussed previously, a decontamination factor of 100 is applied to the aerosol release fractions to model impaction on the tubes in the secondary system.~~

The source term releases for Release Category BP are presented in Figures 45-13 through 45-24.

45.2.3 Release Category CI

Release category CI represents fission product releases to the environment from an unisolated containment. Fission products are released from the reactor coolant system to the containment; however, the containment is not isolated from the environment from the beginning of the accident. Release category CI contributes to the LERF of the AP1000.

The fission product release fractions from an accident class ~~3BE~~-3C, sequence with the failure of containment isolation are used to represent the CI release category.

The source term releases for Release Category CI are presented in Figures 45-25 through 45-36.

45.2.4 Release Category CFE

Release category CFE represents fission product releases to the environment from containment failure induced by severe accident phenomena that may occur during the core melting and relocation phase of the accident sequence. Fission products are released from the reactor coolant system to the containment. Before significant deposition of the aerosol fission products, the containment fails due to a high-energy event (i.e. hydrogen combustion or steam explosion). Release category CFE contributes to the LERF of the AP1000.

The fission product release fractions from an accident class ~~3BE~~-3D sequence with early containment failure induced by hydrogen detonation were used to represent release category CFE.

The source term releases for Release Category CFE are presented in Figures 45-37 through 45-48.

45.2.5 Release Category CFI

Release category CFI represents fission product releases to the environment from containment failure that may occur after the melting and relocation phenomena and within 24 hours after the onset of core damage. Fission products are released from the reactor coolant system to the containment. The containment atmosphere is well-mixed and significant aerosol deposition has begun when the containment fails due to severe accident phenomena (i.e. hydrogen combustion or long-term containment pressurization from decay heat). Release category CFI contributes to the large release frequency of the AP1000, but is not an early release contributing to LERF.

The fission product release fractions from an accident class 3BE sequence with intermediate containment failure induced by hydrogen detonation were used to represent release category CFI.

The source term releases for Release Category CFI are presented in Figures 45-49 through 45-60.

45.2.6 Release Category CFL

Release category CFL represents fission product releases to the environment from containment failure that may occur after 24 hours. Fission products are released from the reactor coolant system to the containment. The containment atmosphere is pressurized with steam from decay heat. Significant aerosol deposition occurs over the long term of the accident. Containment fails from overpressure due to loss of containment cooling. Release category CFL contributes to large release frequency, but is not an early release contributing to LERF.

The fission product releases from an accident class 3BE sequence ~~with loss of passive containment cooling~~ and containment failure induced by long-term containment pressurization is used to represent release category CFL.

The source term releases for Release Category CFL are presented in Figures 45-61 through 45-72.

45.2.7 Release Category CFV

Release category CFV represents fission product releases to the environment from containment venting, which occurs after 24 hours. Fission products are released to the containment. The containment atmosphere is pressurized with steam from decay heat. Significant aerosol deposition occurs over the long term of the accident. The operator vents the containment at a pressure well below the failure pressure of the containment. No filtering is assumed. Release category CFV contributes to large release frequency, but is not an early release contributing to LERF.

Release category CFV did not exist for the AP600 PRA (Reference 45-1). The failure frequency for successful operator venting in the containment event tree analysis is unity. Therefore, the frequency of CFV in the AP1000 PRA is zero occurrences per reactor year. No source term is calculated for this release category.

45.3 Direct-Release Sensitivity Case

For release category IC, the leakage from the containment is assumed to pass through the middle annulus of the auxiliary building. This room has restricted leakage to the environment. Thus, the fission products have long residence times. Significant deposition occurs in the middle annulus, attenuating the release of fission products to the environment. A decontamination factor of three is credited for the aerosol fission products because of this deposition.

To account for uncertainty in the probability of the fission products bypassing the middle annulus decontamination effect, the release is assumed, in the IC direct-release sensitivity case, to be released directly to the environment at the design leak rate.

The source term releases for the Direct Release Sensitivity Case are presented in Figures 45-73 through 45-84.

45.4 Summary

The AP1000 release categories and associated source terms over the first 24 and 72 hours after core damage are summarized in Tables 45-1 and 45-2, respectively. ~~For the IC release category, the releases are very similar over the first 24 hours after core uncovery. This is because the containment remains intact. The release at 24 hours after core damage for each of these sequences is based upon design basis containment leakage area. The differences in the releases among release categories occur over the long term as the containment pressurizes.~~

~~The BP, C1, CFE, and CF1 release categories represent large releases that occur prior to 24 hours after the onset of core damage. Release category CFL represents large releases that occur after 24 hours. Release category CFV represents containment venting releases, which occur after 24 hours, but have not been credited in the AP1000 PRA.~~

A fission-product release source term (direct) is also developed to address the sensitivity assuming that the IC source term from the containment is released directly to the environment with no holdup or decontamination in the auxiliary building. The sensitivity release fractions also represent source terms for consequence analysis, as discussed in Chapter 49.

45.5 References

45-1 GW-GL-021, AP600 Probabilistic Risk Assessment, August 1998.

45-2 EPRI Letter to James Wilson, USNRC, dated April 30, 1993, attachment titled, "Passive ALWR Secondary Building Mixing and Leak Rate Monitoring."

Table 45-1

ENVIRONMENTAL RELEASE FRACTIONS AT 24 HOURS AFTER CORE DAMAGE PER RELEASE CATEGORY

Release Cat.	Environmental Release Fractions at 24 Hours After Core Damage											
	Xe, Kr	CsI	TeO_2	SrO	MoO_2	CsOH	BaO	La_2O_3	CeO_2	Sb	Te_2	UO_2
IC	~~9.2E-4~~ 1.0E-3	~~4.7E-6~~ 1.2E-5	~~0.0E0~~ 9.5E-6	~~2.9E-7~~ 1.1E-5	~~4.7E-6~~ 1.3E-5	~~4.5E-6~~ 1.1E-5	~~3.1E-6~~ 1.2E-5	~~9.1E-7~~ 1.3E-6	~~1.0E-6~~ 1.5E-6	~~8.2E-6~~ 1.3E-5	0.0E0	0.0E0
BP	~~8.6E-1~~ 6.3E-1	~~3.8E-3~~ 1.2E-2	~~0.0E0~~ 1.9E-2	~~2.4E-5~~ 3.0E-4	~~6.9E-4~~ 5.0E-3	~~3.2E-3~~ 1.1E-2	~~2.4E-4~~ 7.9E-4	~~3.9E-6~~ 1.2E-5	~~9.8E-6~~ 3.5E-5	~~6.1E-2~~ 5.5E-2	0.0E0	0.0E0
CI	~~8.4E-1~~ 6.4E-1	~~3.4E-2~~ 4.6E-2	~~0.0E0~~ 2.1E-2	~~2.1E-3~~ 2.0E-2	~~4.1E-2~~ 4.0E-2	~~3.5E-2~~ 1.8E-2	~~2.2E-2~~ 3.2E-2	~~5.1E-3~~ 2.4E-4	~~6.5E-3~~ 7.4E-4	~~6.1E-2~~ 2.7E-2	0.0E0	0.0E0
CFE	~~7.0E-1~~ 8.1E-1	~~8.3E-2~~ 5.7E-2	~~0.0E0~~ 3.2E-2	~~9.6E-4~~ 3.5E-3	~~2.7E-2~~ 1.4E-2	~~8.1E-2~~ 5.5E-2	~~9.9E-3~~ 5.3E-3	~~6.0E-4~~ 6.5E-5	~~8.8E-4~~ 2.5E-4	~~7.2E-2~~ 2.3E-2	0.0E0	0.0E0
CFI	~~6.2E-1~~ 7.5E-1	~~3.4E-3~~ 8.6E-3	~~0.0E0~~ 1.3E-2	~~5.8E-4~~ 1.42E-2	~~7.8E-3~~ 1.7E-2	~~2.9E-3~~ 8.6E-3	~~5.6E-3~~ 1.7E-2	~~1.1E-3~~ 1.2E-3	~~1.5E-3~~ 1.2E-3	~~1.0E-2~~ 1.5E-2	0.0E0	0.0E0
CFL	~~1.1E-3~~ 8.7E-1	~~1.2E-5~~ 3.7E-2	~~0.0E0~~ 4.4E-2	~~5.9E-7~~ 2.0E-3	~~1.1E-5~~ 1.0E-2	~~1.1E-5~~ 2.8E-2	~~6.1E-6~~ 2.0E-3	~~1.6E-6~~ 2.3E-5	~~1.9E-6~~ 1.6E-4	~~1.7E-5~~ 2.9E-2	0.0E0	0.0E0
DIRECT	~~9.2E-4~~ 3.0E-3	~~1.4E-5~~ 3.6E-5	~~0.0E0~~ 2.9E-5	~~8.8E-7~~ 3.3E-5	~~1.4E-5~~ 3.9E-5	~~1.3E-5~~ 3.3E-5	~~9.2E-6~~ 2.8E-5	~~2.7E-6~~ 3.9E-6	~~3.0E-6~~ 4.5E-6	~~2.5E-5~~ 3.9E-5	0.0E0	0.0E0

Table 45-2

ENVIRONMENTAL RELEASE FRACTIONS AT 72 HOURS AFTER CORE DAMAGE PER RELEASE CATEGORY

Release Cat.	Environmental Release Fractions at 72 Hours After Core Damage											
	Xe, Kr	CsI	TeO_2	SrO	MoO_2	CsOH	BaO	La_2O_3	CeO_2	Sb	Te_2	UO_2
IC	~~2.8E-3~~ 2.6E-3	~~4.7E-6~~ 1.2E-5	~~0.0E0~~ 9.5E-6	~~2.9E-7~~ 1.1E-5	~~4.7E-6~~ 1.3E-5	~~4.5E-6~~ 1.1E-5	~~3.1E-6~~ 1.2E-5	~~9.1E-7~~ 1.4E-6	~~1.0E-6~~ 1.5E-6	~~9.6E-6~~ 1.3E-5	0.0E0	0.0E0
BP	~~8.6E-1~~ 6.3E-1	~~3.8E-3~~ 1.2E-2	~~0.0E0~~ 1.9E-2	~~2.4E-5~~ 3.0E-4	~~6.9E-4~~ 5.0E-3	~~3.2E-3~~ 1.1E-2	~~2.4E-4~~ 7.9E-4	~~3.9E-6~~ 1.2E-5	~~9.8E-6~~ 3.5E-5	~~6.1E-2~~ 5.5E-2	0.0E0	0.0E0
CI	~~9.7E-1~~ 7.8E-1	~~3.4E-2~~ 4.6E-2	~~0.0E0~~ 2.1E-2	~~2.2E-3~~ 2.0E-2	~~4.1E-2~~ 4.0E-2	~~3.5E-2~~ 1.8E-2	~~2.2E-2~~ 2.2E-2	~~5.1E-3~~ 2.4E-4	~~6.5E-3~~ 7.4E-4	~~6.1E-2~~ 2.9E-2	0.0E0	0.0E0
CFE	~~7.3E-1~~ 9.6E-1	~~8.3E-2~~ 5.7E-2	~~0.0E0~~ 3.2E-2	~~9.6E-4~~ 3.5E-3	~~2.7E-2~~ 1.4E-2	~~8.1E-2~~ 5.5E-2	~~9.9E-3~~ 5.3E-3	~~6.0E-4~~ 6.5E-5	~~8.8E-4~~ 2.5E-4	~~7.2E-2~~ 2.3E-2	0.0E0	0.0E0
CFI	~~6.5E-1~~ 8.5E-1	~~3.5E-3~~ 8.6E-3	~~0.0E0~~ 1.3E-2	~~5.8E-4~~ 1.5E-2	~~7.8E-3~~ 1.7E-2	~~3.2E-3~~ 8.6E-3	~~5.6E-3~~ 1.8E-2	~~1.1E-3~~ 2.1E-3	~~1.5E-3~~ 2.2E-3	~~1.0E-2~~ 1.5E-2	0.0E0	0.0E0
CFL	~~1.0~~ 4.7E-1	~~3.5E-4~~ 3.7E-2	~~0.0E0~~ 4.4E-2	~~6.9E-6~~ 2.0E-3	~~1.3E-4~~ 1.0E-2	~~1.5E-3~~ 2.8E-2	~~6.6E-5~~ 3.0E-3	~~1.7E-5~~ 3.3E-5	~~2.0E-5~~ 1.6E-4	~~8.7E-2~~ 2.9E-2	0.0E0	0.0E0
DIRECT	~~2.8E-3~~ 7.8E-3	~~1.4E-5~~ 3.6E-5	~~0.0E0~~ 2.9E-5	~~8.8E-7~~ 3.3E-5	~~1.4E-5~~ 3.9E-5	~~1.3E-5~~ 3.3E-5	~~9.2E-6~~ 3.6E-5	~~2.7E-6~~ 4.2E-6	~~3.0E-6~~ 4.5E-6	~~2.9E-5~~ 3.9E-5	0.0E0	0.0E0

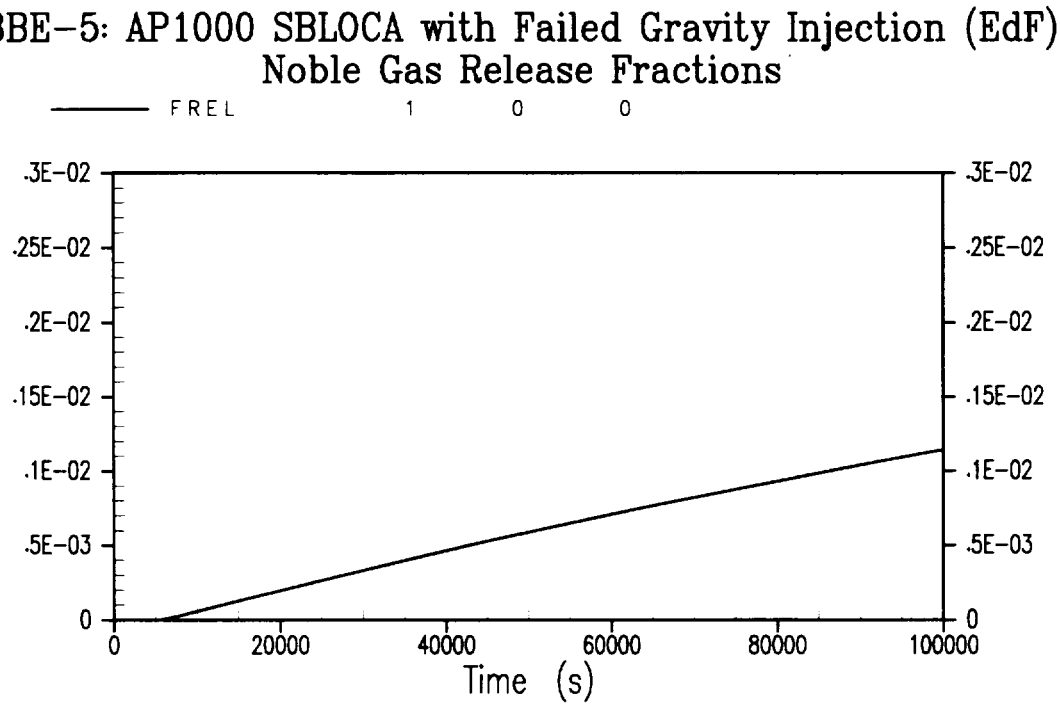


Figure 45-1
Release Category IC, Run 3BE-5: Release Fraction of Noble Gases

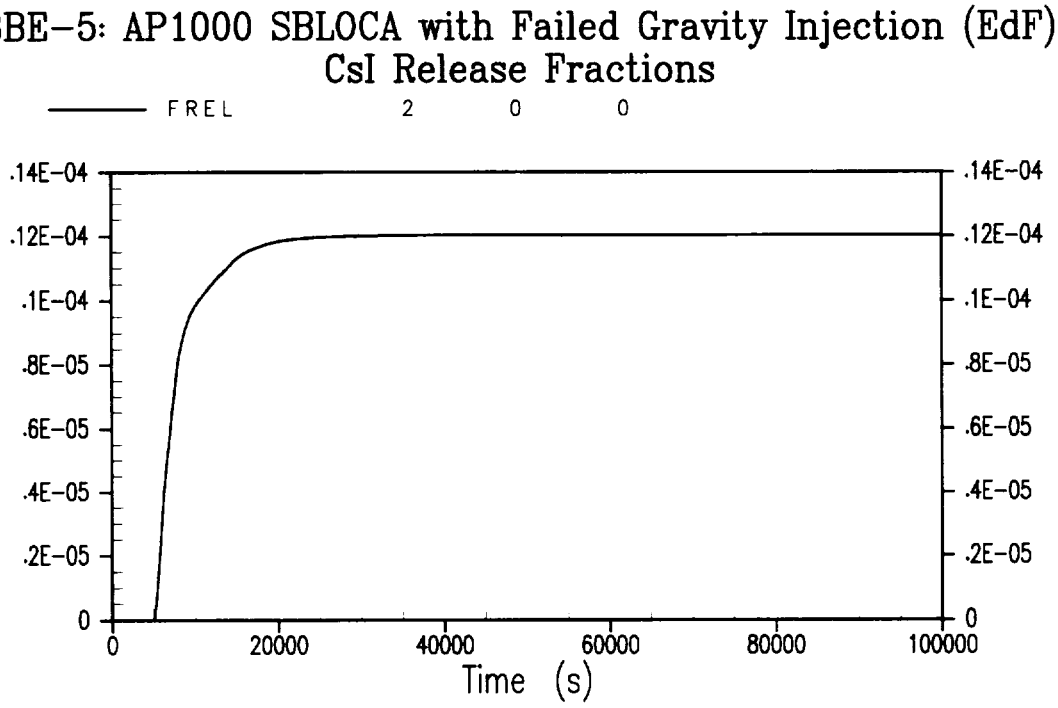


Figure 45-2
Release Category IC, Run 3BE-5: Release Fraction of Cesium Iodide

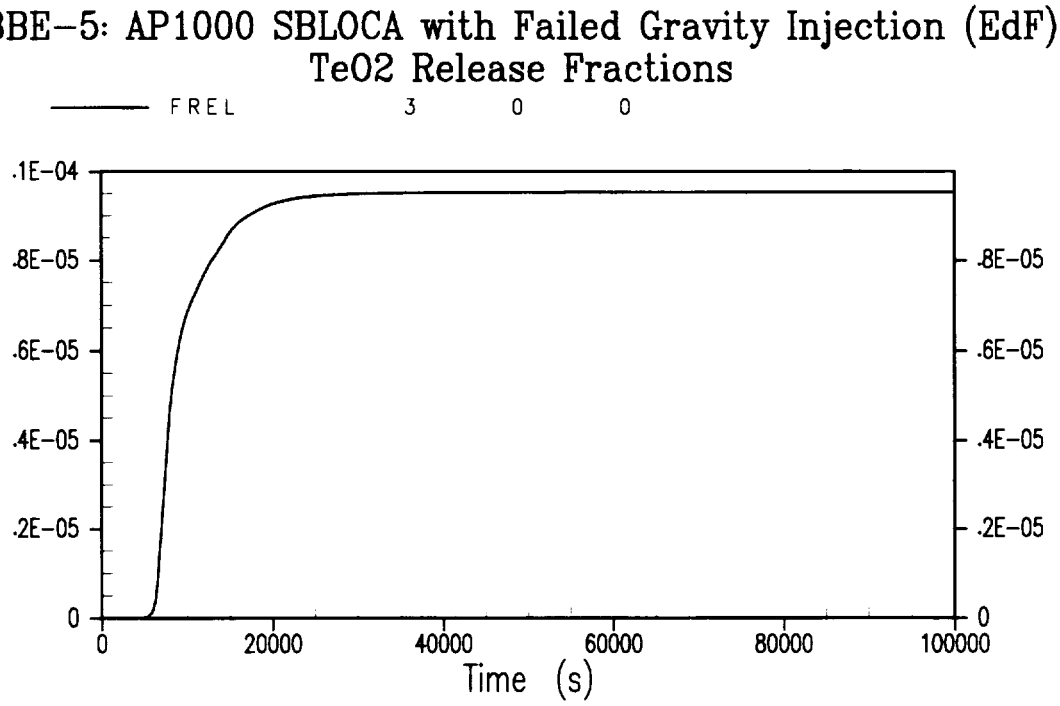


Figure 45-3
Release Category IC, Run 3BE-5: Release Fraction of Tellurium Dioxide

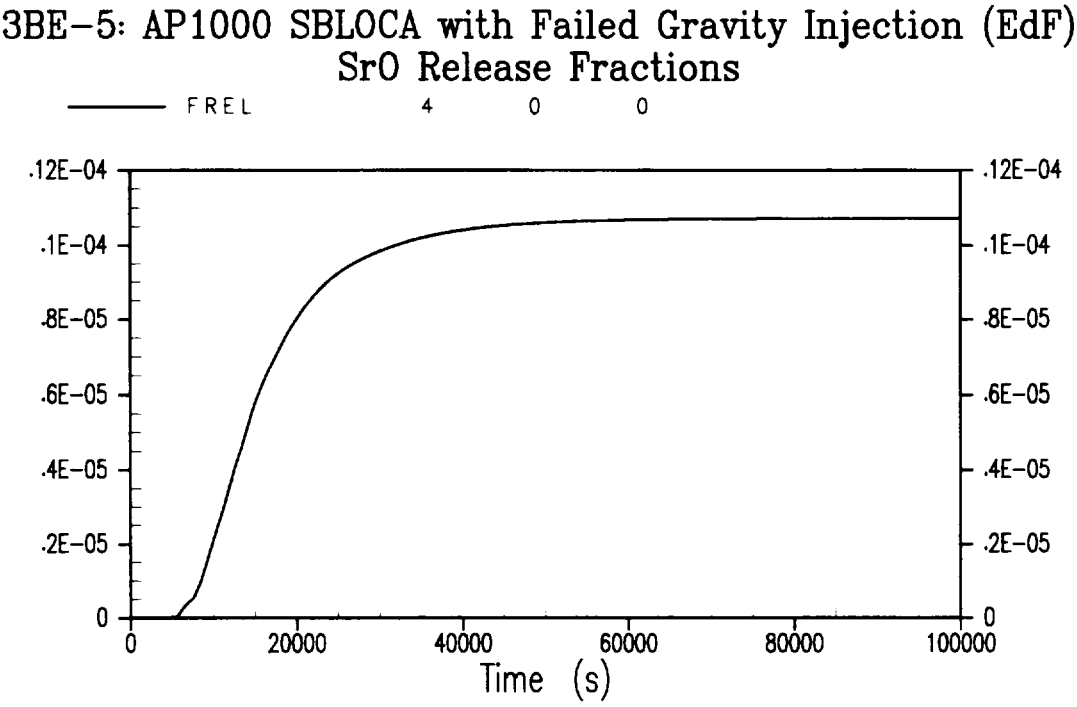


Figure 45-4
Release Category IC, Case 3BE-5: Release Fraction of Strontium Oxide

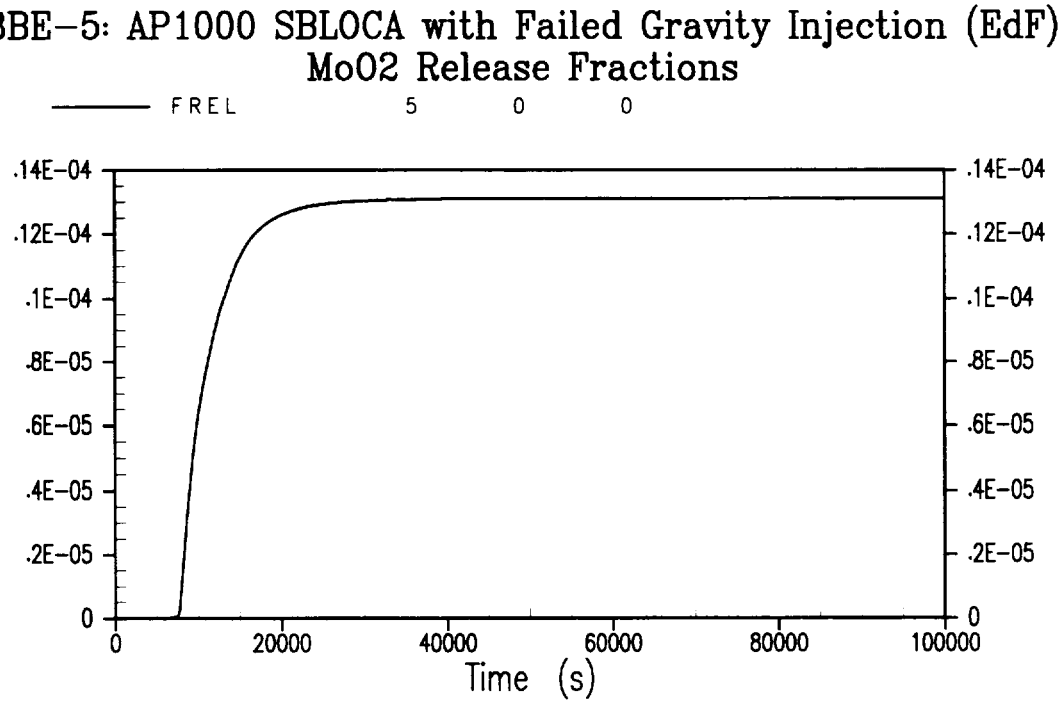


Figure 45-5
Release Category IC, Run 3BE-5: Release Fraction of Molybdenum Dioxide

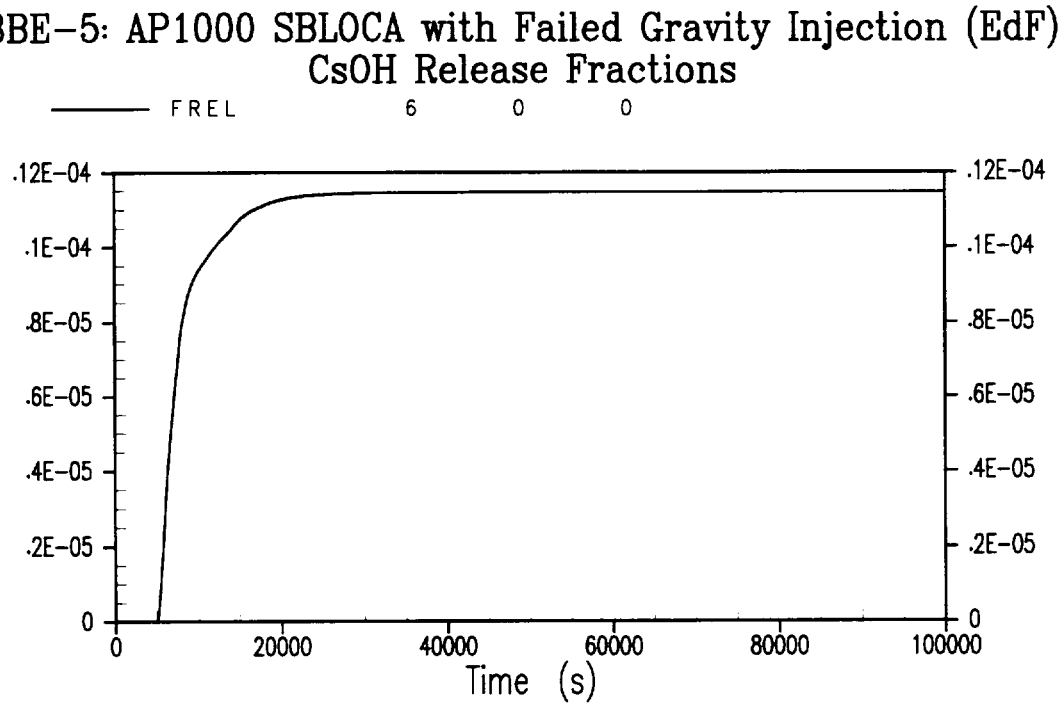


Figure 45-6
Release Category IC, Run 3BE-5: Release Fraction of Cesium Hydroxide

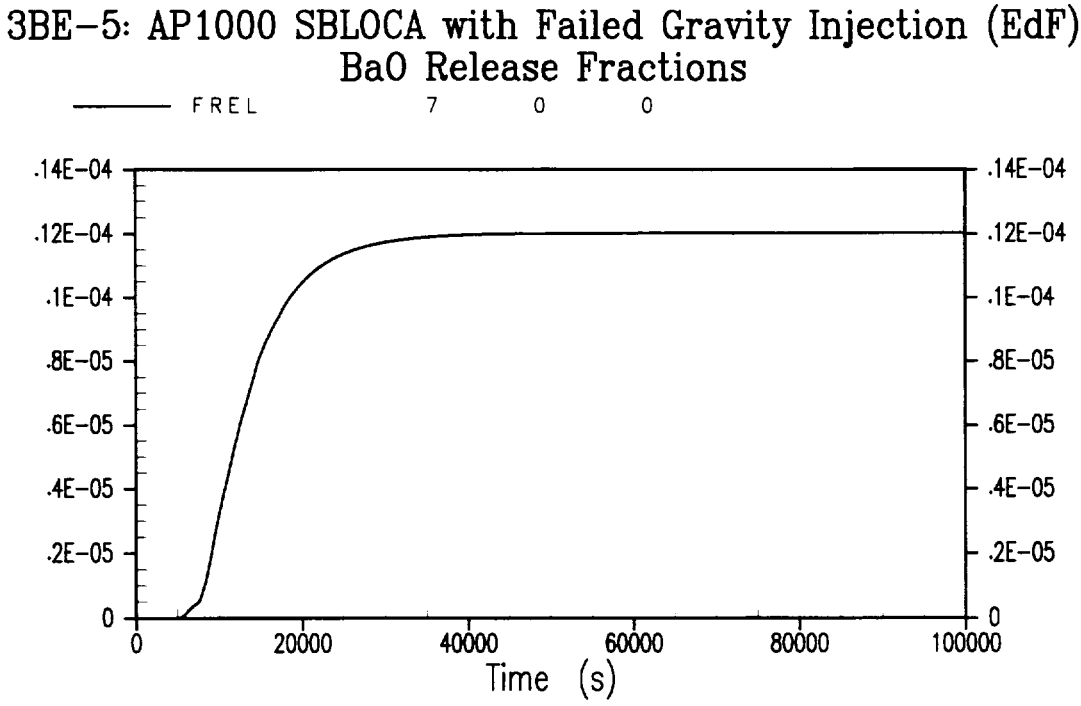


Figure 45-7
Release Category IC, Run 3BE-5: Release Fraction of Barium Oxide

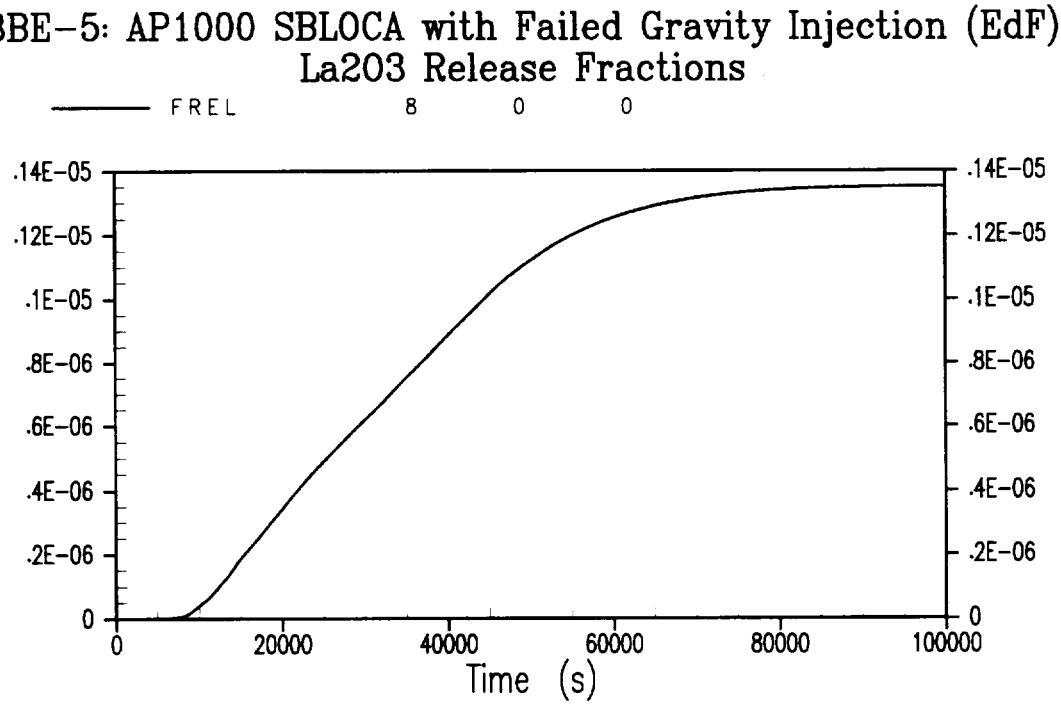


Figure 45-8
Release Category IC, Run 3BE-5: Release Fraction of Dilanthanum Trioxide

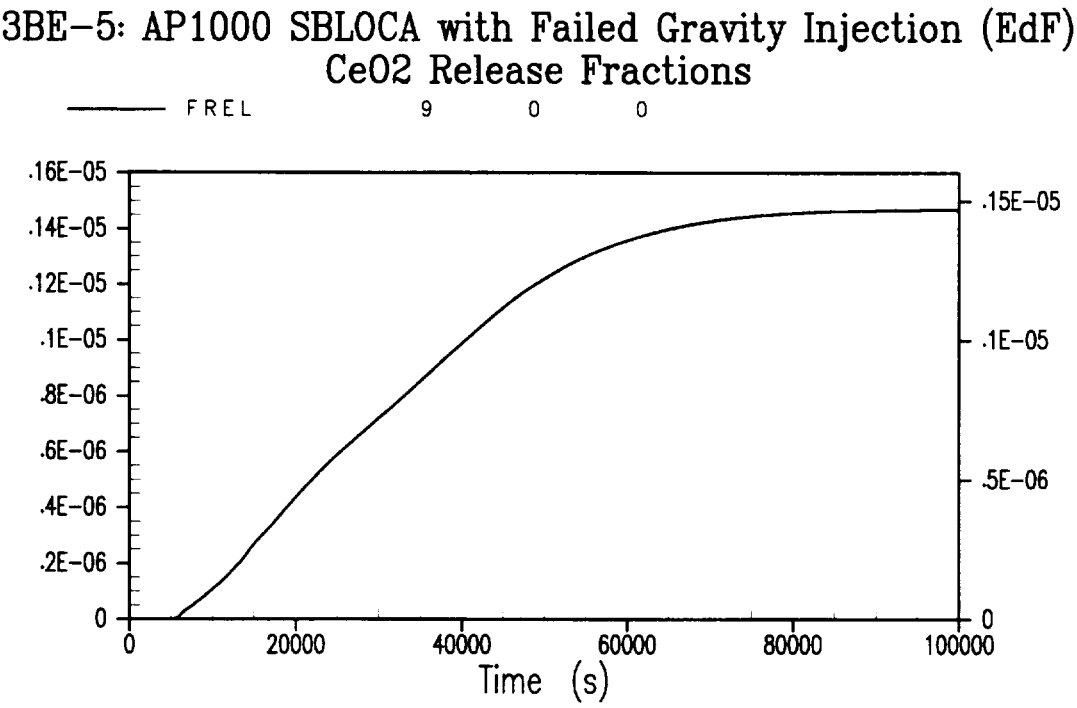


Figure 45-9
Release Category IC, Run 3BE-5: Release Fraction of Cerium Dioxide

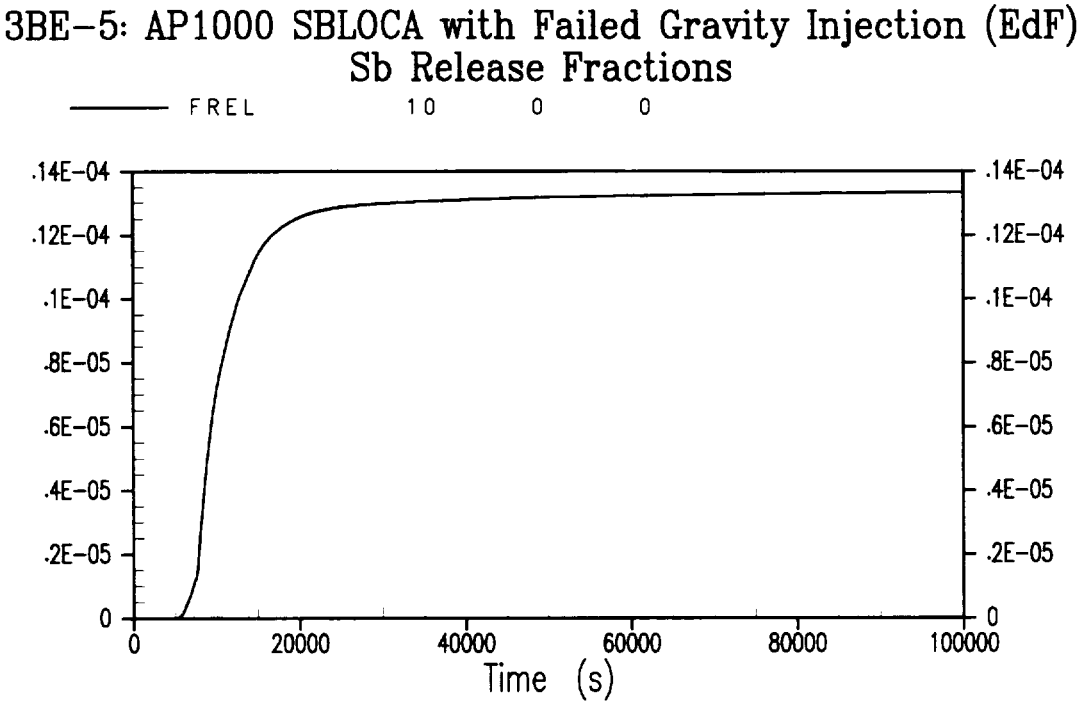


Figure 45-10
Release Category IC, Run 3BE-5: Release Fraction of Tin

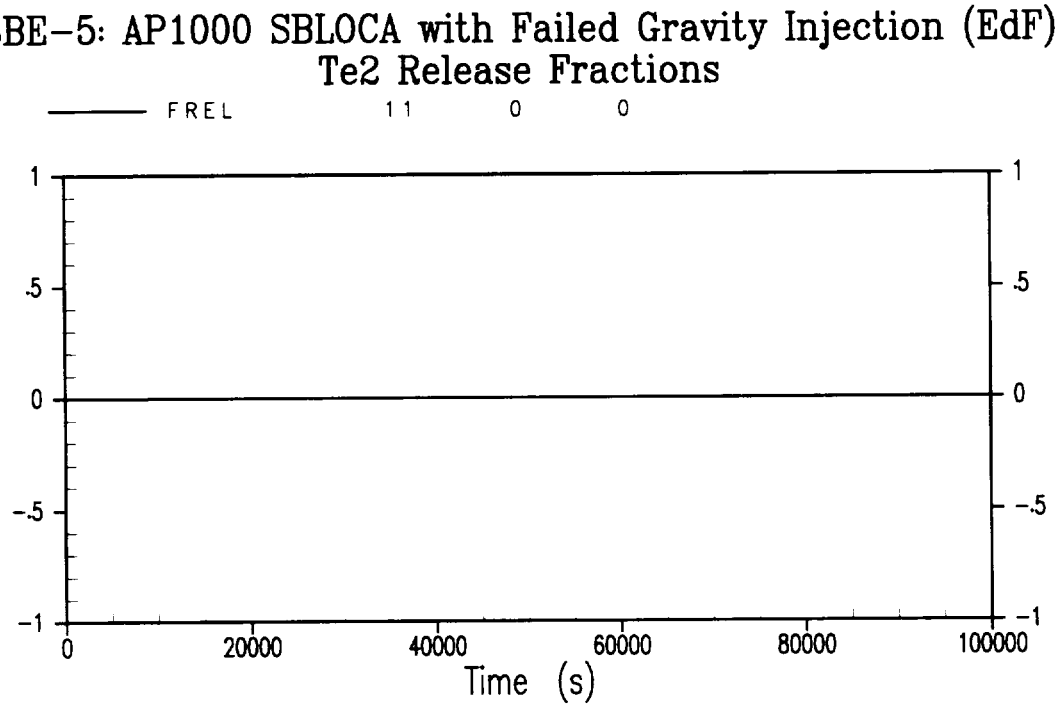


Figure 45-11
Release Category IC, Run 3BE-5: Release Fraction of Tellurium

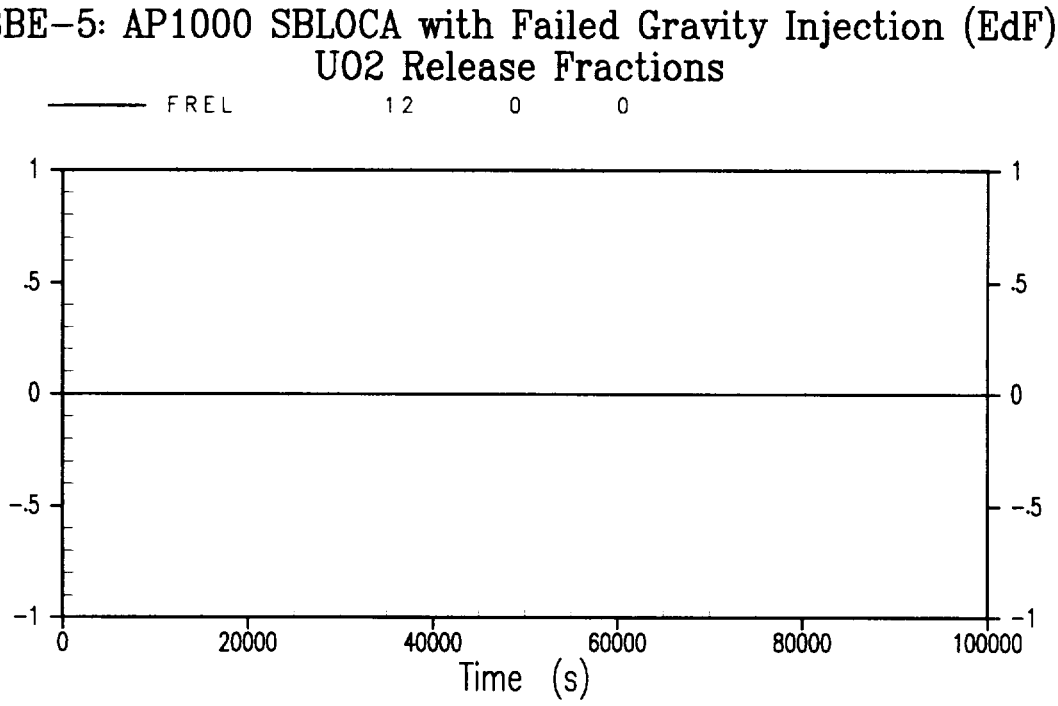


Figure 45-12
Release Category IC, Run 3BE-5: Release Fraction of Uranium Dioxide

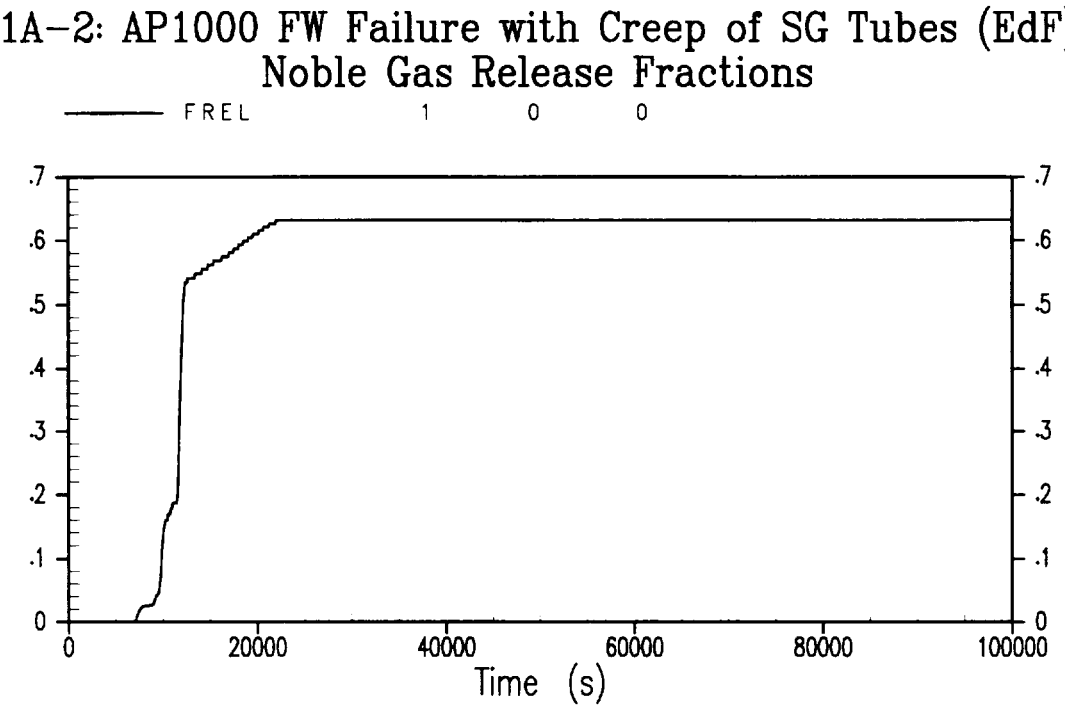


Figure 45-13
Release Category BP, Case 1A-2: Release Fraction of Noble Gases

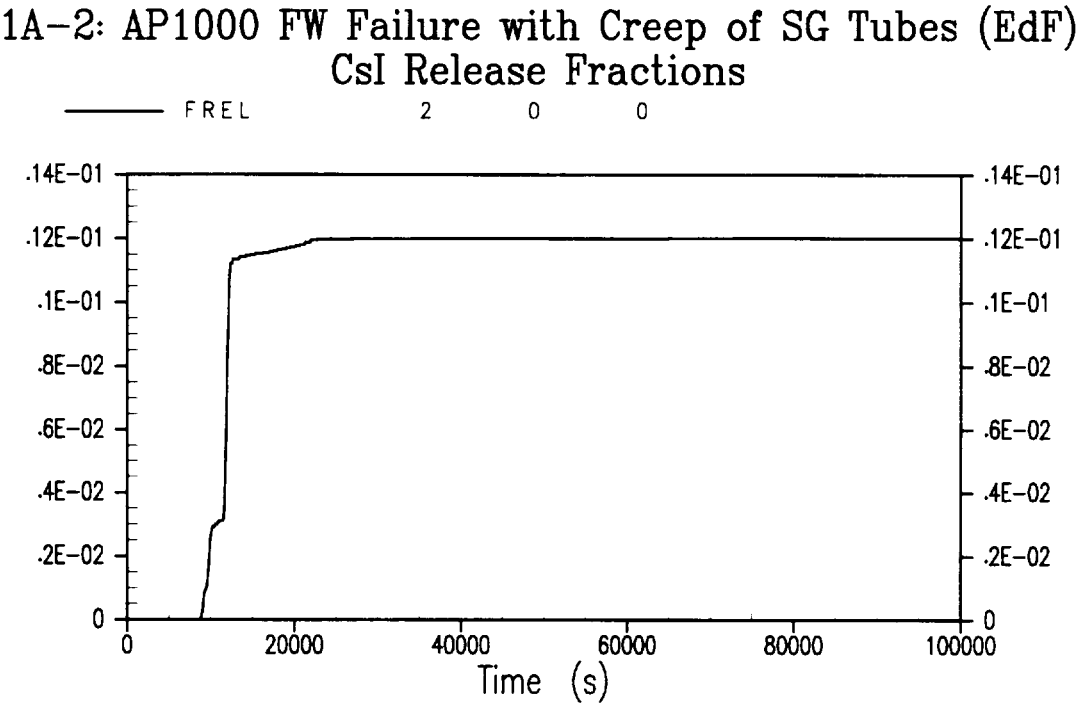


Figure 45-14
Release Category BP, Case 1A-2: Release Fraction of Cesium Iodide

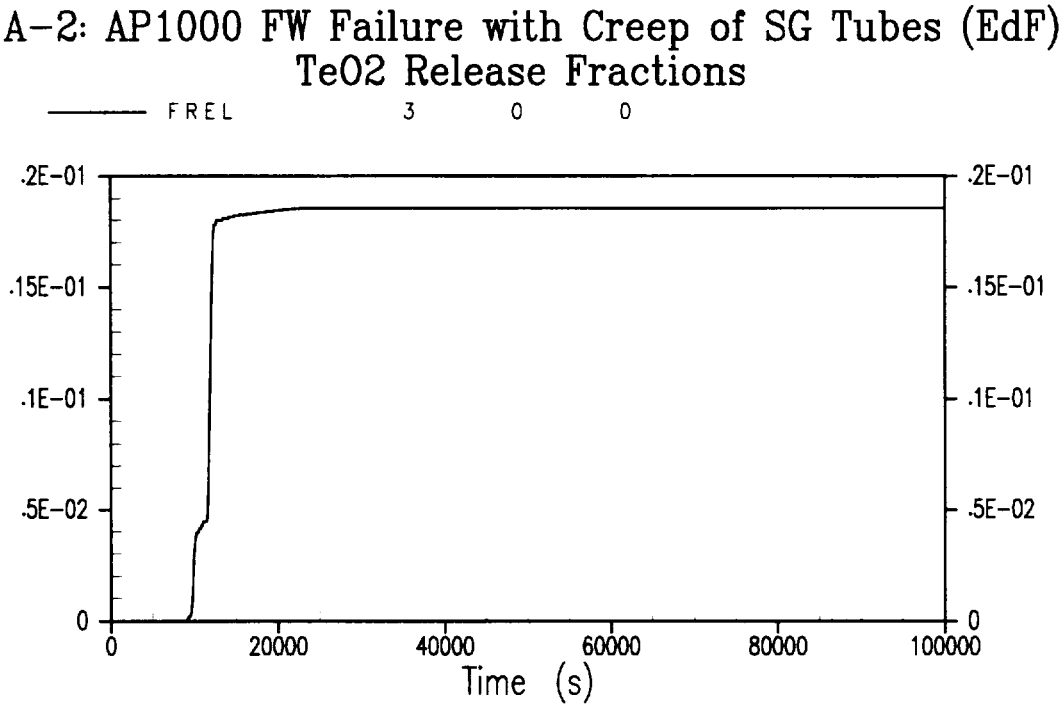


Figure 45-15
Release Category BP, Case 1A-2: Release Fraction of Tellurium Dioxide

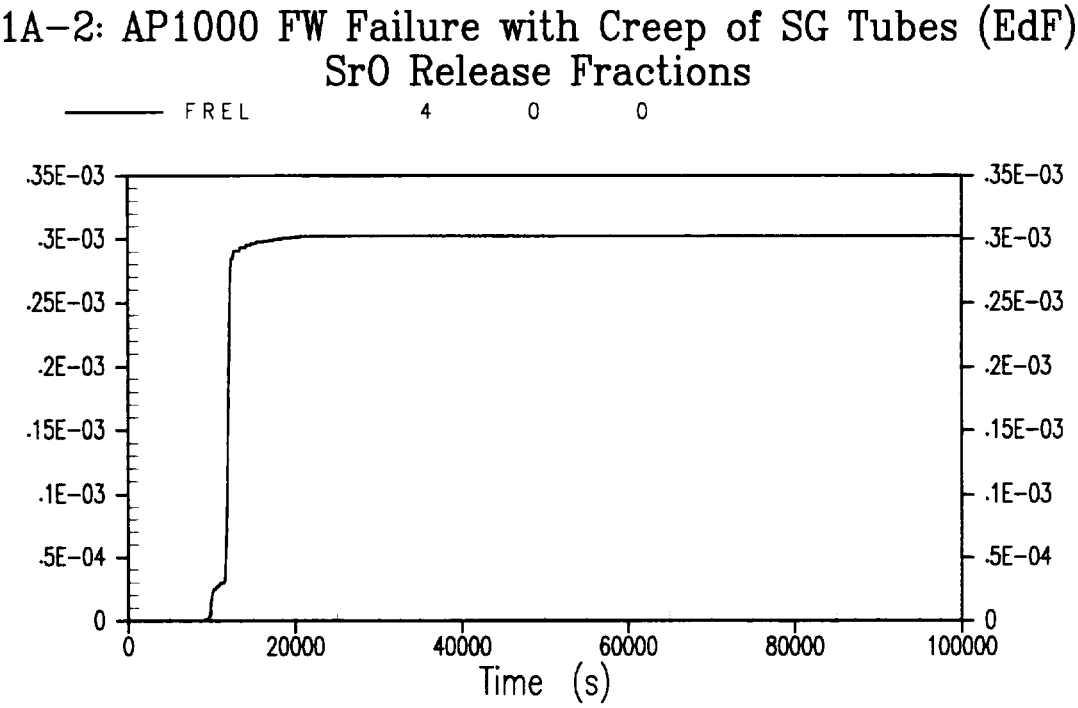


Figure 45-16
Release Category BP, Case 1A-2: Release Fraction of Strontium Oxide

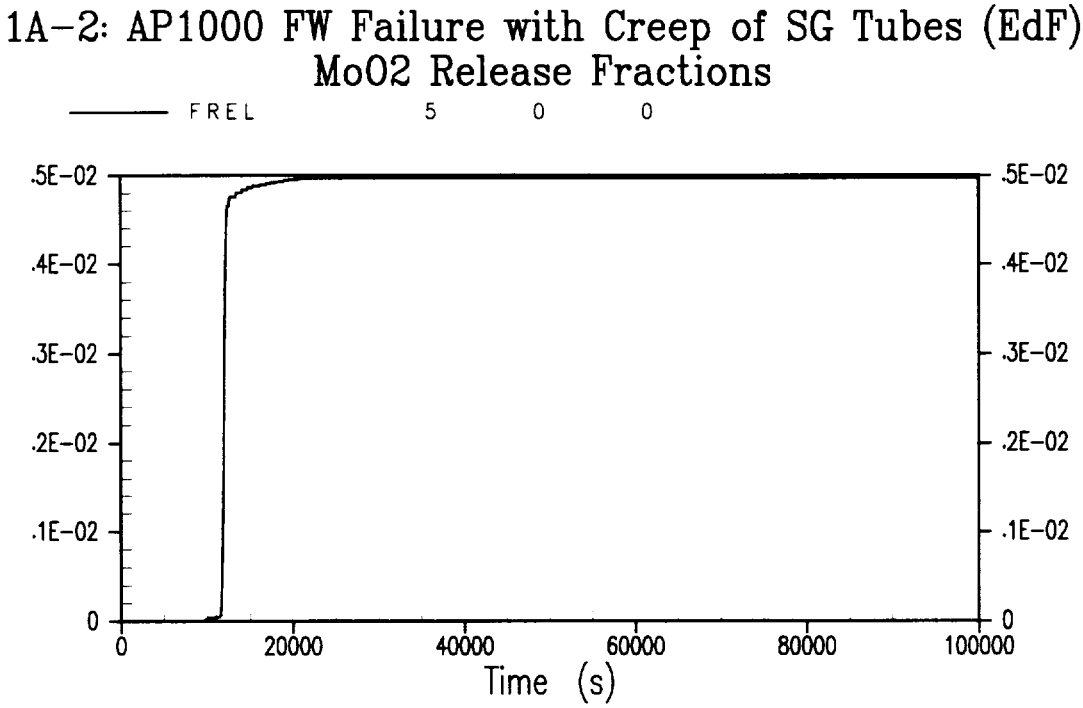


Figure 45-17
Release Category BP, Case 1A-2: Release Fraction of Molybdenum Dioxide

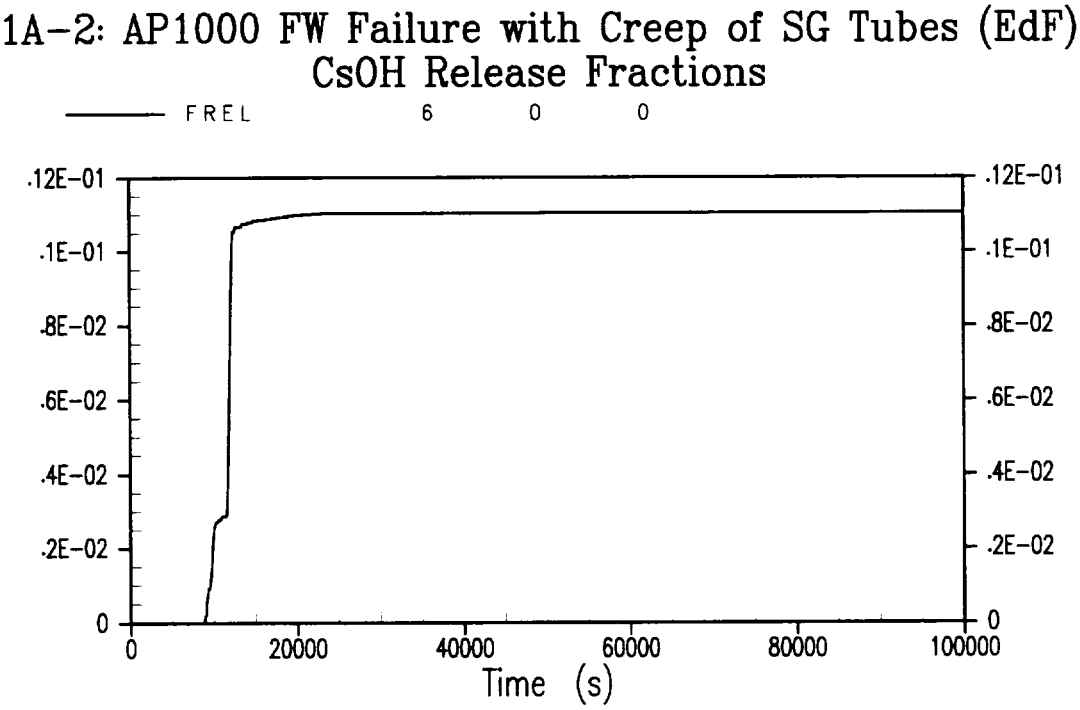


Figure 45-18
Release Category BP, Case 1A-2: Release Fraction of Cesium Hydroxide

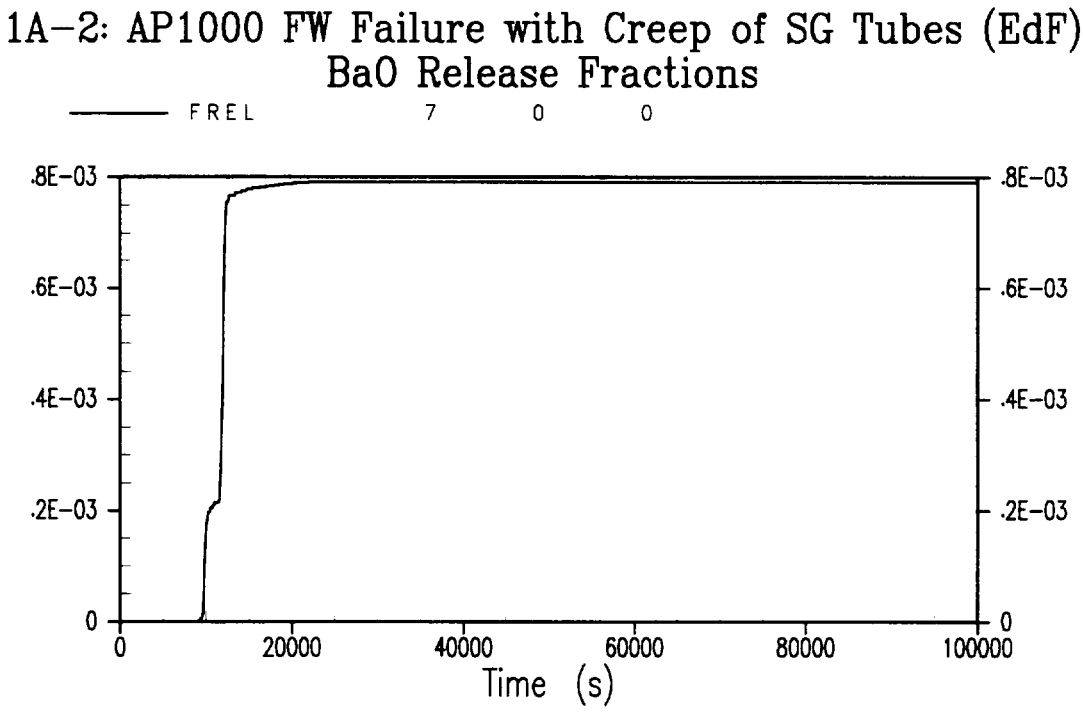


Figure 45-19
Release Category BP, Case 1A-2: Release Fraction of Barium Oxide

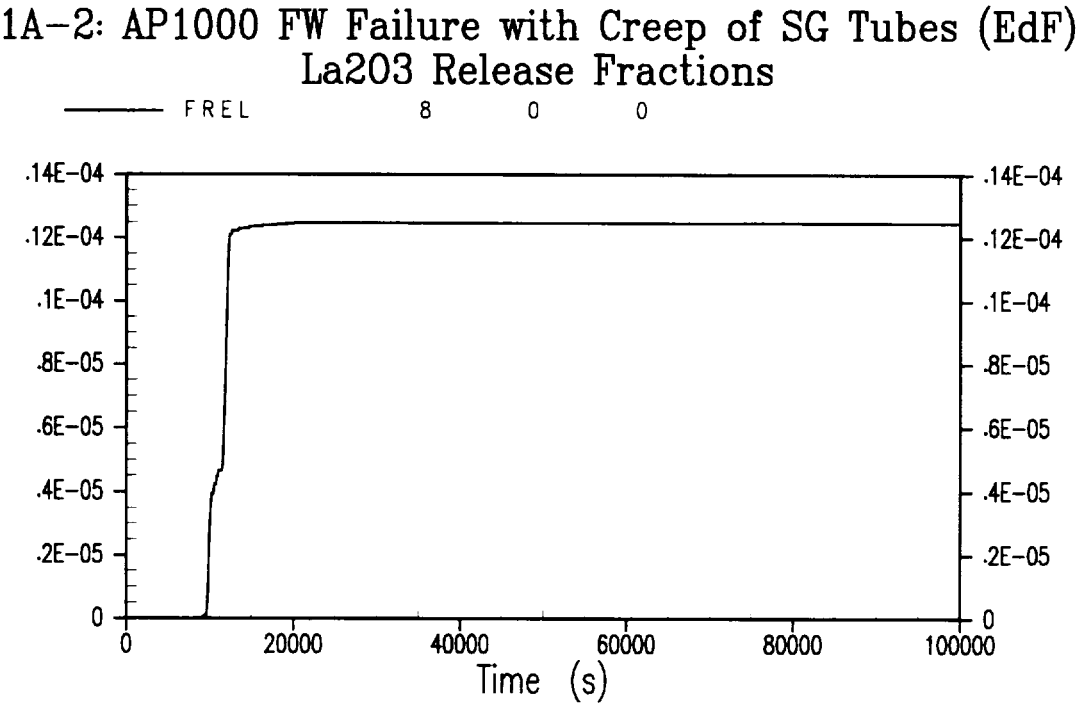


Figure 45-20
Release Category BP, Case 1A-2: Release Fraction of Dilanthanum Trioxide

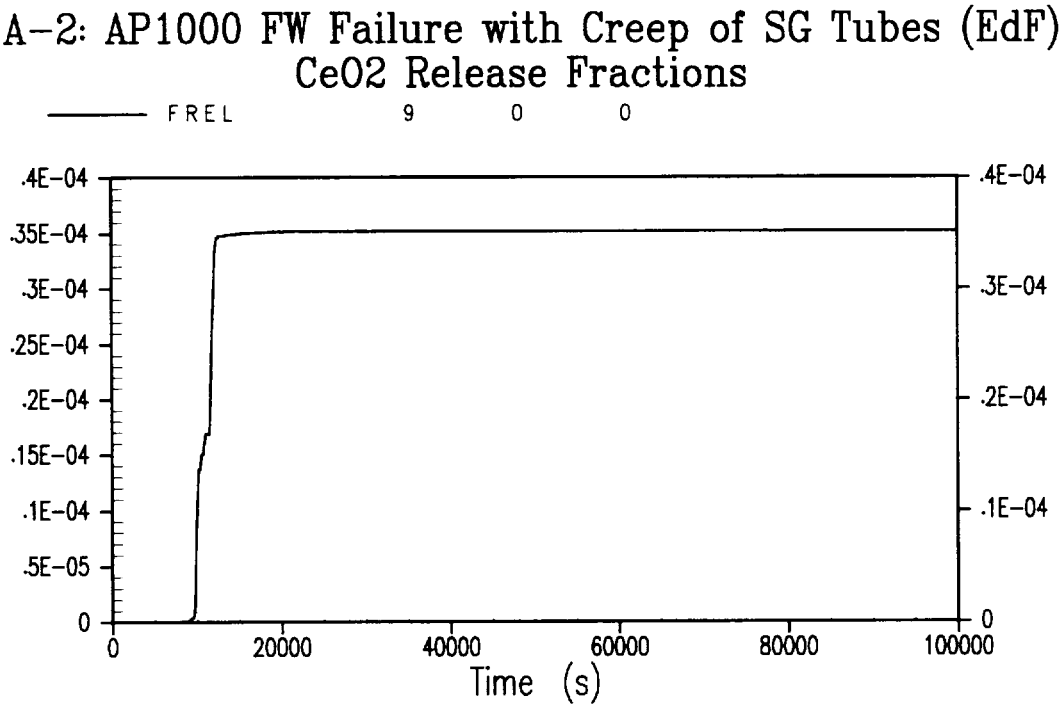


Figure 45-21
Release Category BP, Case 1A-2: Release Fraction of Cerium Dioxide

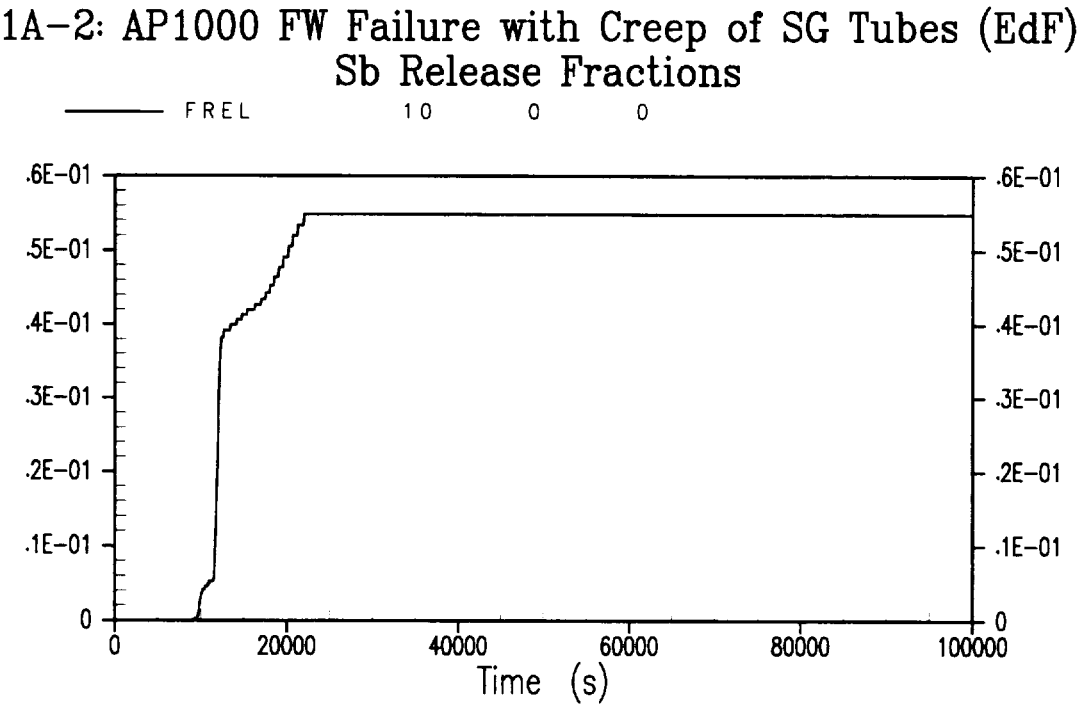


Figure 45-22
Release Category BP, Case 1A-2: Release Fraction of Tin

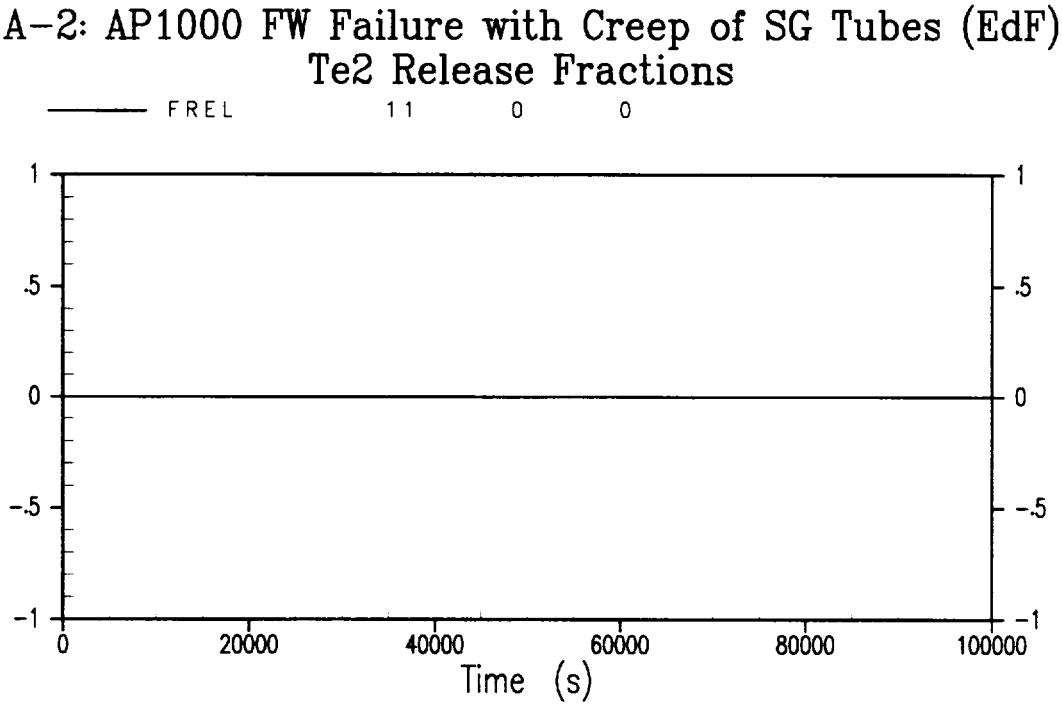


Figure 45-23
Release Category BP, Case 1A-2: Release Fraction of Tellurium

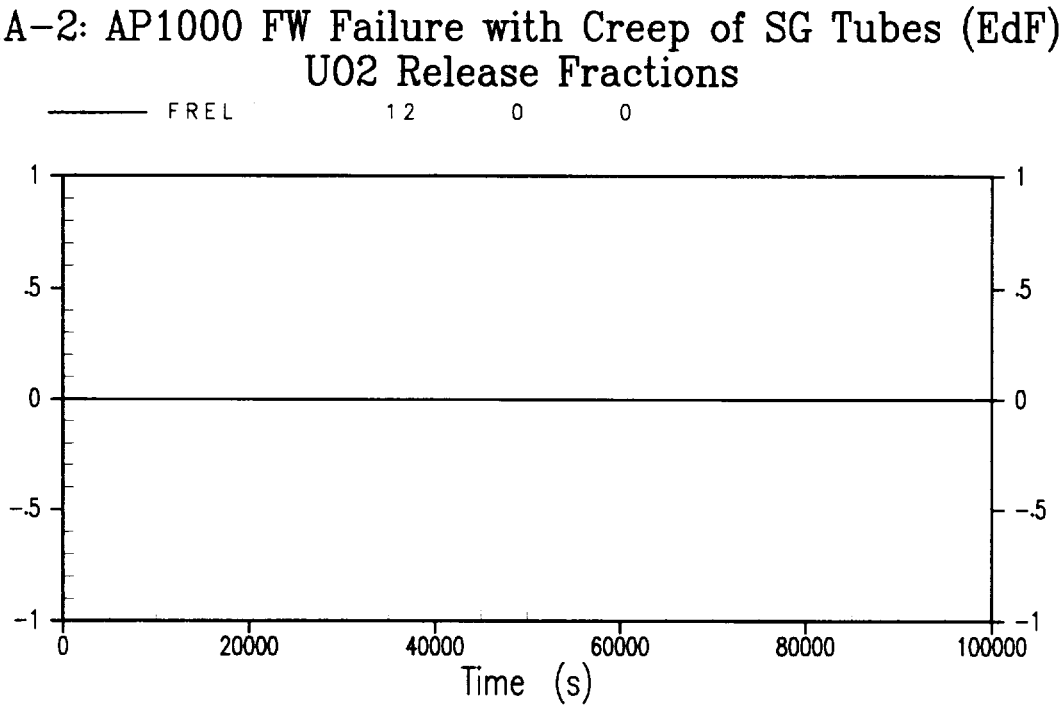


Figure 45-24
Release Category BP, Case 1A-2: Release Fraction of Uranium Dioxide

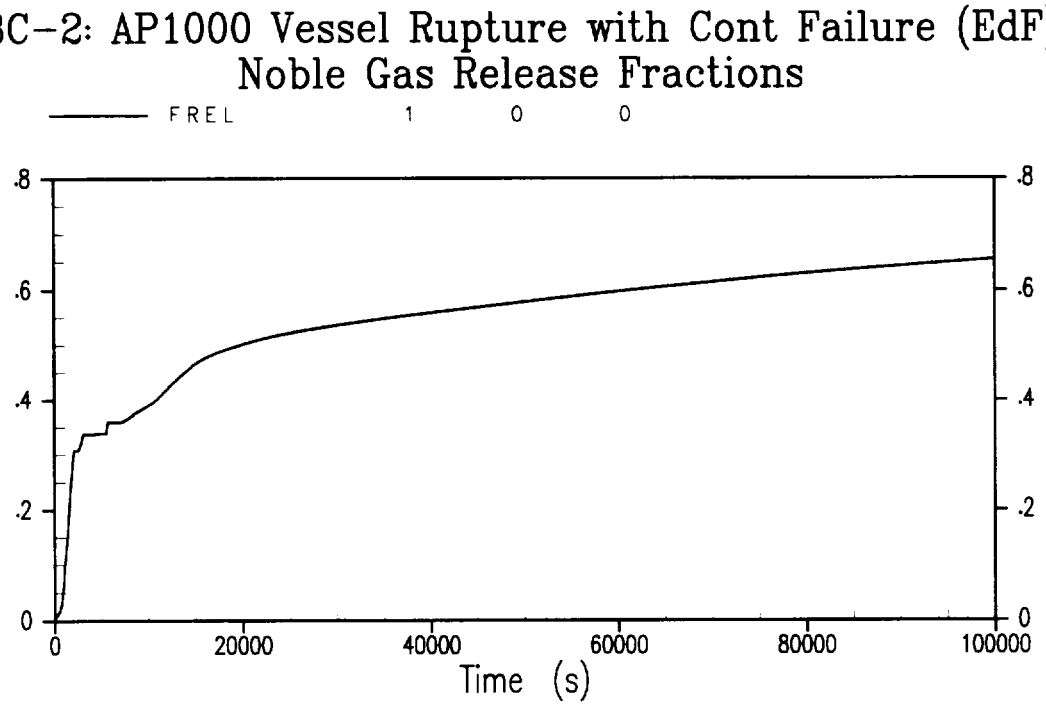


Figure 45-25
Release Category CI, Case 3C-2: Release Fraction of Noble Gases

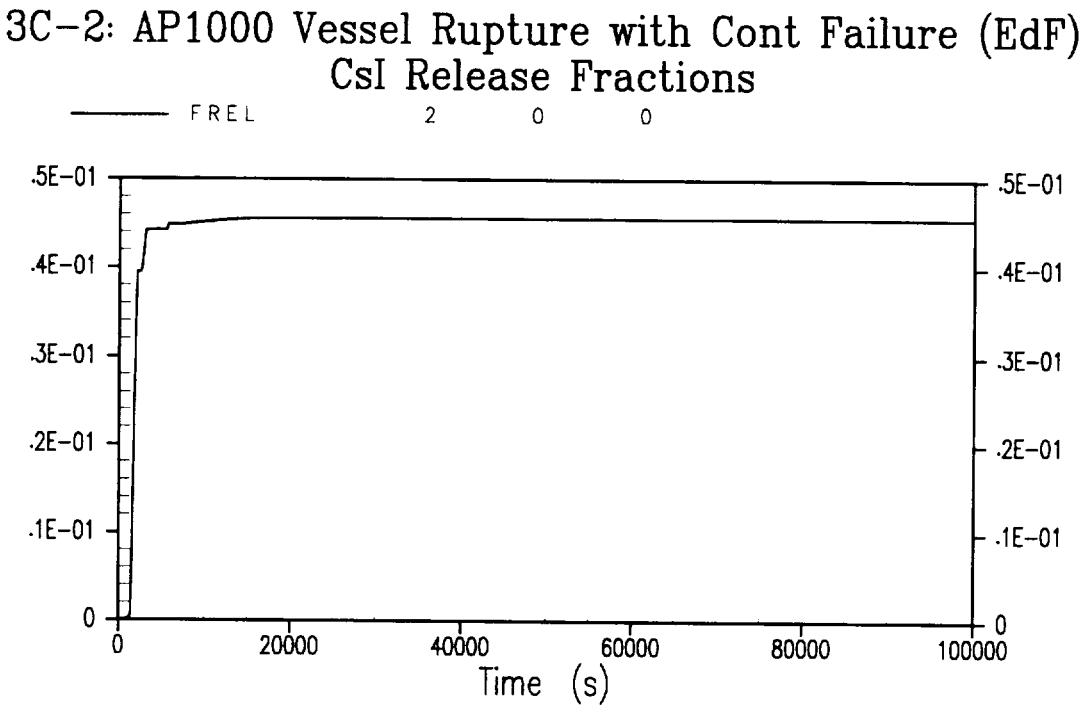


Figure 45-26
Release Category CI, Case 3C-2: Release Fraction of Cesium Iodide

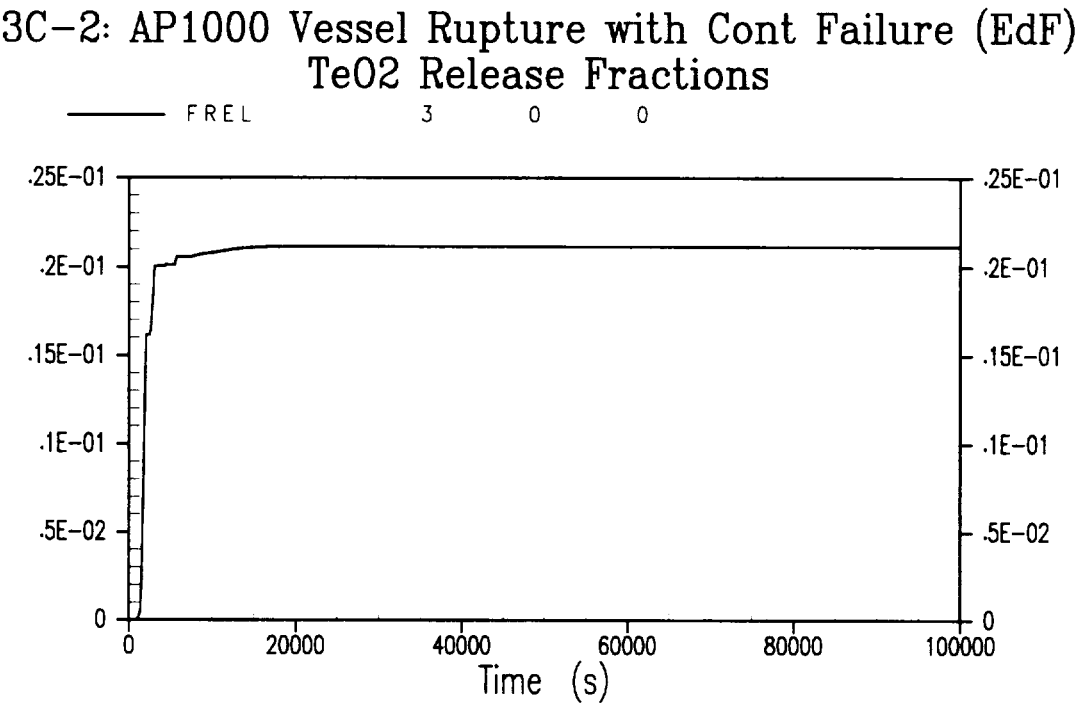


Figure 45-27
Release Category CI, Case 3C-2: Release Fraction of Tellurium Dioxide

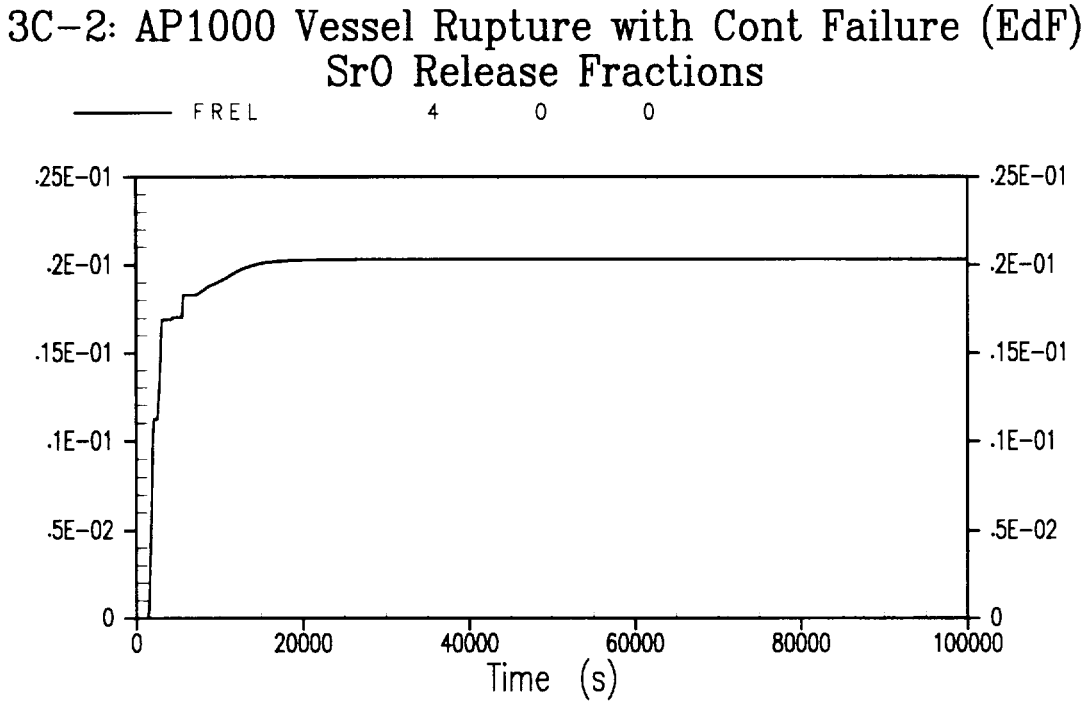


Figure 45-28
Release Category CI, Case 3C-2: Release Fraction of Strontium Oxide

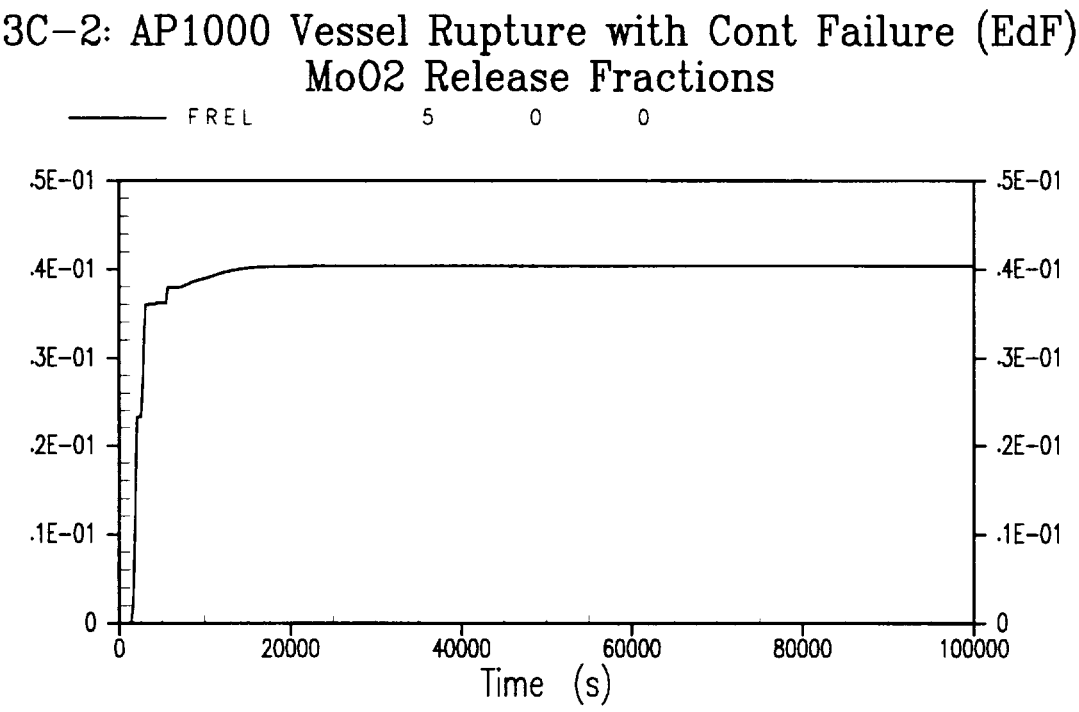


Figure 45-29
Release Category CI, Case 3C-2: Release Fraction of Molybdenum Dioxide

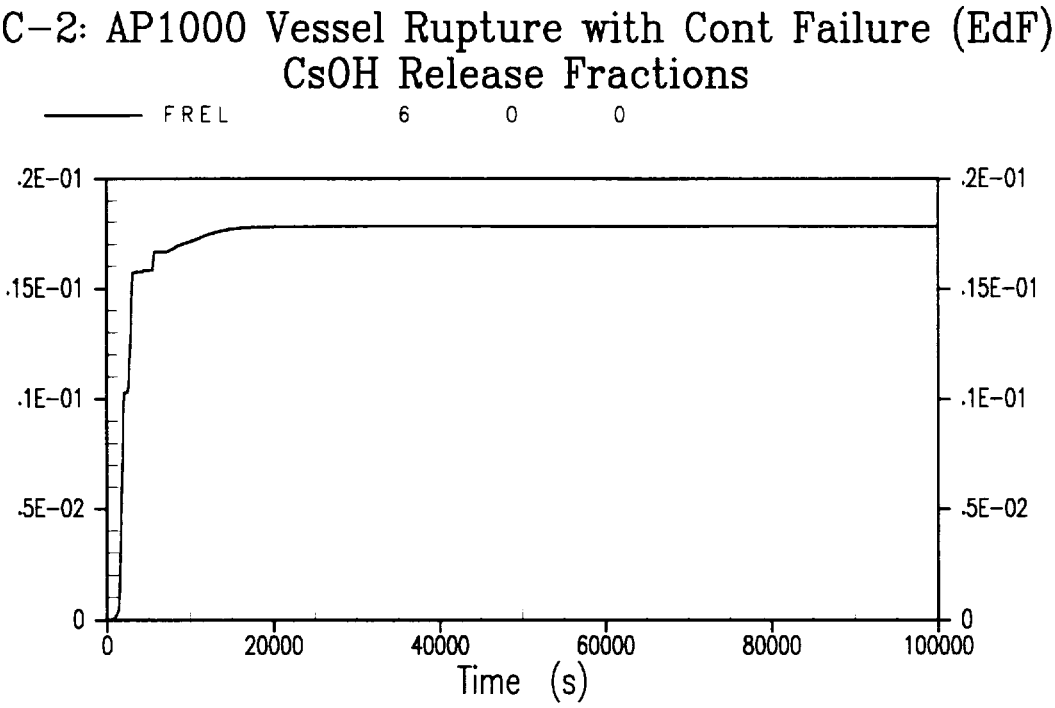


Figure 45-30
Release Category CI, Case 3C-2: Release Fraction of Cesium Hydroxide

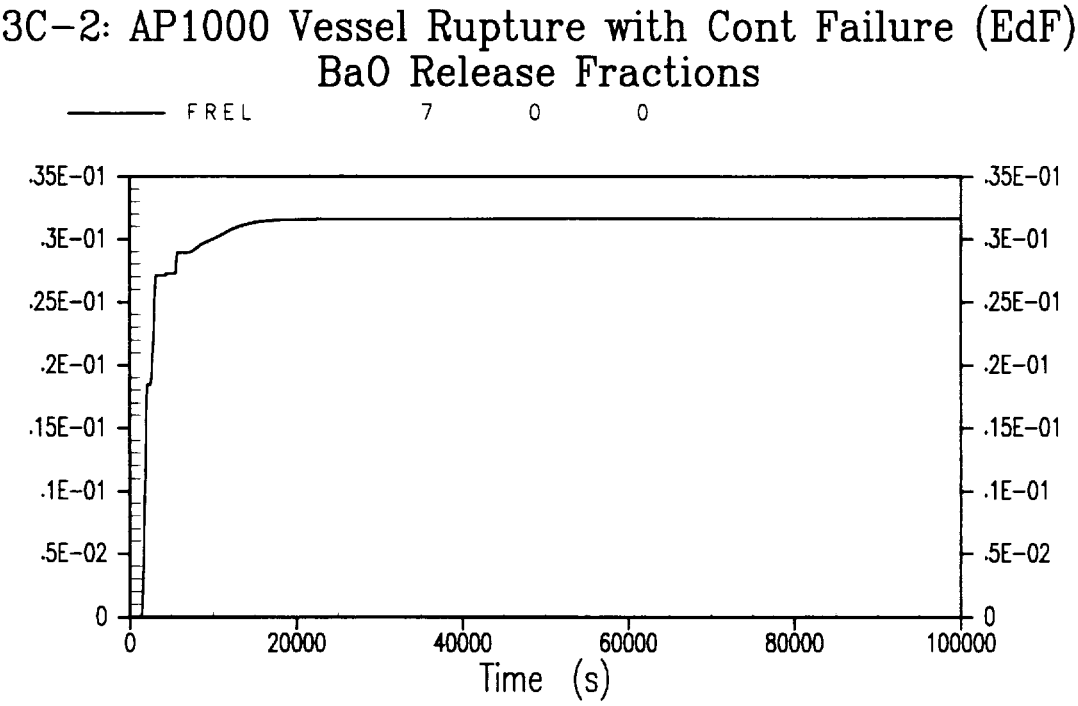


Figure 45-31
Release Category CI, Case 3C-2: Release Fraction of Barium Oxide

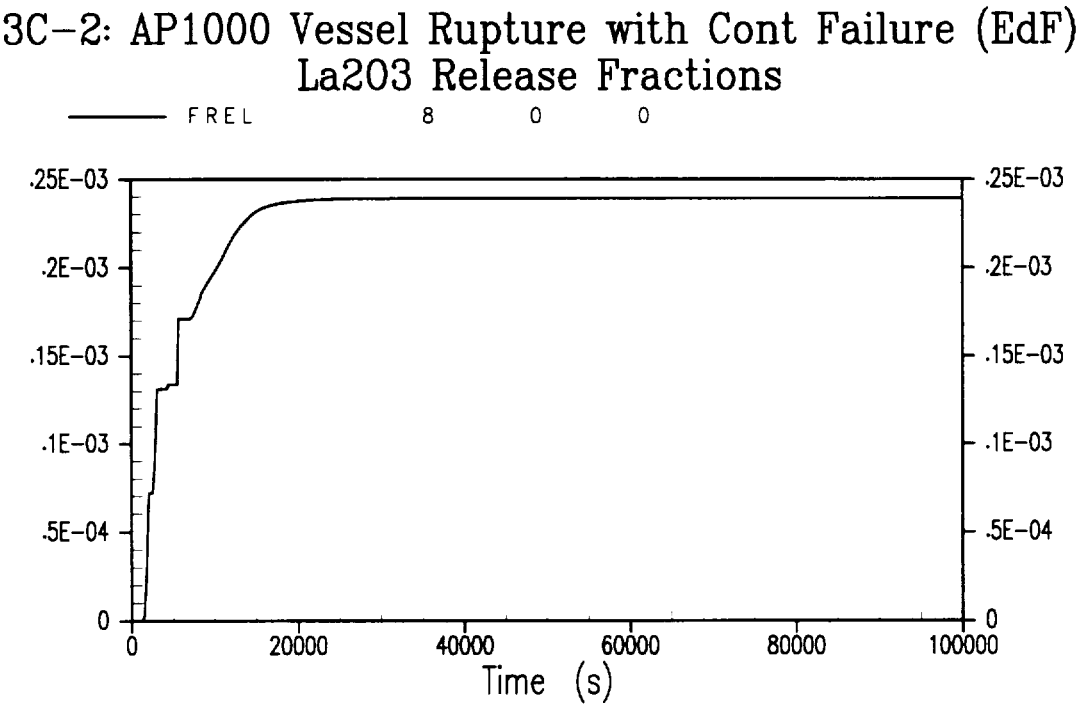


Figure 45-32
Release Category CI, Case 3C-2: Release Fraction of Dilanthanum Trioxide

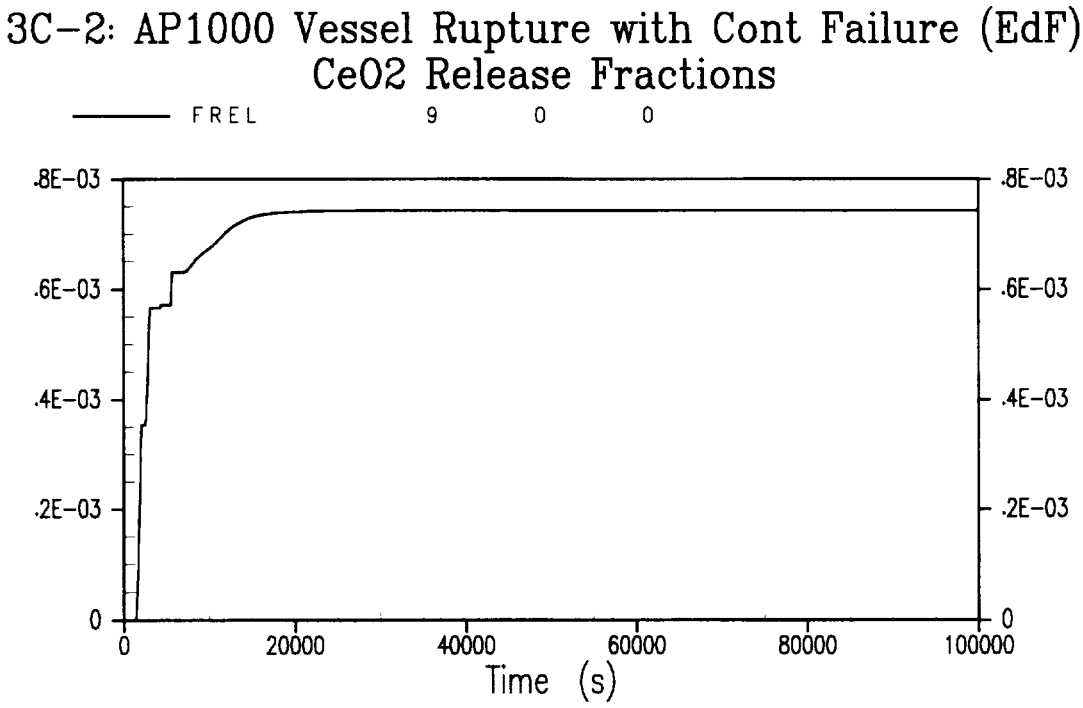


Figure 45-33
Release Category CI, Case 3C-2: Release Fraction of Cerium Dioxide

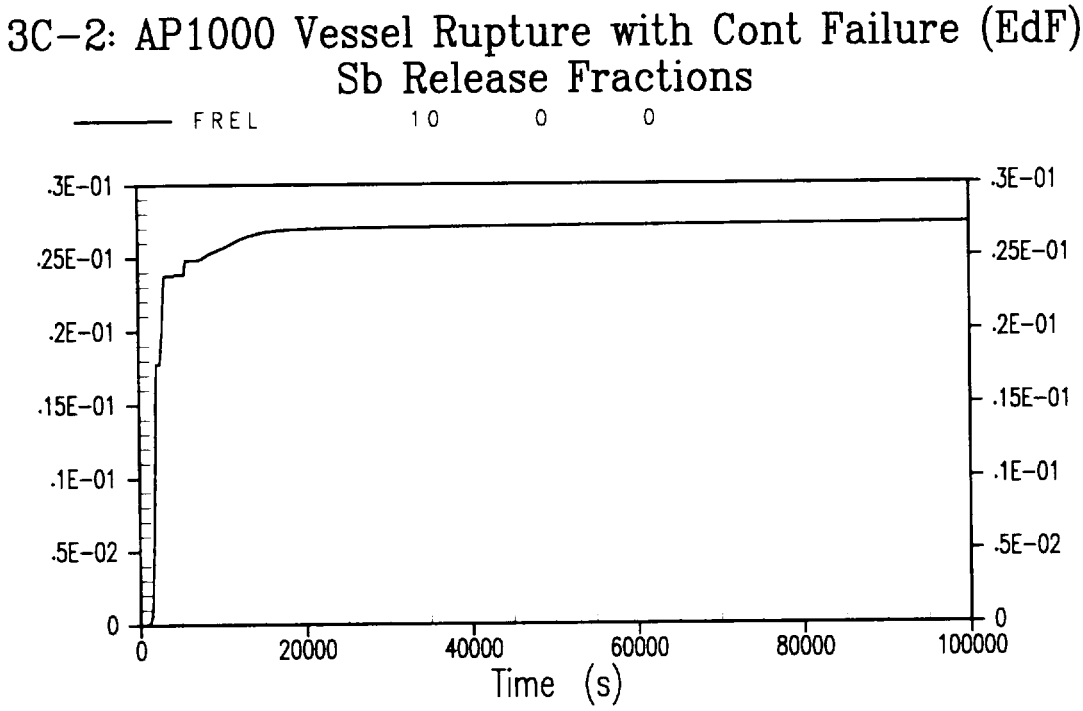


Figure 45-34
Release Category CI, Case 3C-2: Release Fraction of Tin

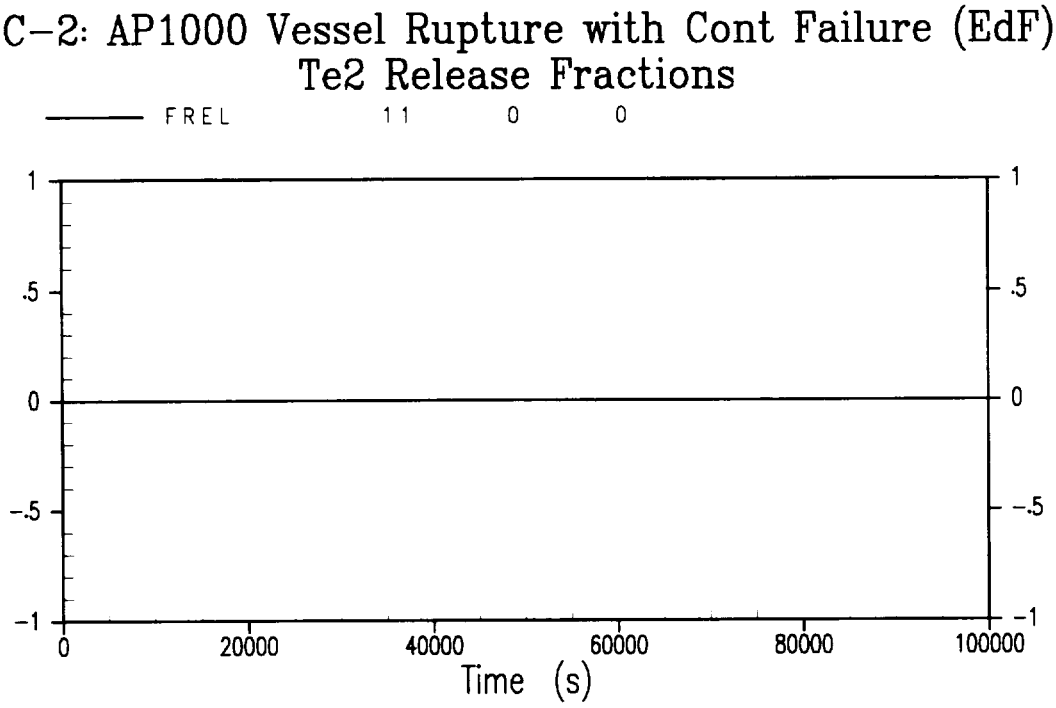


Figure 45-35
Release Category CI, Case 3C-2: Release Fraction of Tellurium

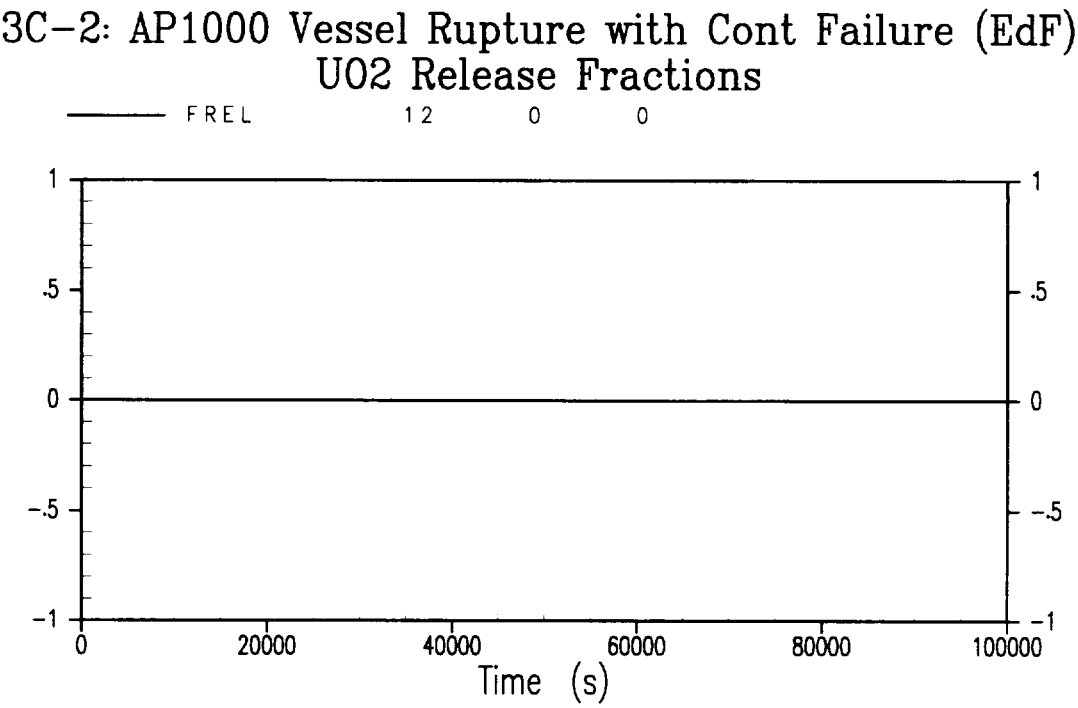


Figure 45-36
Release Category CI, Case 3C-2: Release Fraction of Uranium Dioxide

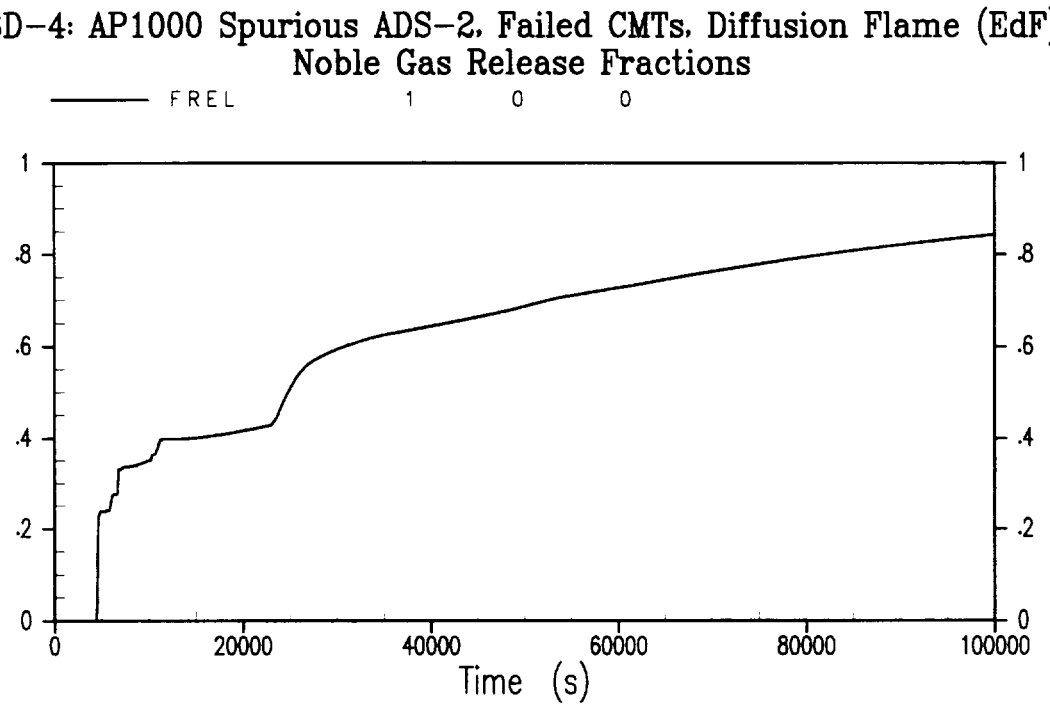


Figure 45-37
Release Category CFE, Case 3D-4: Release Fraction of Noble Gases

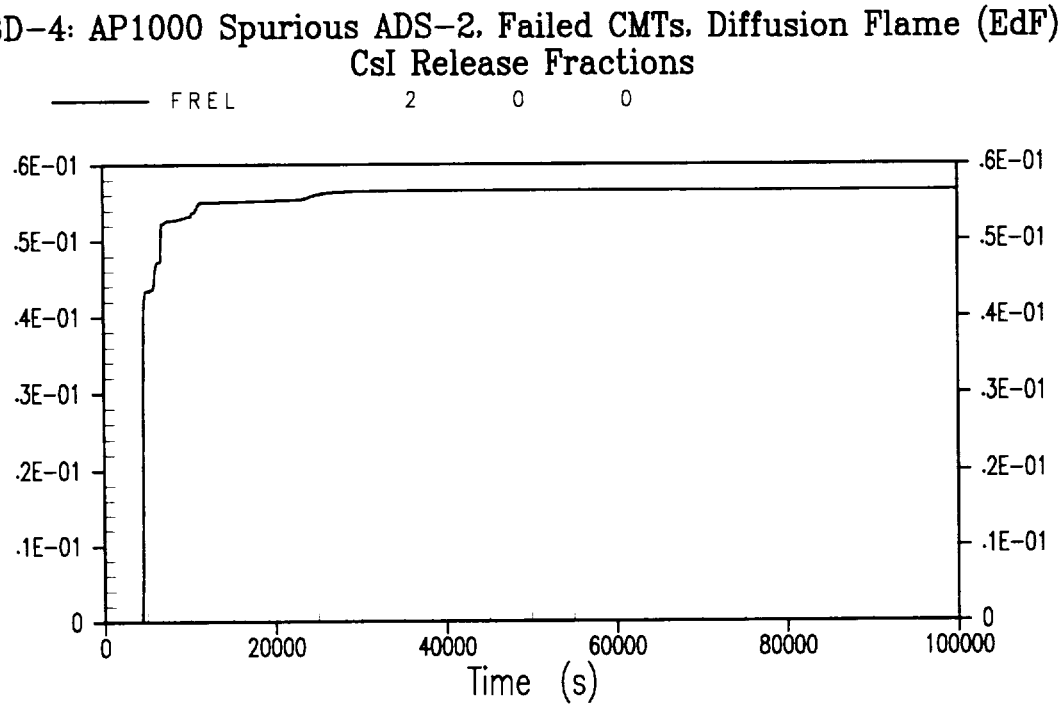


Figure 45-38
Release Category CFE, Case 3D-4: Release Fraction of Cesium Iodide

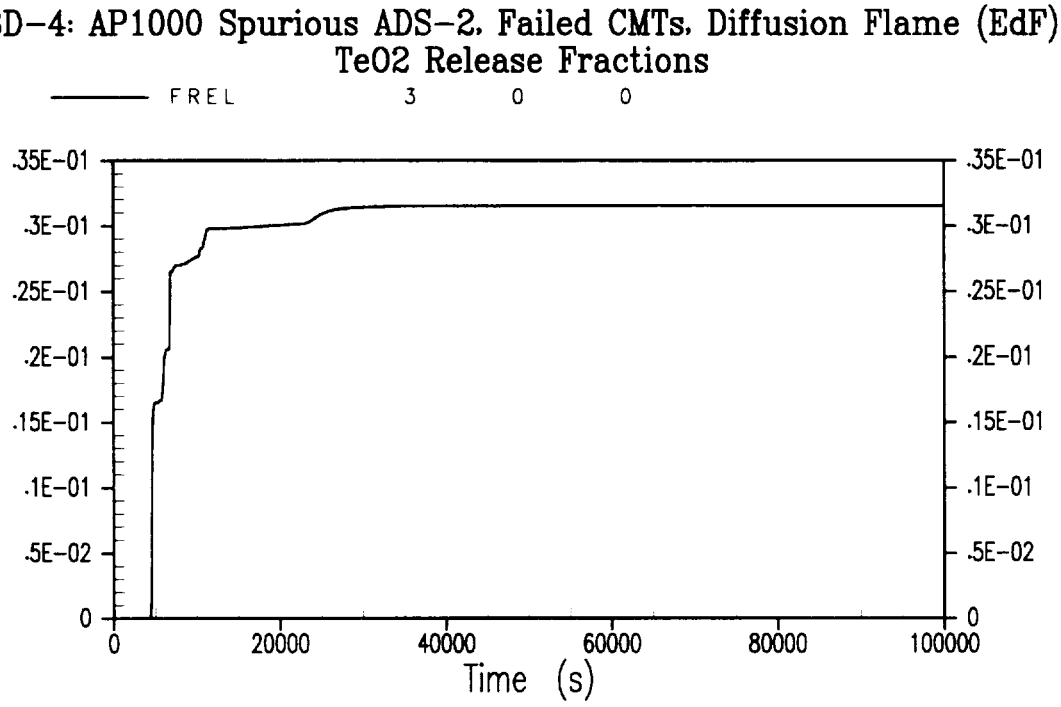


Figure 45-39
Release Category CFE, Case 3D-4: Release Fraction of Tellurium Dioxide

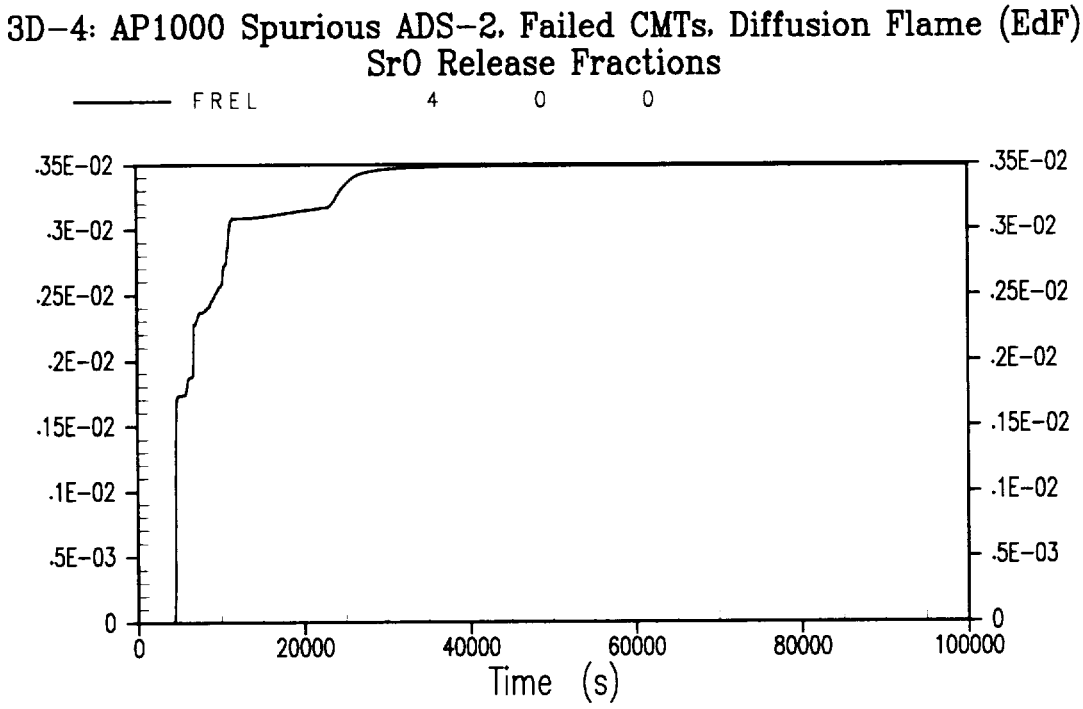


Figure 45-40
Release Category CFE, Case 3D-4: Release Fraction of Strontium Oxide

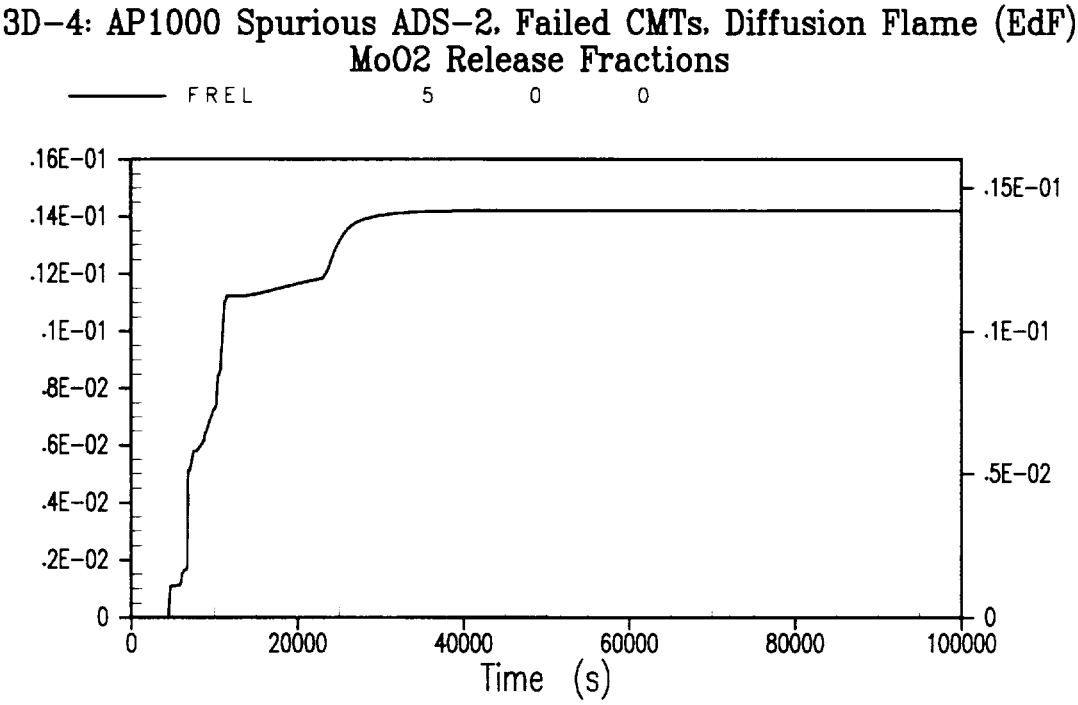


Figure 45-41
Release Category CFE, Case 3D-4: Release Fraction of Molybdenum Dioxide

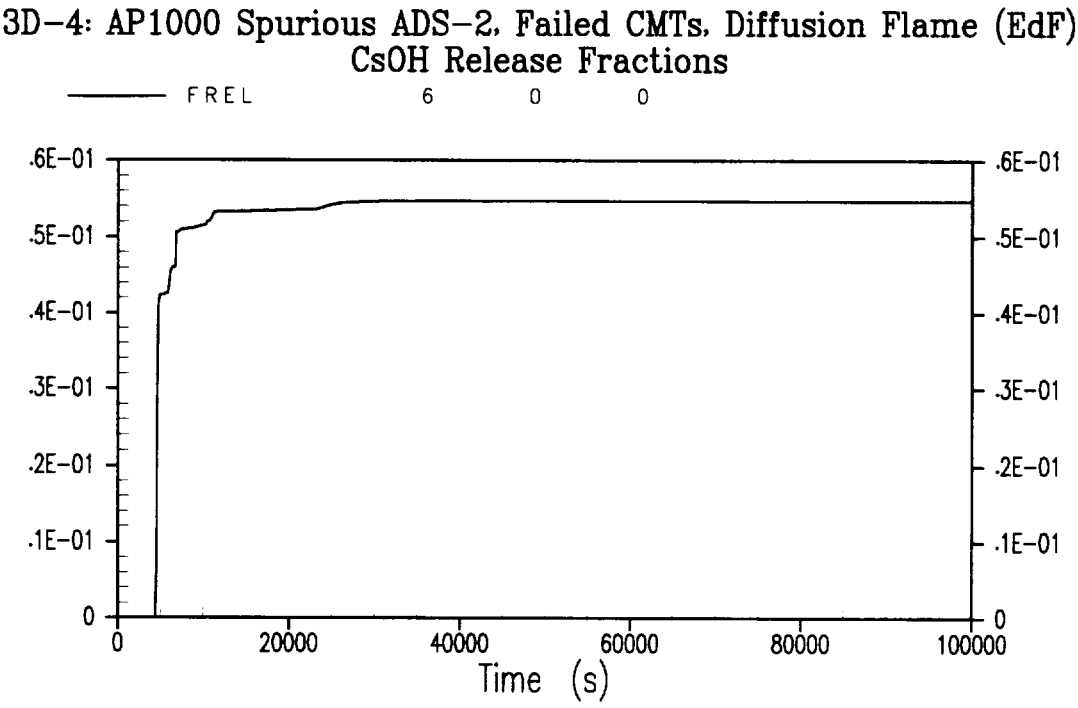


Figure 45-42
Release Category CFE, Case 3D-4: Release Fraction of Cesium Hydroxide

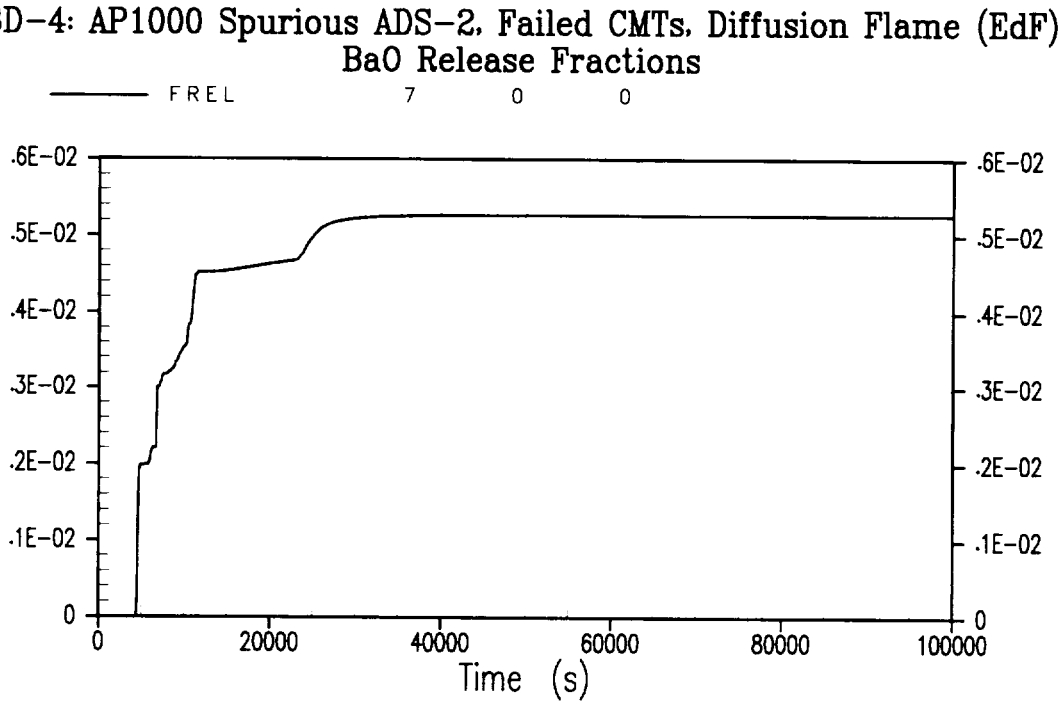


Figure 45-43
Release Category CFE, Case 3D-4: Release Fraction of Barium Oxide

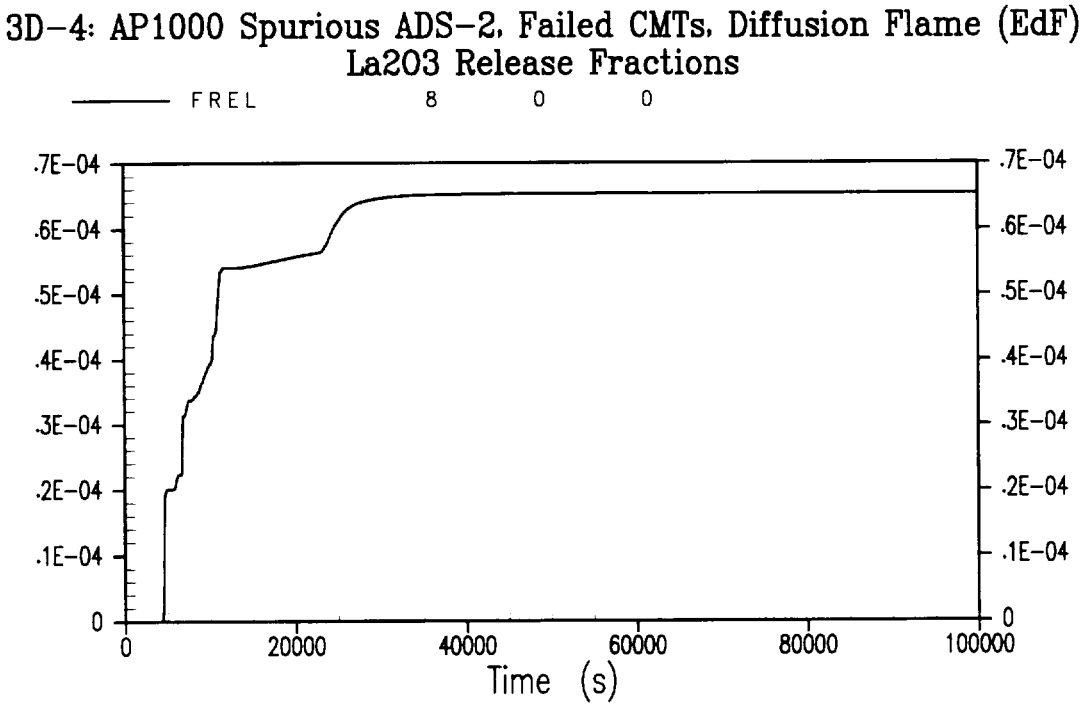


Figure 45-44
Release Category CFE, Case 3D-4: Release Fraction of Dilanthanum Trioxide

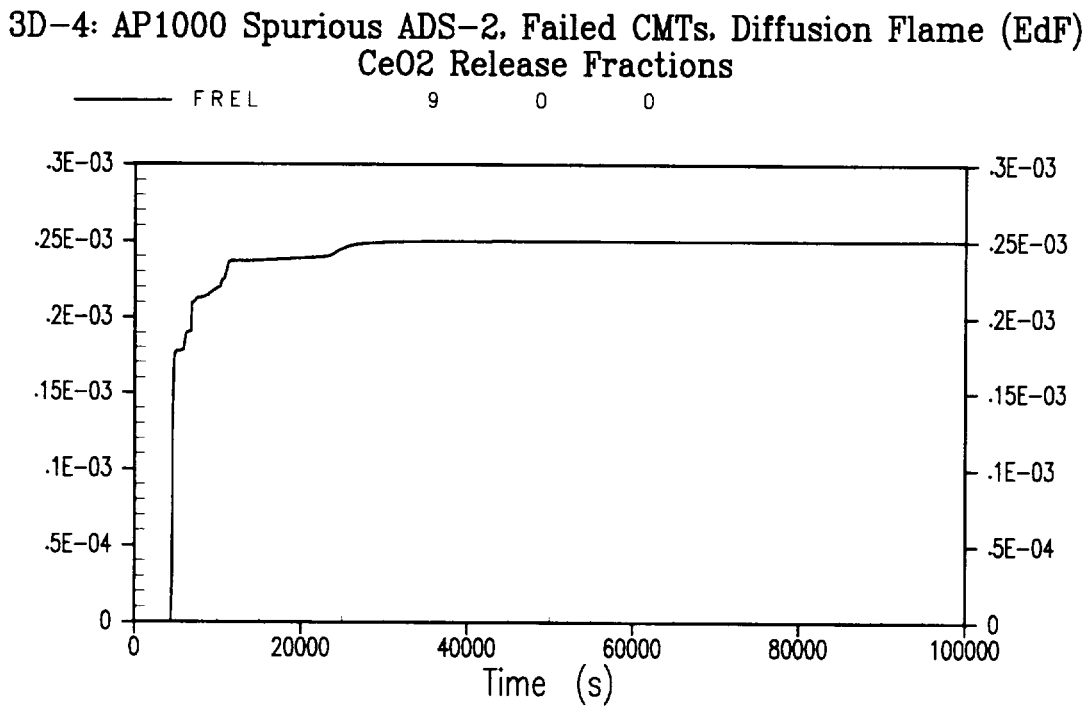


Figure 45-45
Release Category CFE, Case 3D-4: Release Fraction of Cerium Dioxide

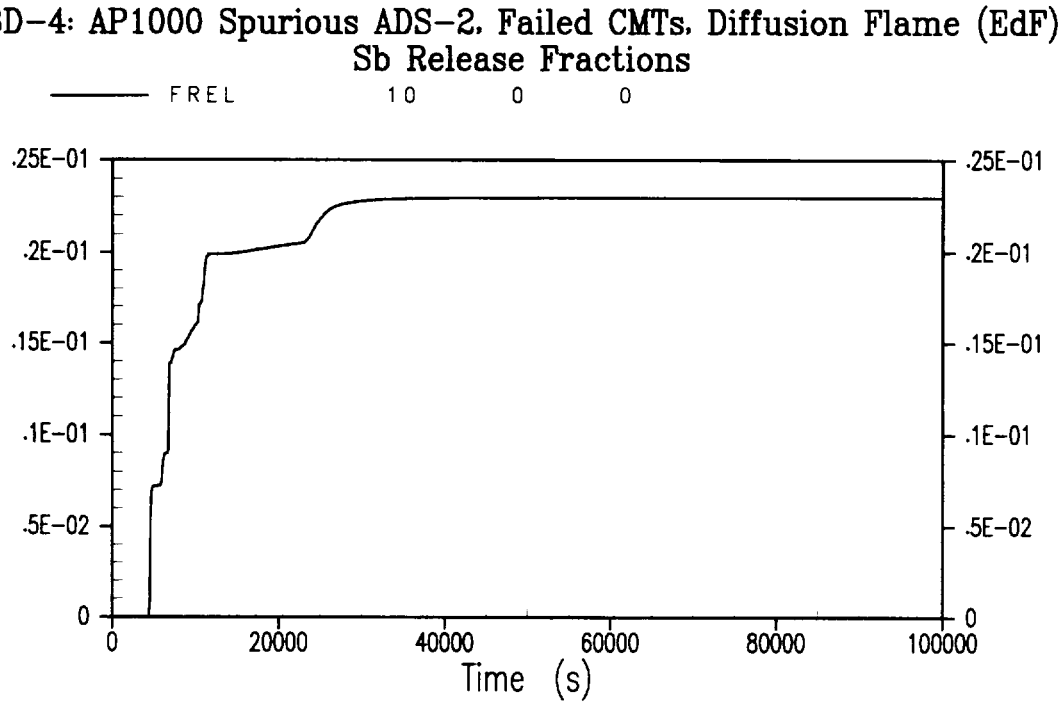


Figure 45-46
Release Category CFE, Case 3D-4: Release Fraction of Tin

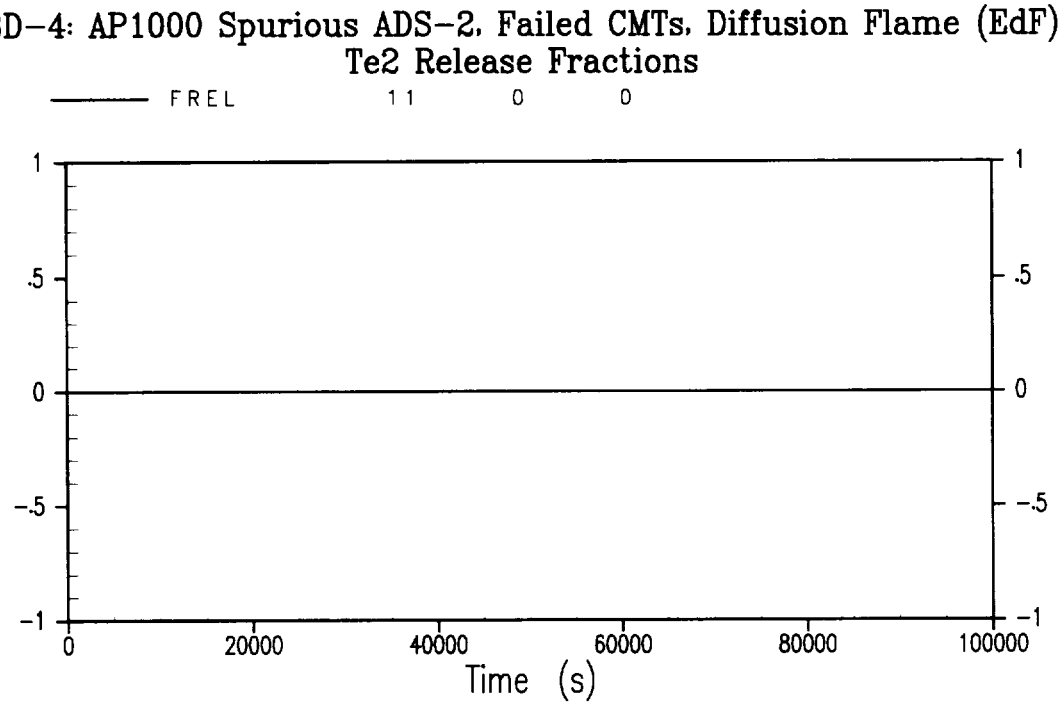


Figure 45-47
Release Category CFE, Case 3D-4: Release Fraction of Tellurium

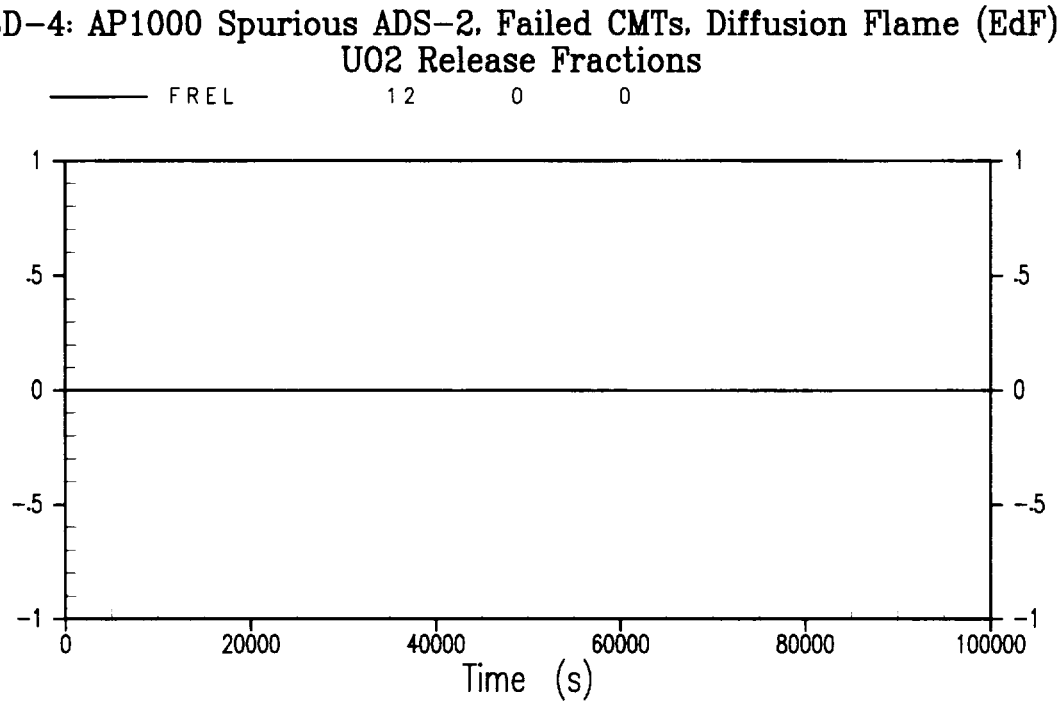


Figure 45-48
Release Category CFE, Case 3D-4: Release Fraction of Uranium Dioxide

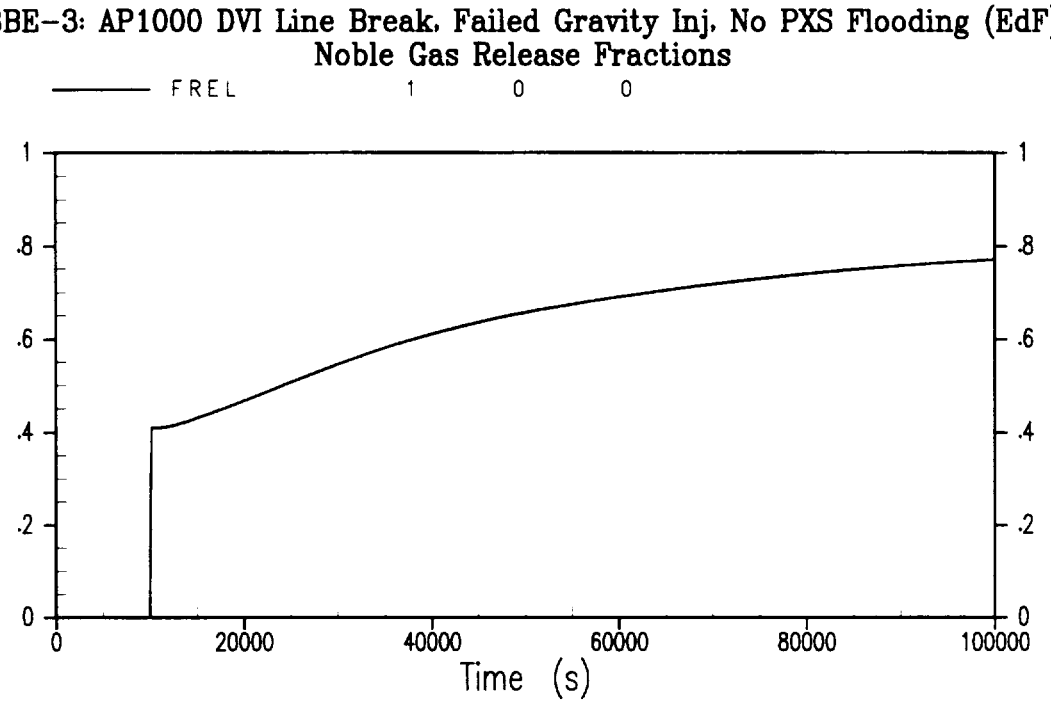


Figure 45-49
Release Category CFI, Case 3BE-3: Release Fraction of Noble Gases

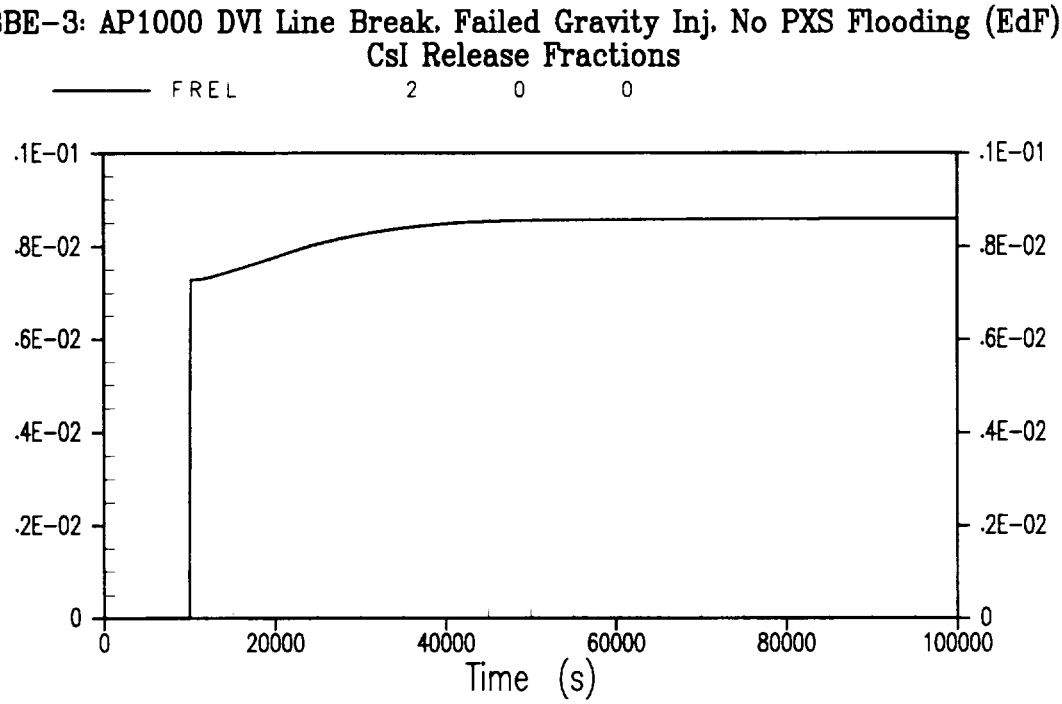


Figure 45-50
Release Category CFI, Case 3BE-3: Release Fraction of Cesium Iodide

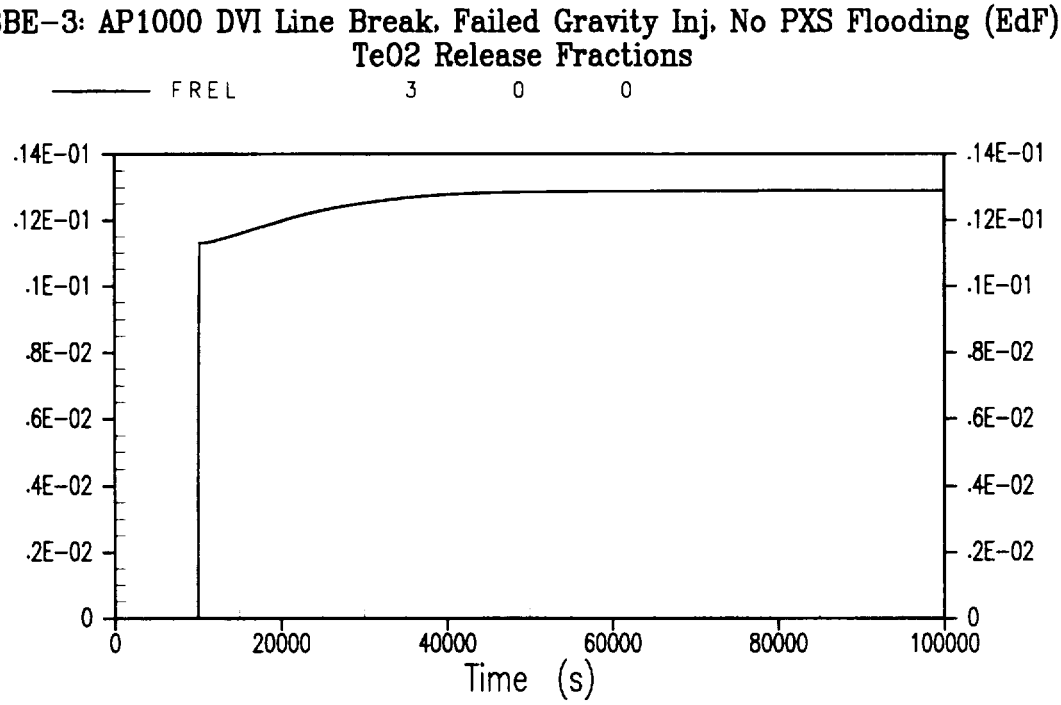


Figure 45-51
Release Category CFI, Case 3BE-3: Release Fraction of Tellurium Dioxide

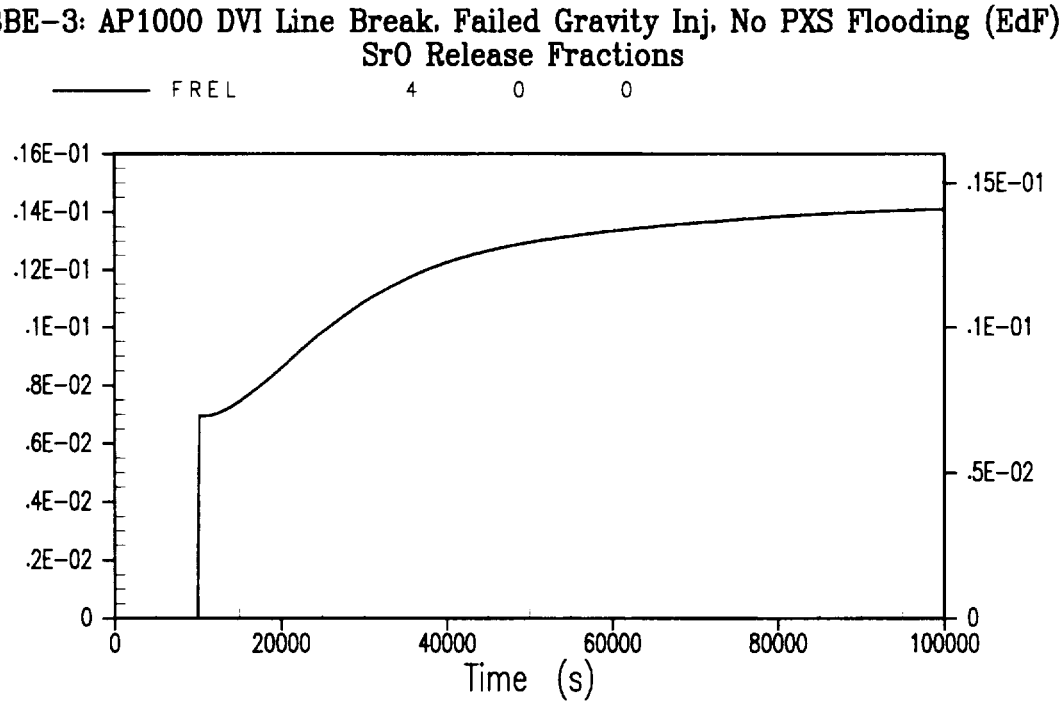


Figure 45-52
Release Category CFI, Case 3BE-3: Release Fraction of Strontium Oxide

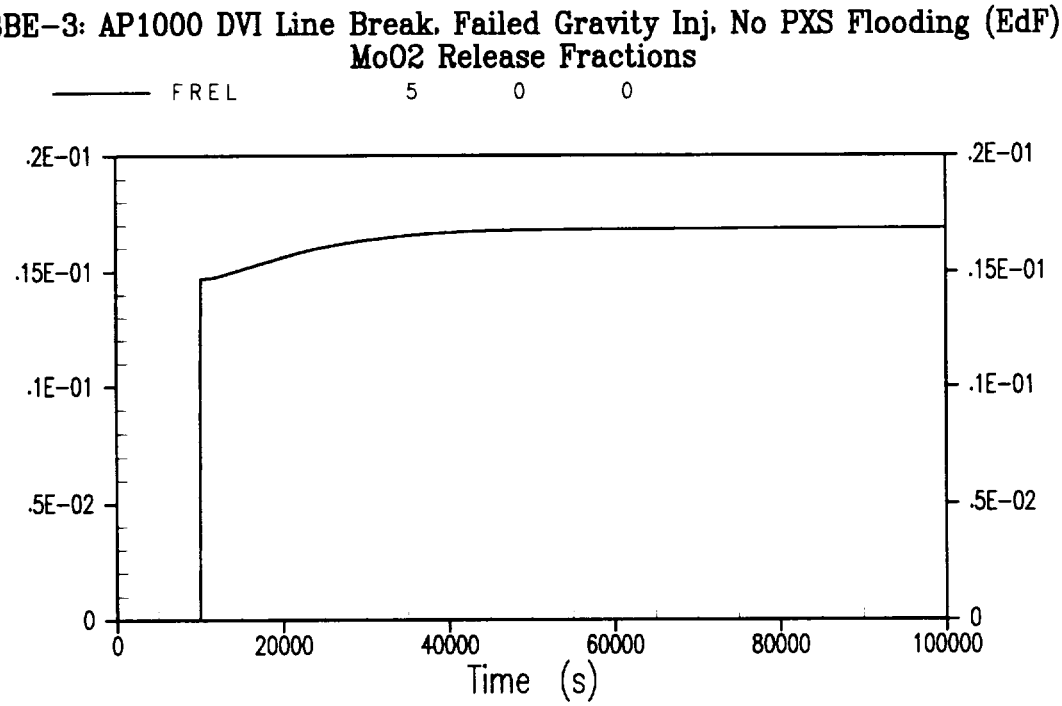


Figure 45-53
Release Category CFI, Case 3BE-3: Release Fraction of Molybdenum Dioxide

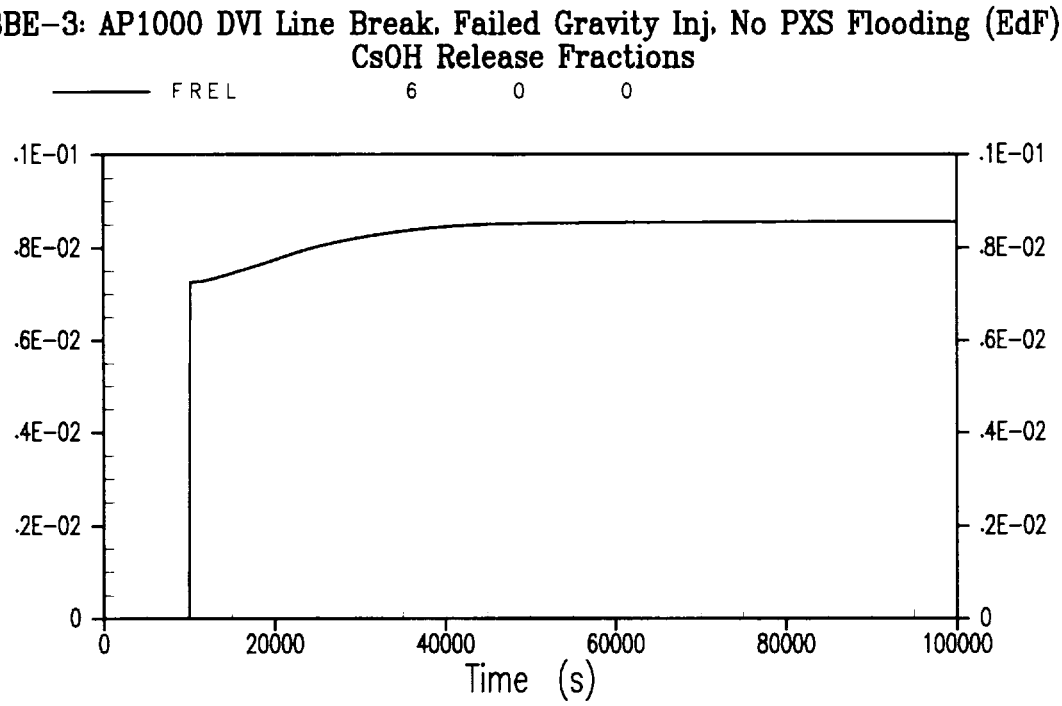


Figure 45-54
Release Category CFI, Case 3BE-3: Release Fraction of Cesium Hydroxide

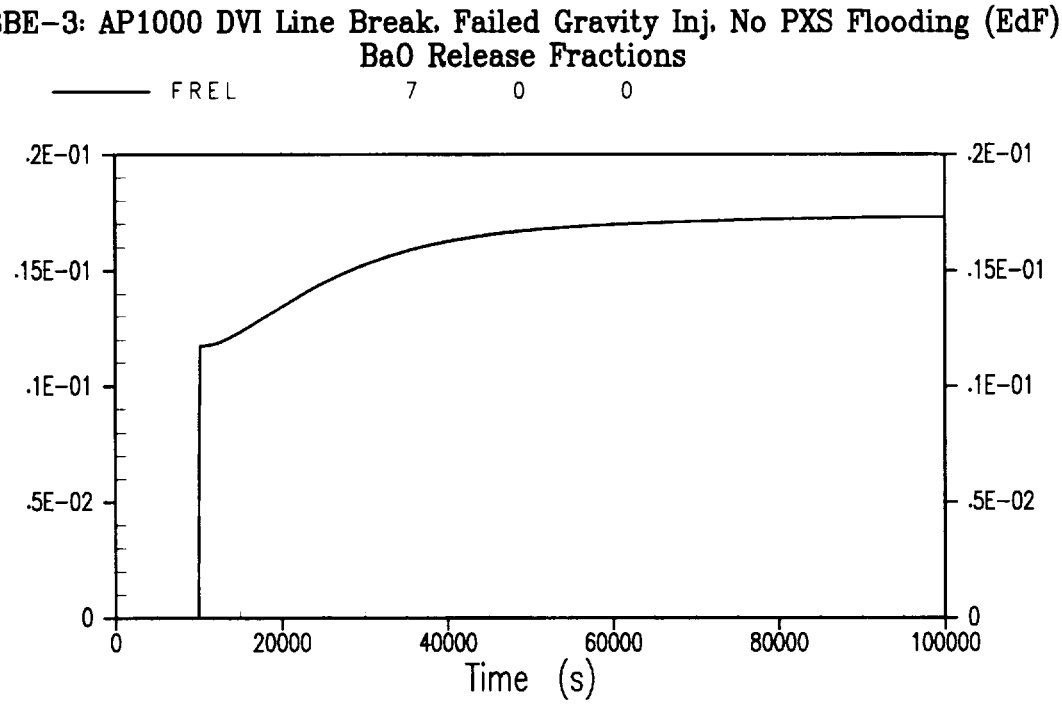


Figure 45-55
Release Category CFI, Case 3BE-3: Release Fraction of Barium Oxide

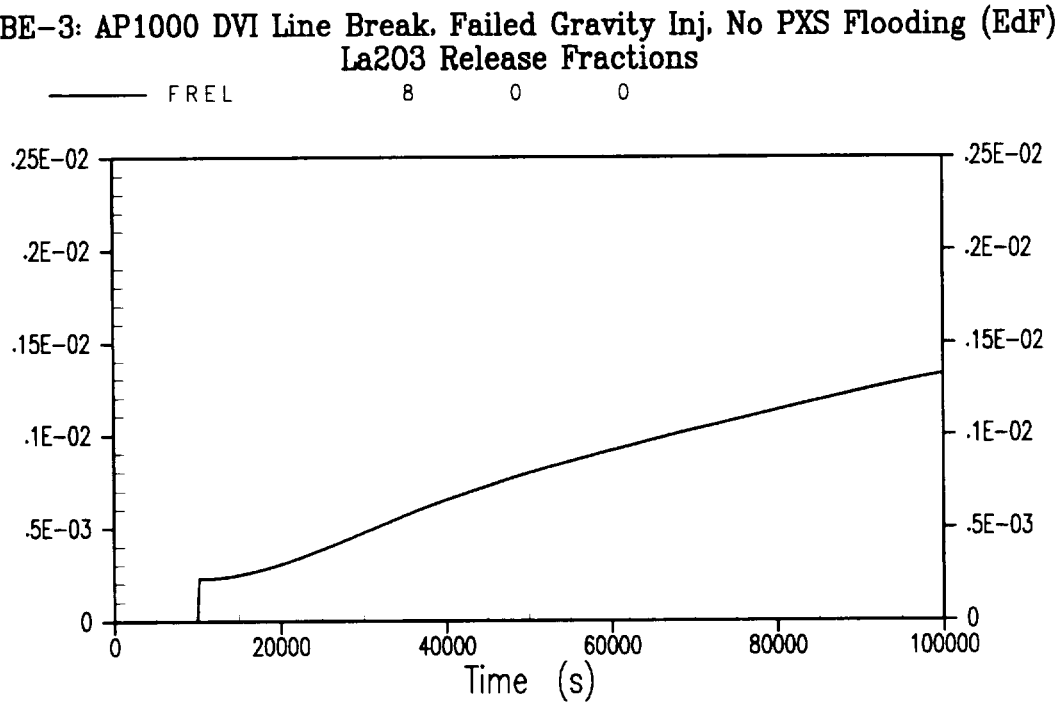


Figure 45-56
Release Category CFI, Case 3BE-3: Release Fraction of Dilanthanum Trioxide

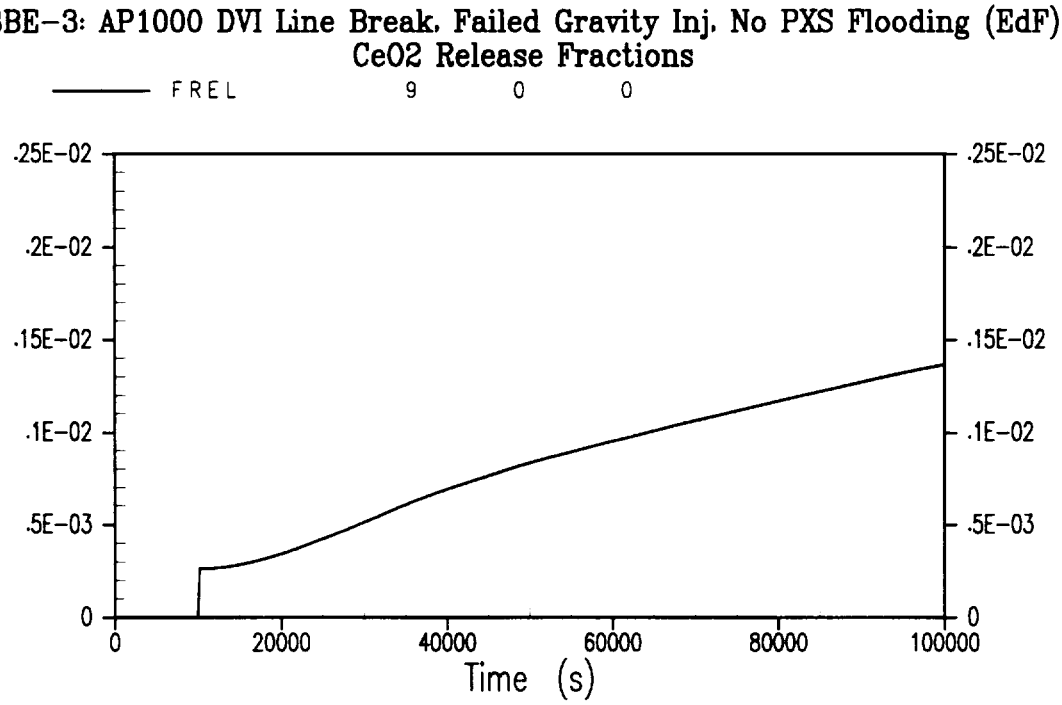


Figure 45-57
Release Category CFI, Case 3BE-3: Release Fraction of Cerium Dioxide

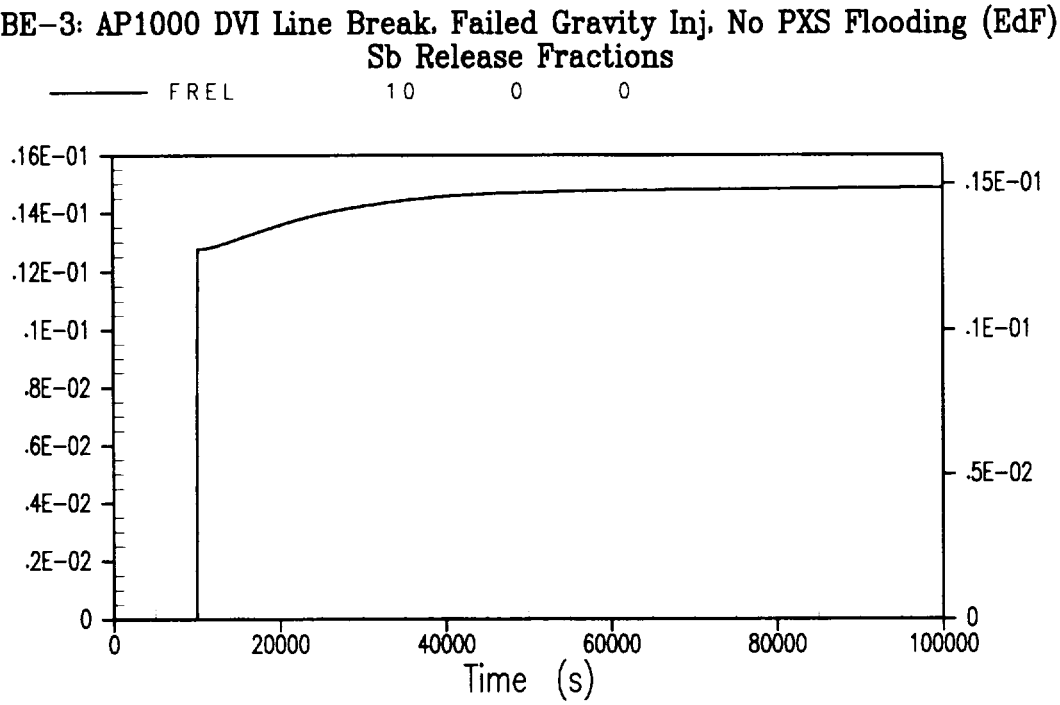


Figure 45-58
Release Category CFI, Case 3BE-3: Release Fraction of Tin

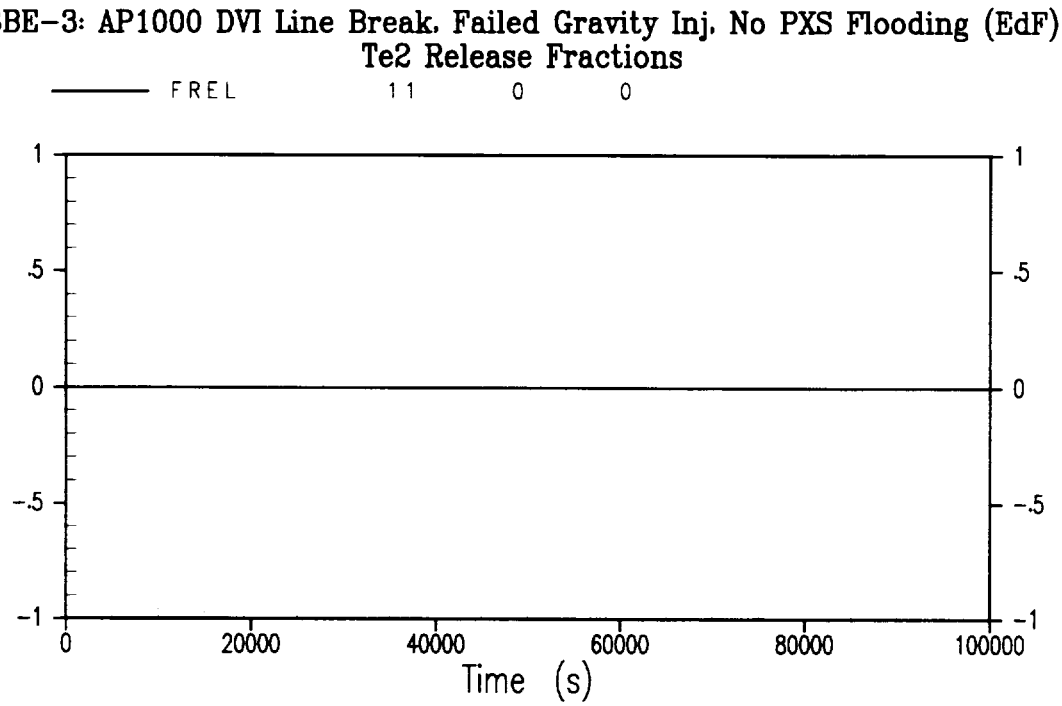


Figure 45-59
Release Category CFI, Case 3BE-3: Release Fraction of Tellurium

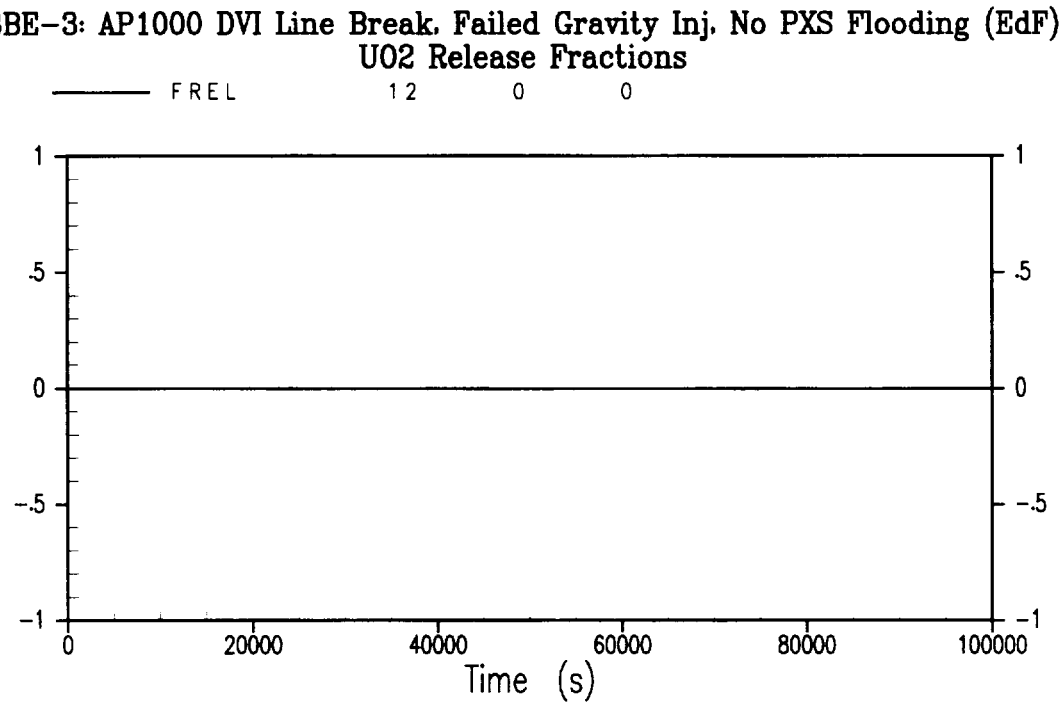


Figure 45-60
Release Category CFI, Case 3BE-3: Release Fraction of Uranium Dioxide

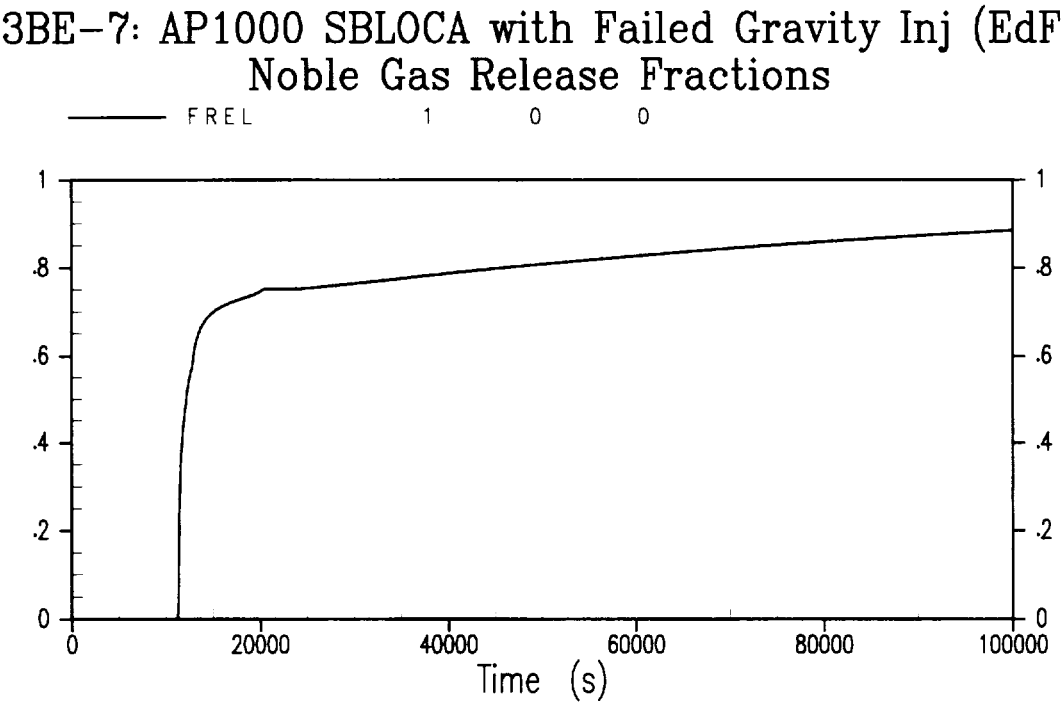


Figure 45-61
Release Category CFL, Case 3BE-7: Release Fraction of Noble Gases

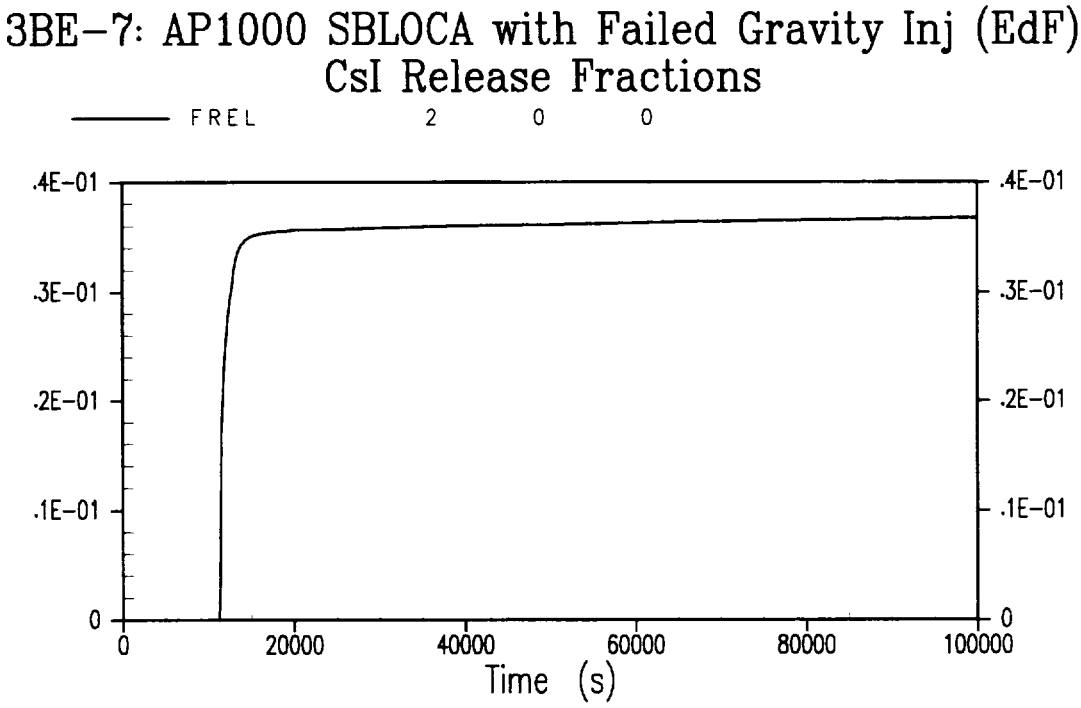


Figure 45-62
Release Category CFL, Case 3BE-7: Release Fraction of Cesium Iodide

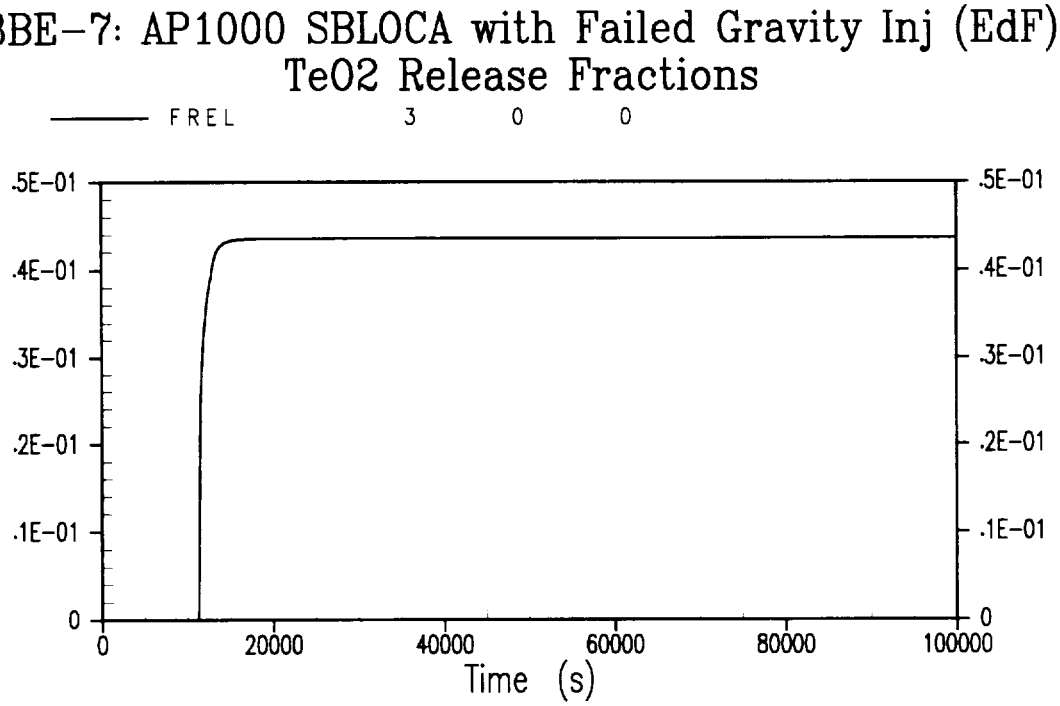


Figure 45-63
Release Category CFL, Case 3BE-7: Release Fraction of Tellurium Dioxide

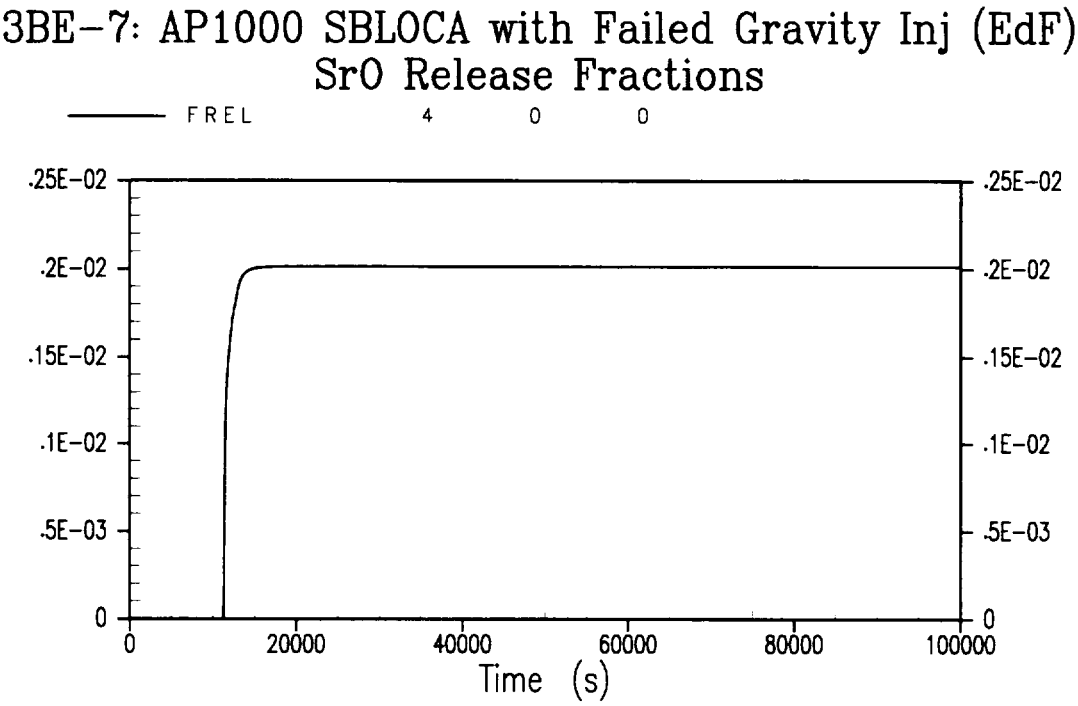


Figure 45-64
Release Category CFL, Case 3BE-7: Release Fraction of Strontium Oxide

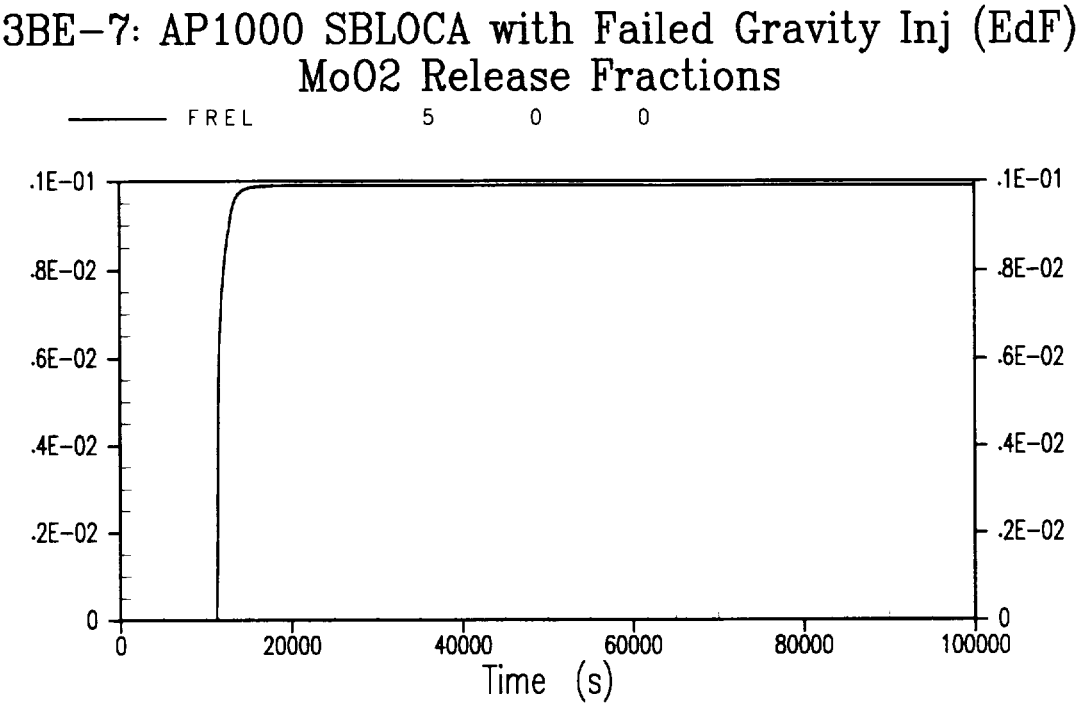


Figure 45-65
Release Category CFL, Case 3BE-7: Release Fraction of Molybdenum Dioxide

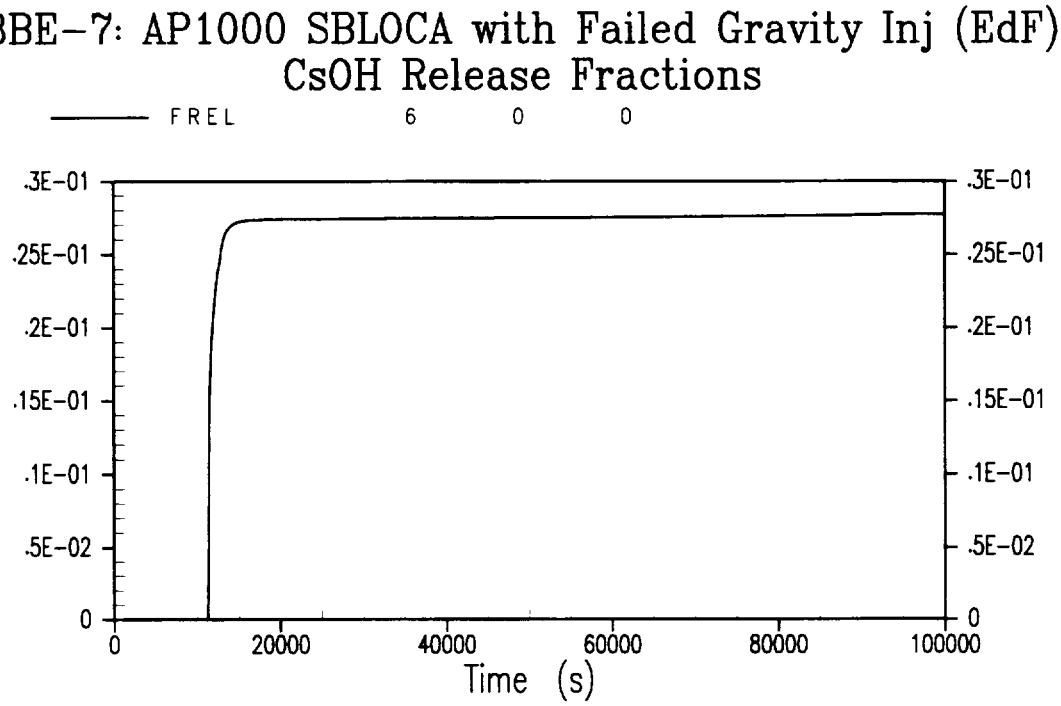


Figure 45-66
Release Category CFL, Case 3BE-7: Release Fraction of Cesium Hydroxide

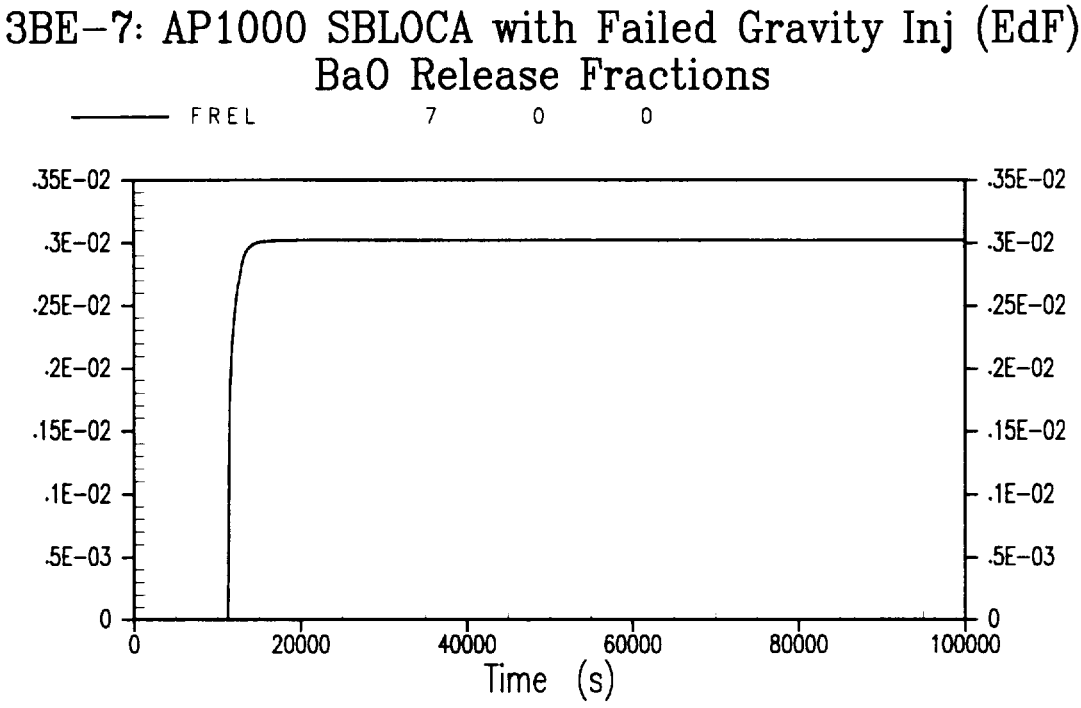


Figure 45-67
Release Category CFL, Case 3BE-7: Release Fraction of Barium Oxide

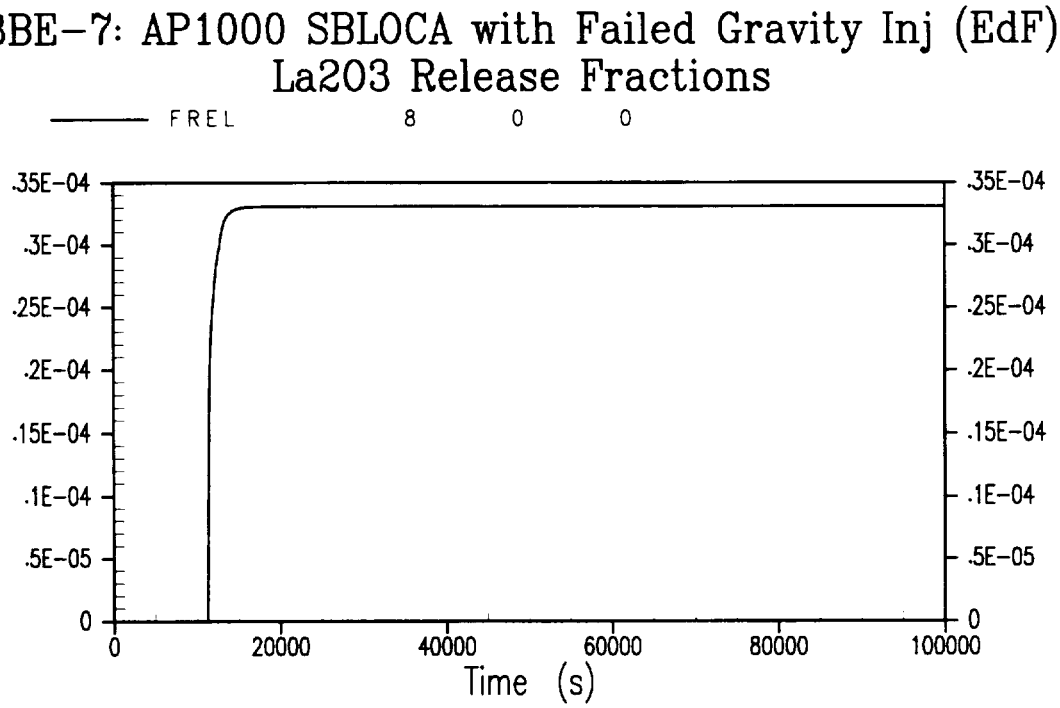


Figure 45-68
Release Category CFL, Case 3BE-7: Release Fraction of Dilanthanum Trioxide

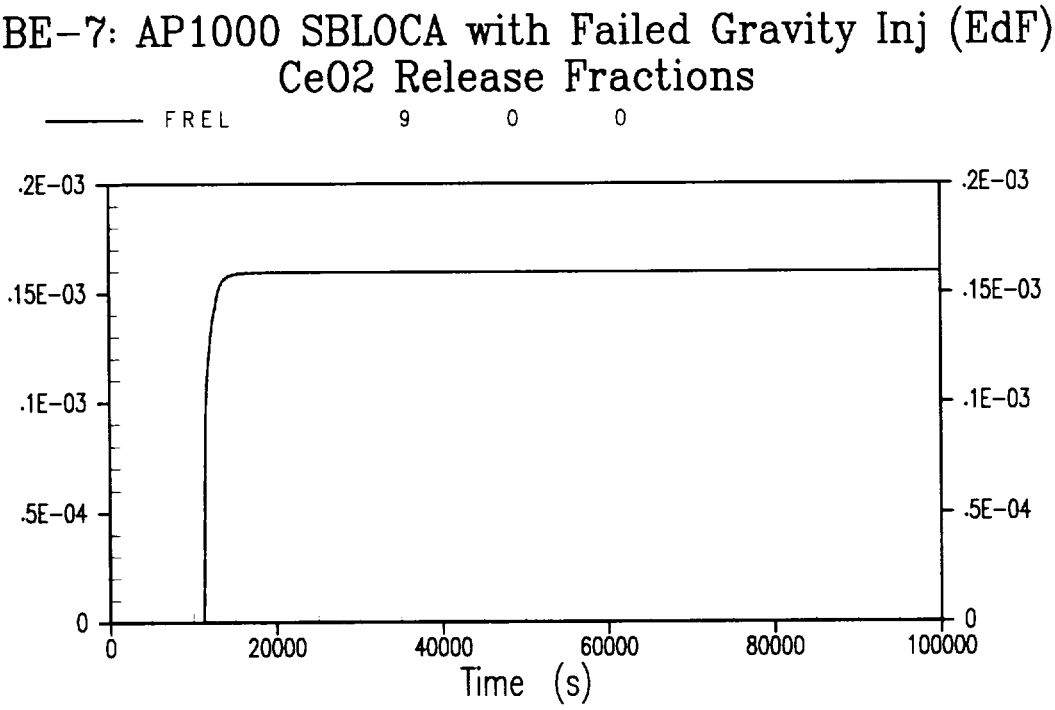


Figure 45-69
Release Category CFL, Case 3BE-7: Release Fraction of Cerium Dioxide

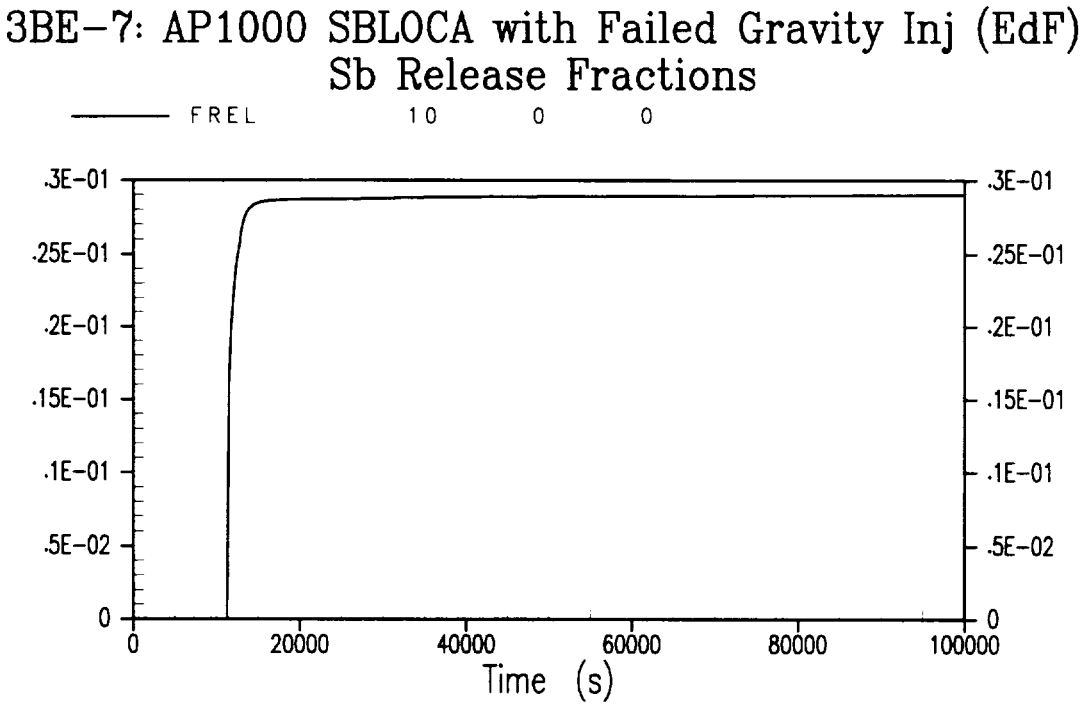


Figure 45-70
Release Category CFL, Case 3BE-7: Release Fraction of Tin

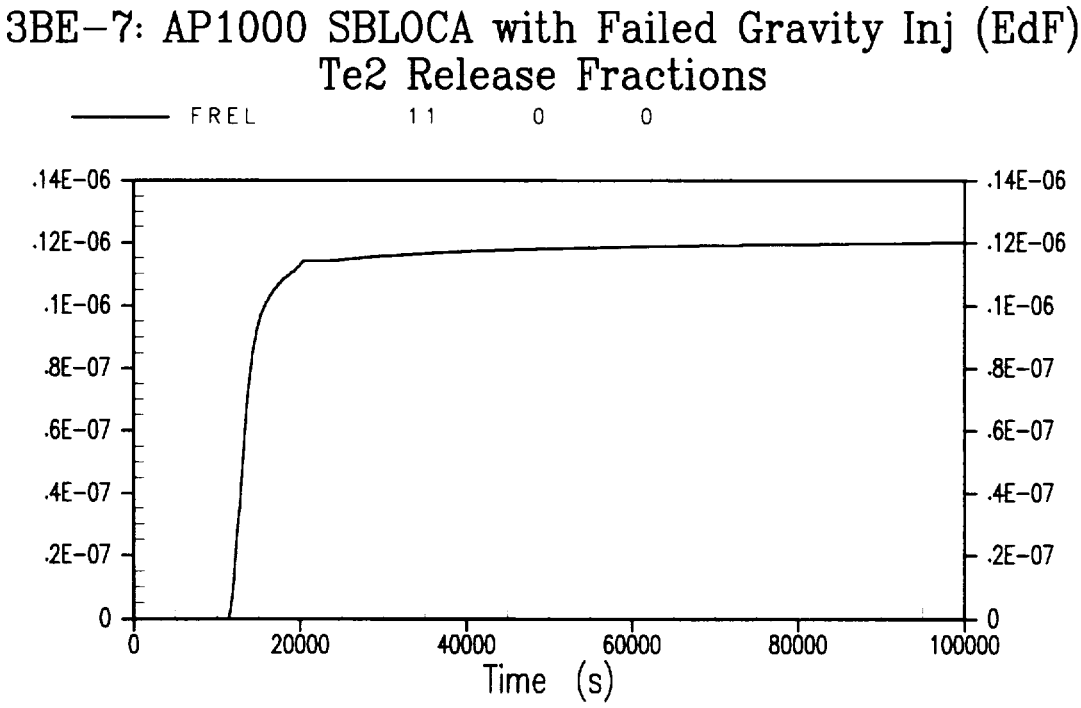


Figure 45-71
Release Category CFL, Case 3BE-7: Release Fraction of Tellurium

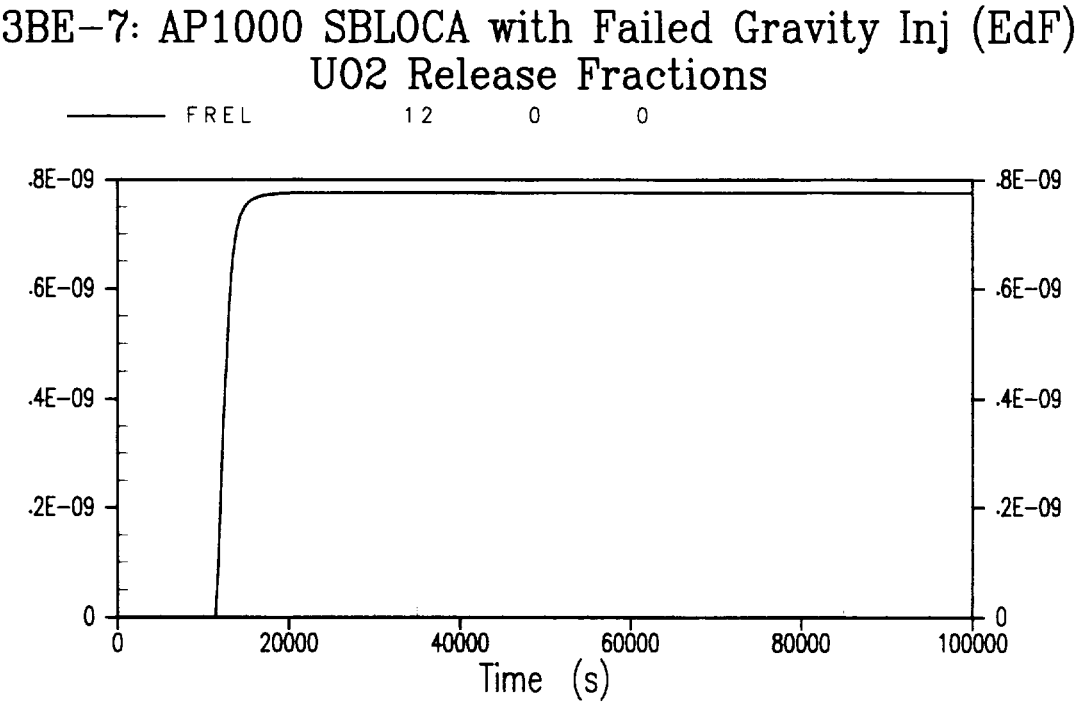


Figure 45-72
Release Category CFL, Case 3BE-7: Release Fraction of Uranium Dioxide

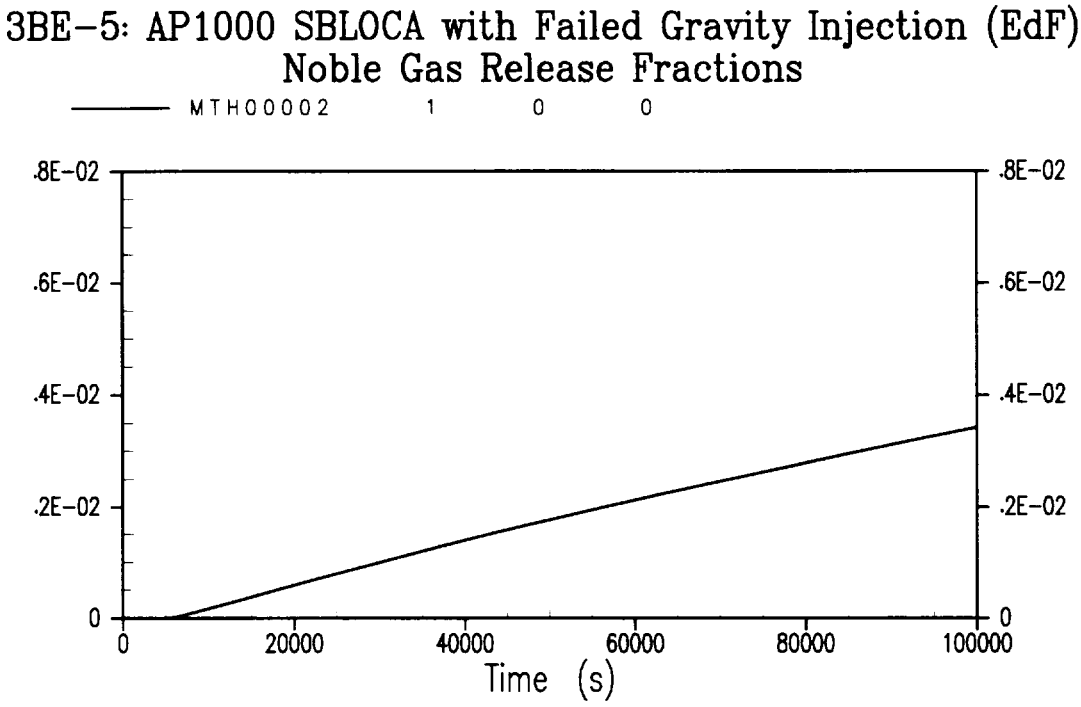


Figure 45-73
Release Category IC, Case 3BE-5 Direct-Release Sensitivity: Release Fraction of Noble Gases

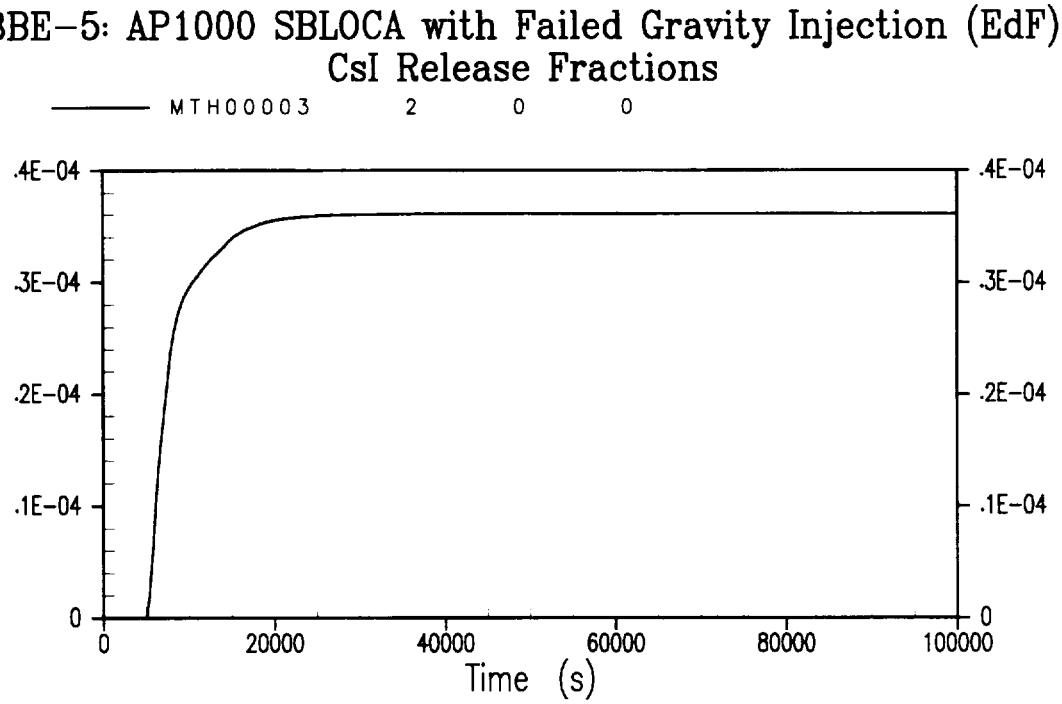


Figure 45-74
Release Category IC, Case 3BE-5 Direct-Release Sensitivity: Release Fraction of Cesium Iodide

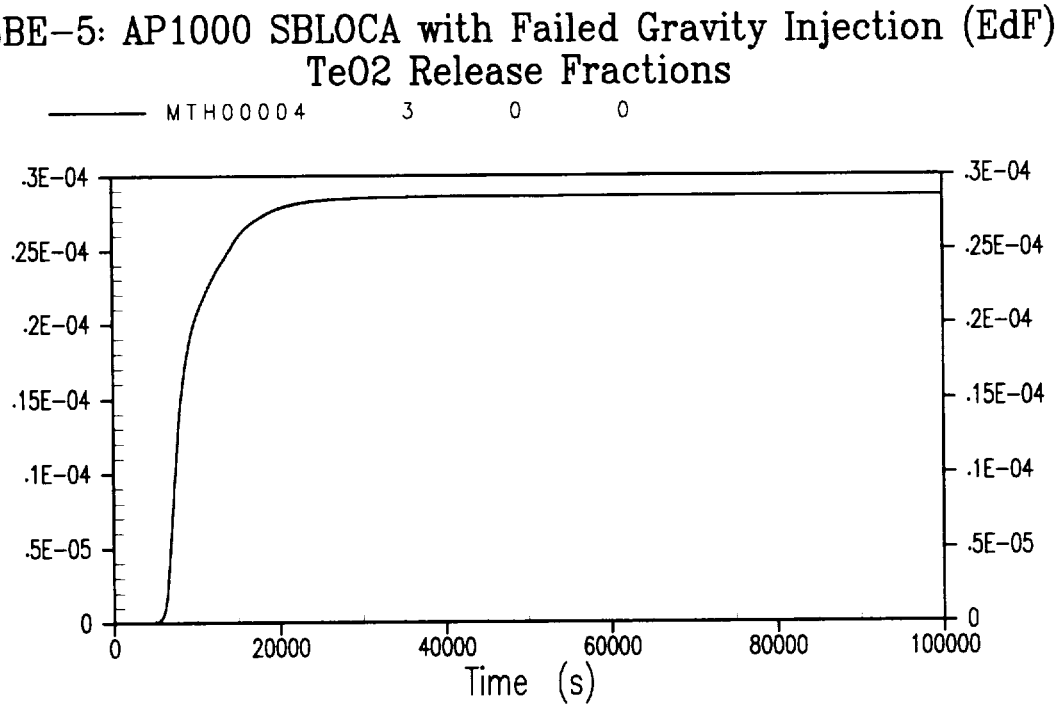


Figure 45-75
Release Category IC, Case 3BE-5 Direct-Release Sensitivity: Release Fraction of Tellurium Dioxide

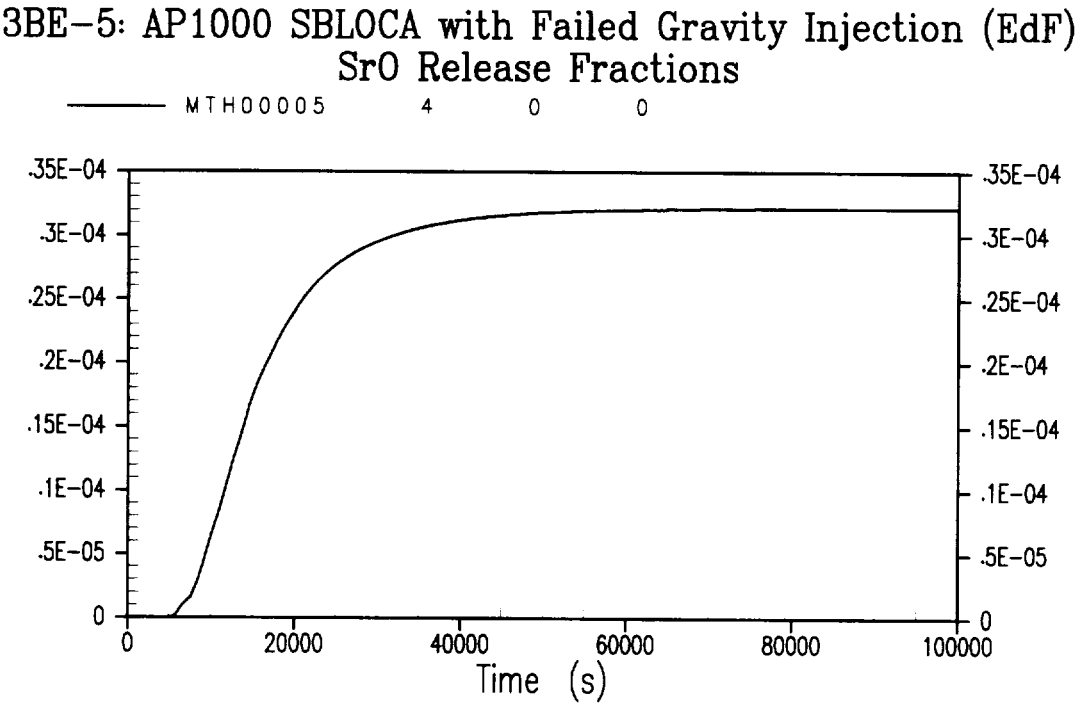


Figure 45-76
Release Category IC, Case 3BE-5 Direct-Release Sensitivity: Release Fraction of Strontium Oxide

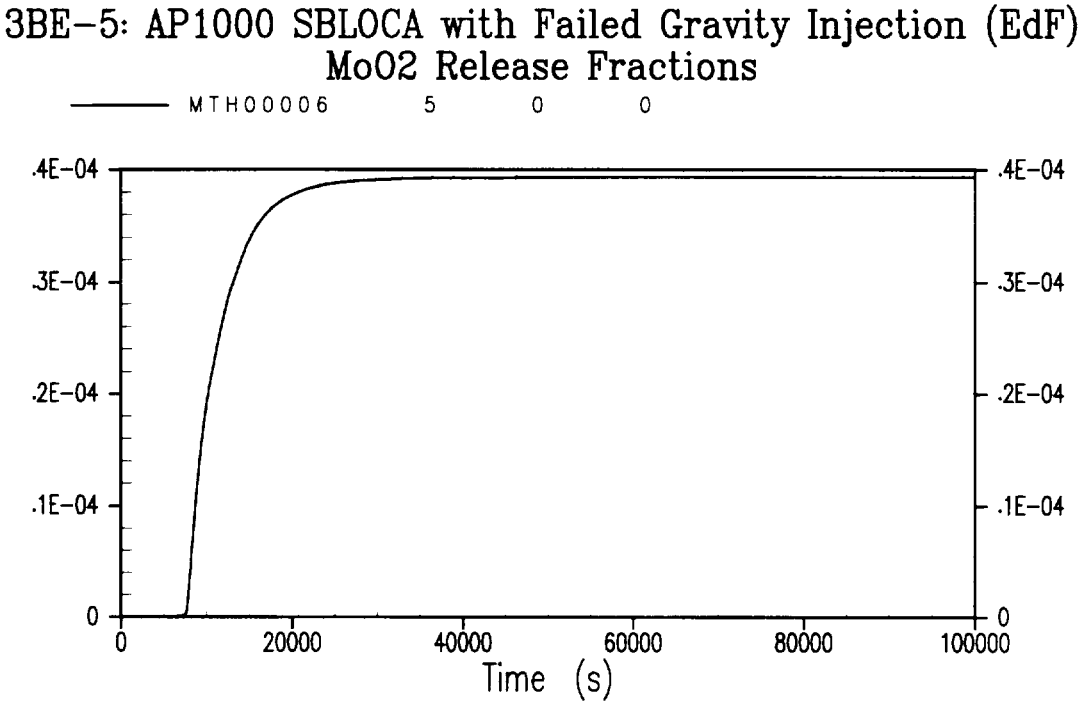


Figure 45-77
Release Category IC, Case 3BE-5 Direct-Release Sensitivity: Release Fraction of Molybdenum Dioxide

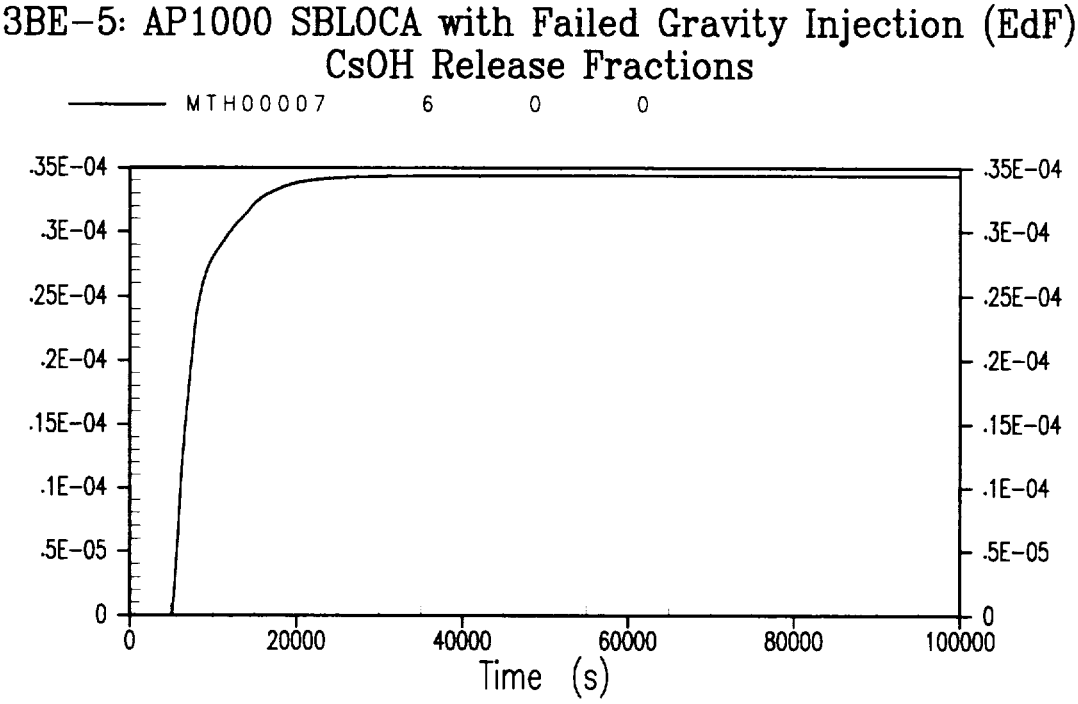


Figure 45-78
Release Category IC, Case 3BE-5 Direct-Release Sensitivity: Release Fraction of Cesium Hydroxide

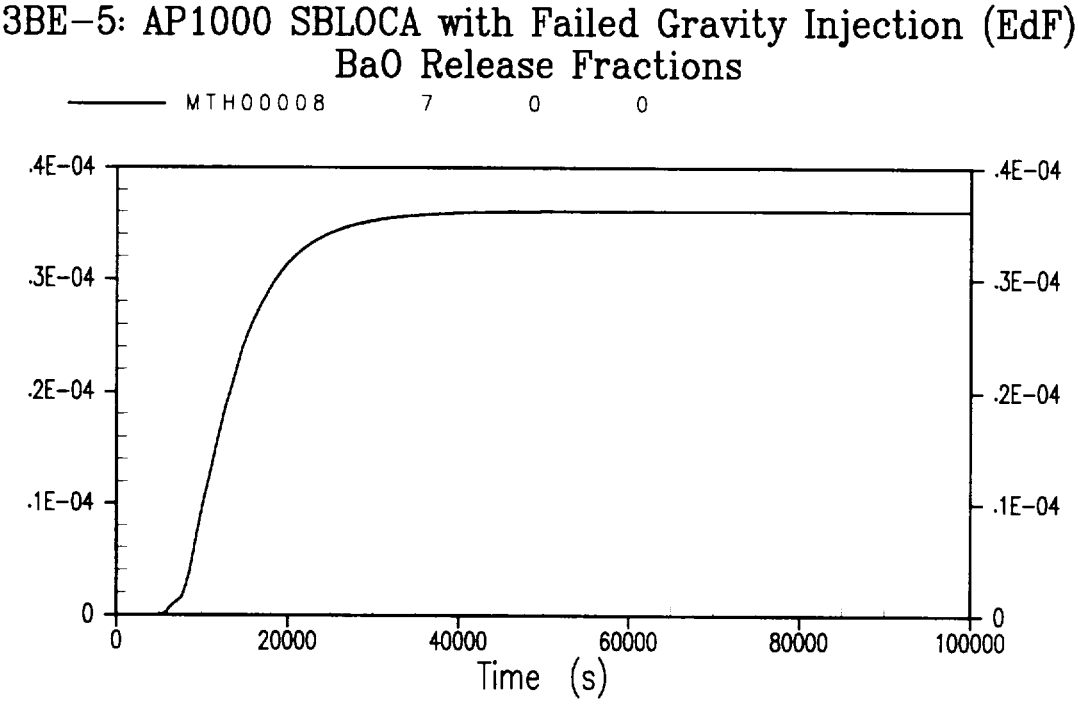


Figure 45-79
Release Category IC, Case 3BE-5 Direct-Release Sensitivity: Release Fraction of Barium Oxide

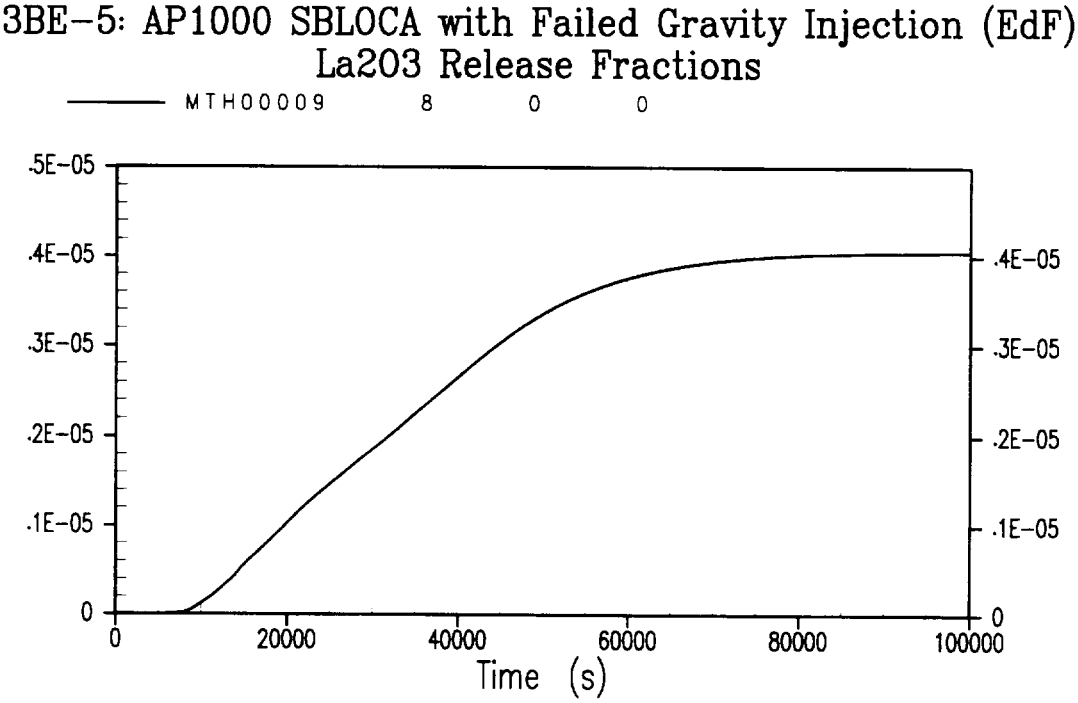


Figure 45-80
Release Category IC, Case 3BE-5 Direct-Release Sensitivity: Release Fraction of Dilanthanum Trioxide

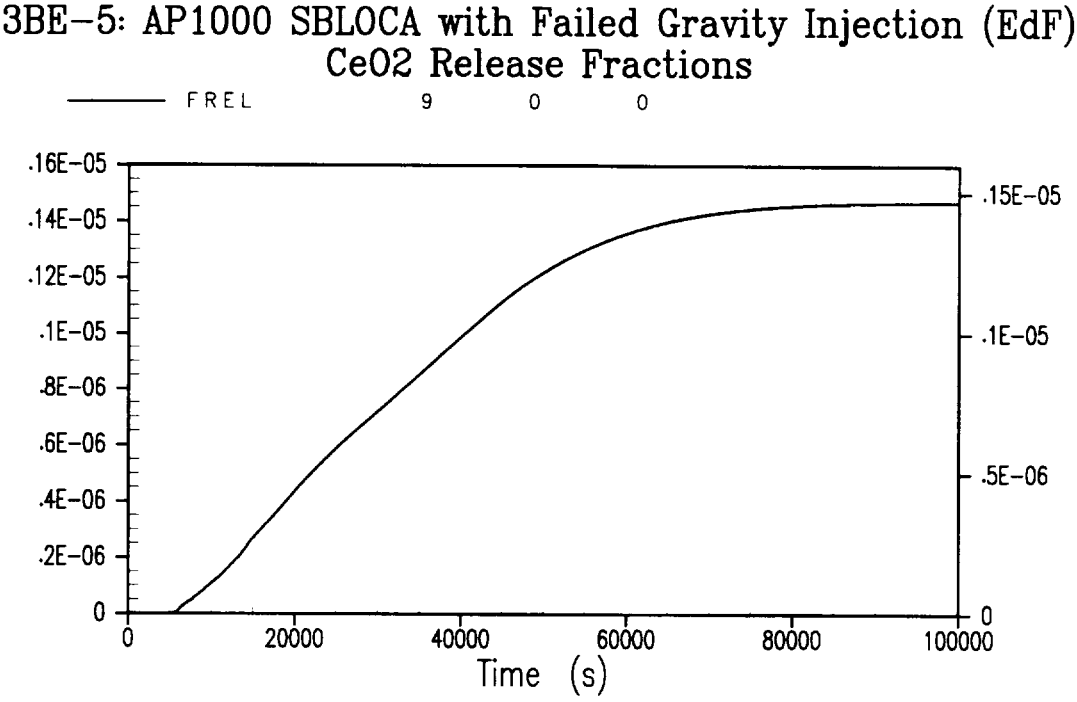


Figure 45-81
Release Category IC, Case 3BE-5 Direct-Release Sensitivity: Release Fraction of Cerium Dioxide

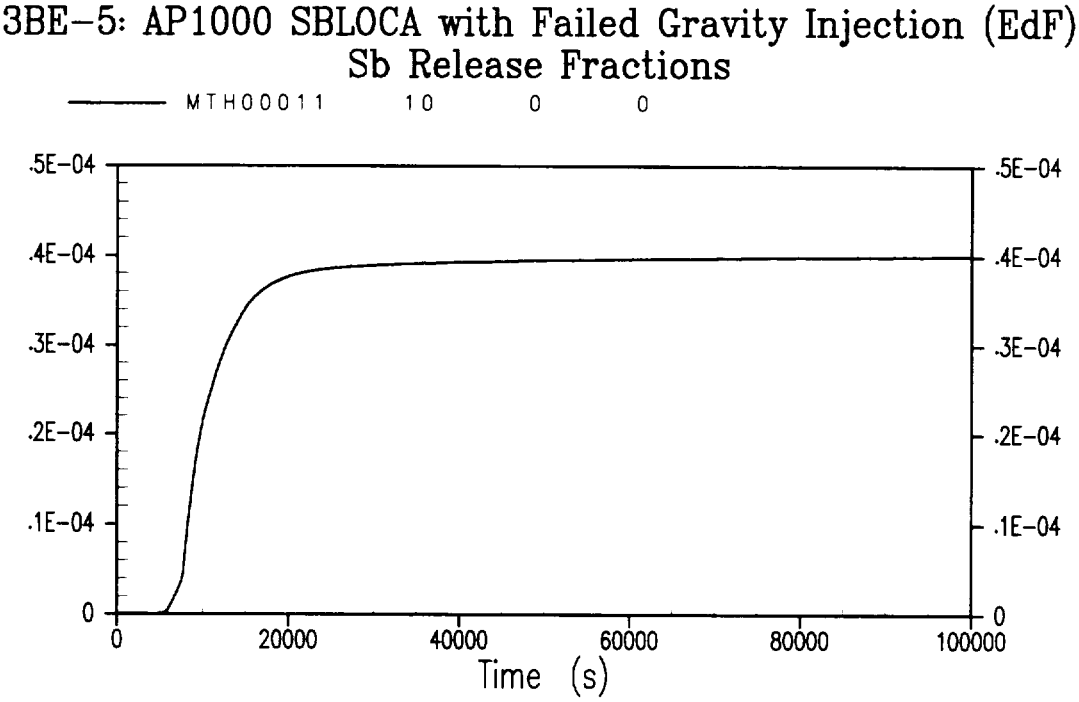


Figure 45-82
Release Category IC, Case 3BE-5 Direct-Release Sensitivity: Release Fraction of Tin

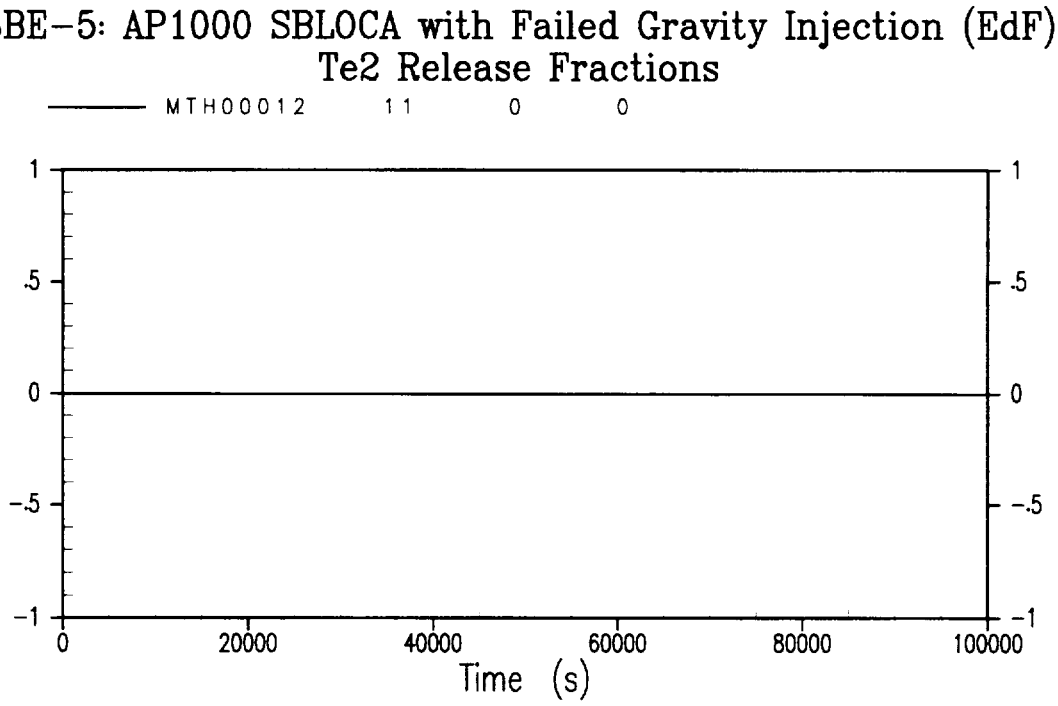


Figure 45-83
Release Category IC, Case 3BE-5 Direct-Release Sensitivity: Release Fraction of Tellurium

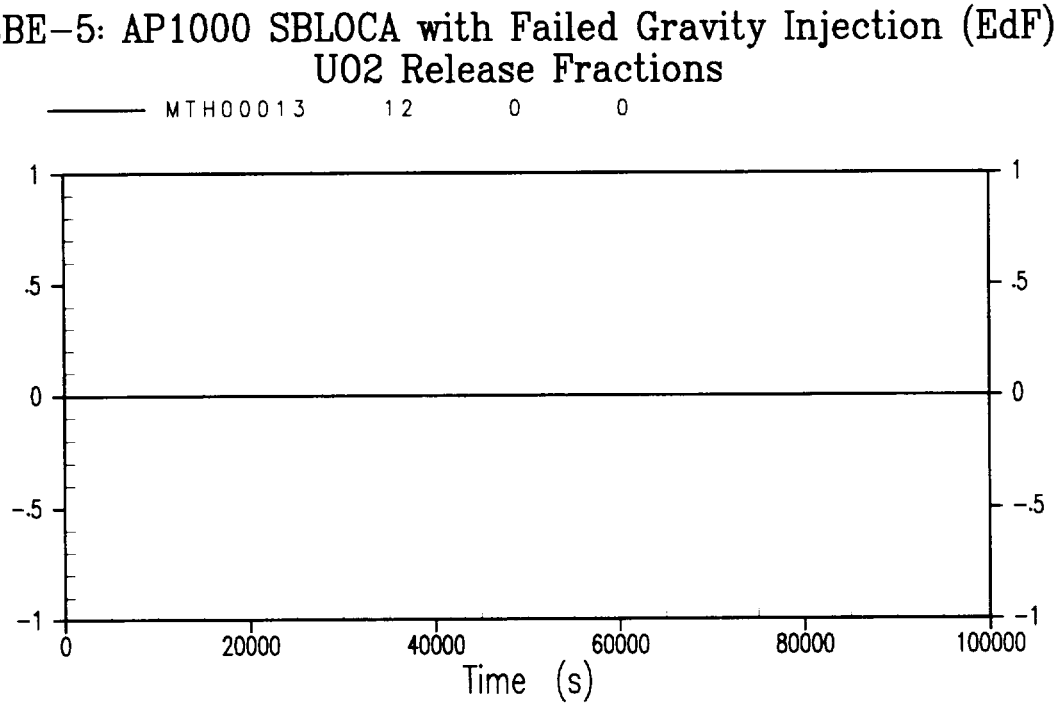


Figure 45-84
Release Category IC, Case 3BE-5 Direct-Release Sensitivity: Release Fraction of Uranium Dioxide