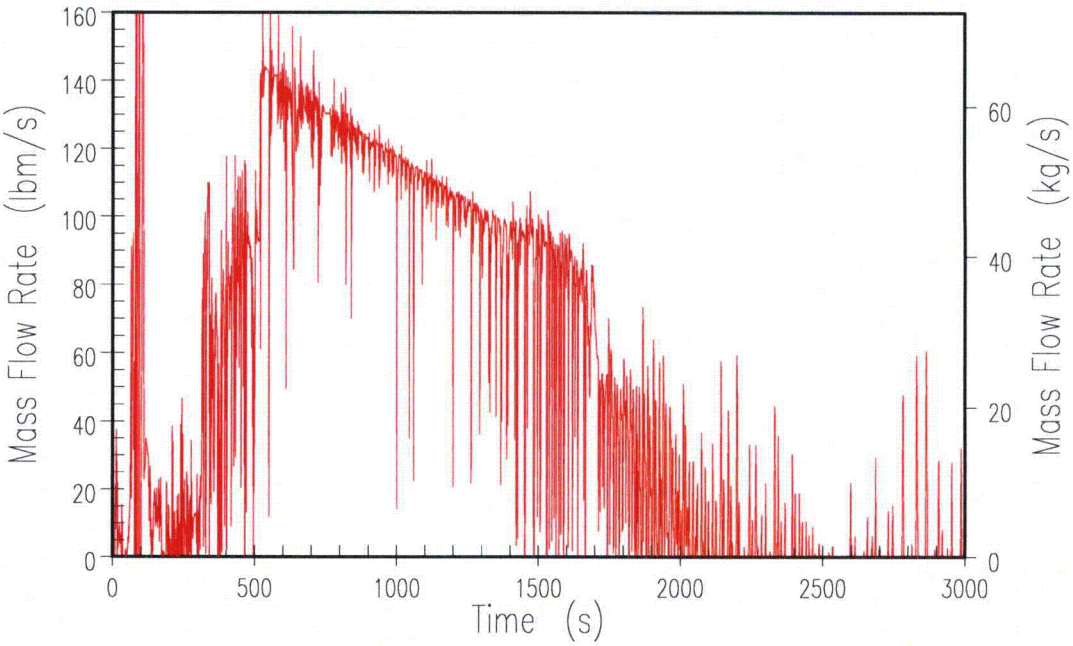

Figure 15.6.5.4B-61

10-Inch Cold Leg Break – CMT-1 Injection Rate

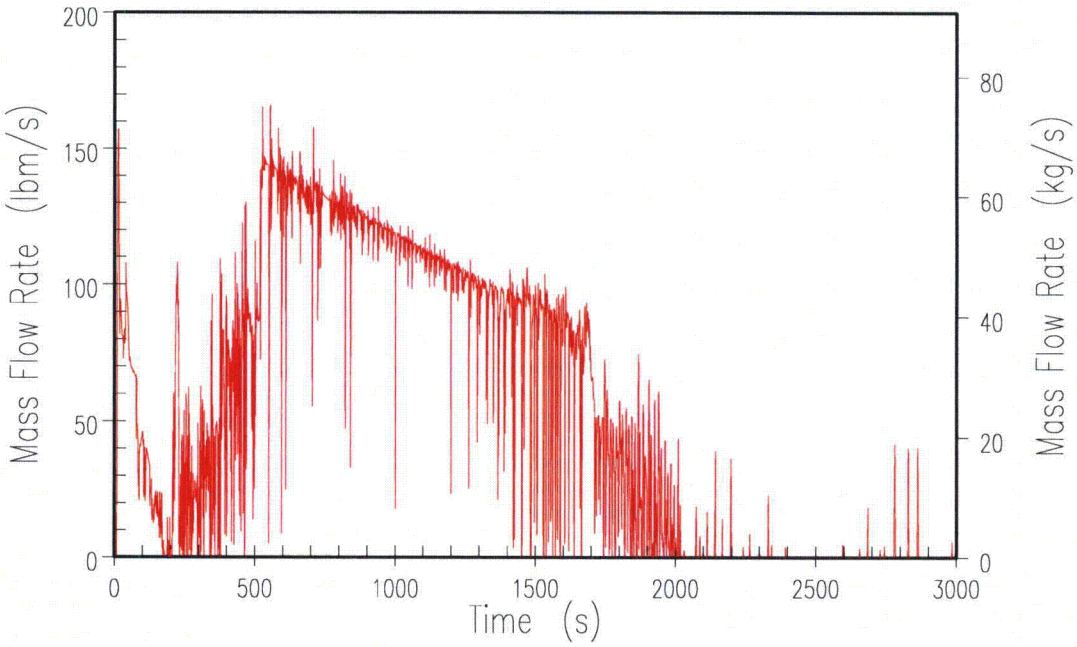

Figure 15.6.5.4B-62

10-Inch Cold Leg Break – CMT-2 Injection Rate

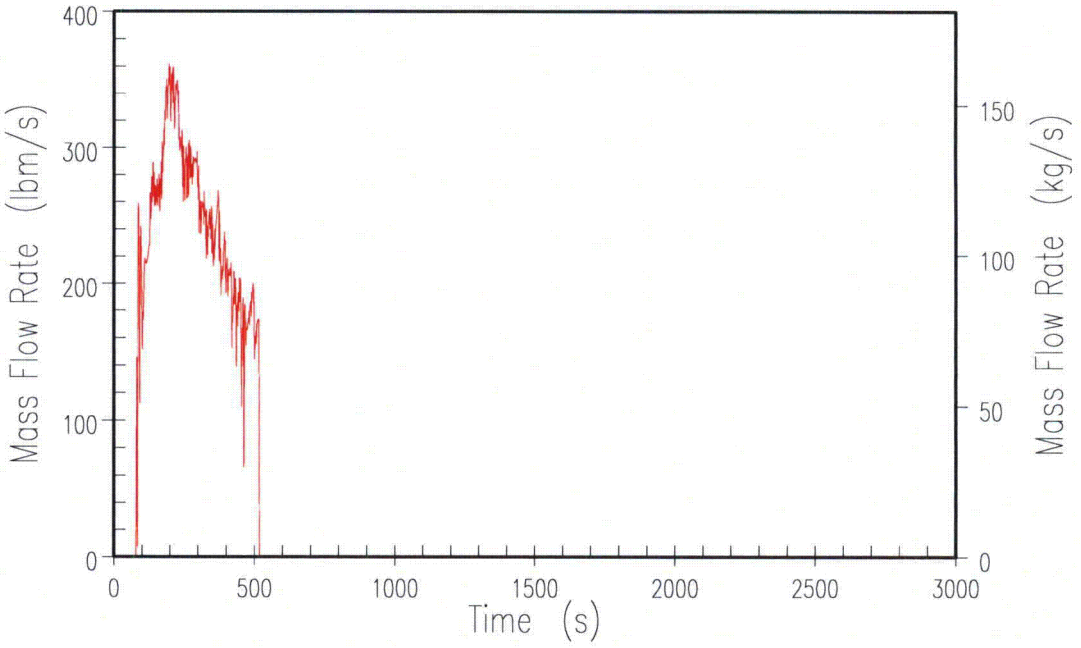

Figure 15.6.5.4B-63

10-Inch Cold Leg Break – Accumulator-1 Injection Rate

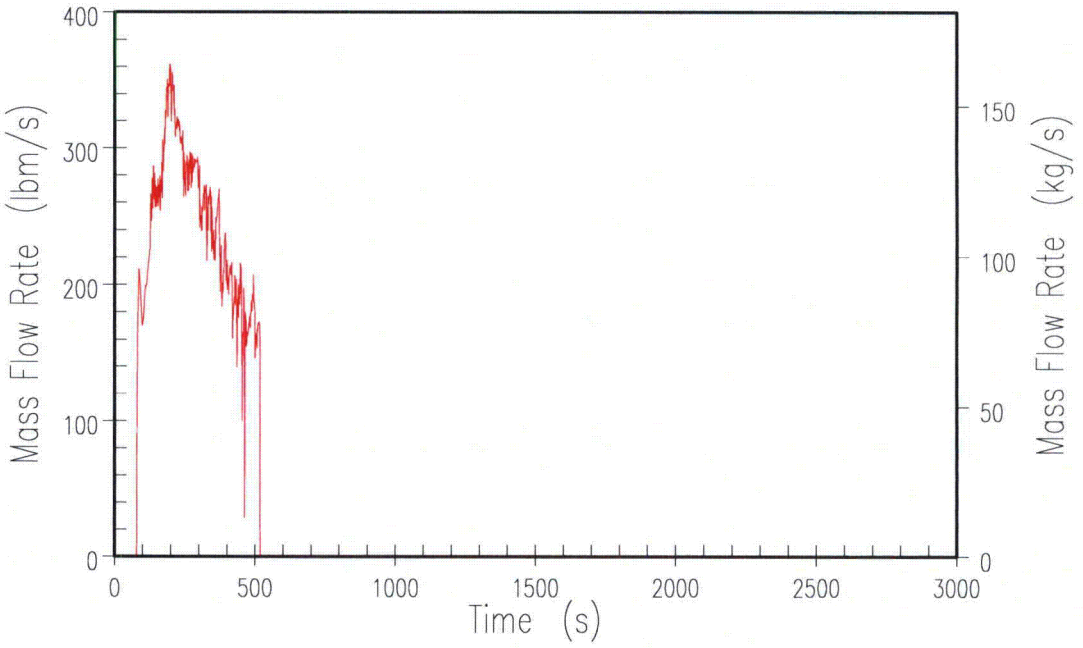

Figure 15.6.5.4B-64

10-Inch Cold Leg Break – Accumulator-2 Injection Rate

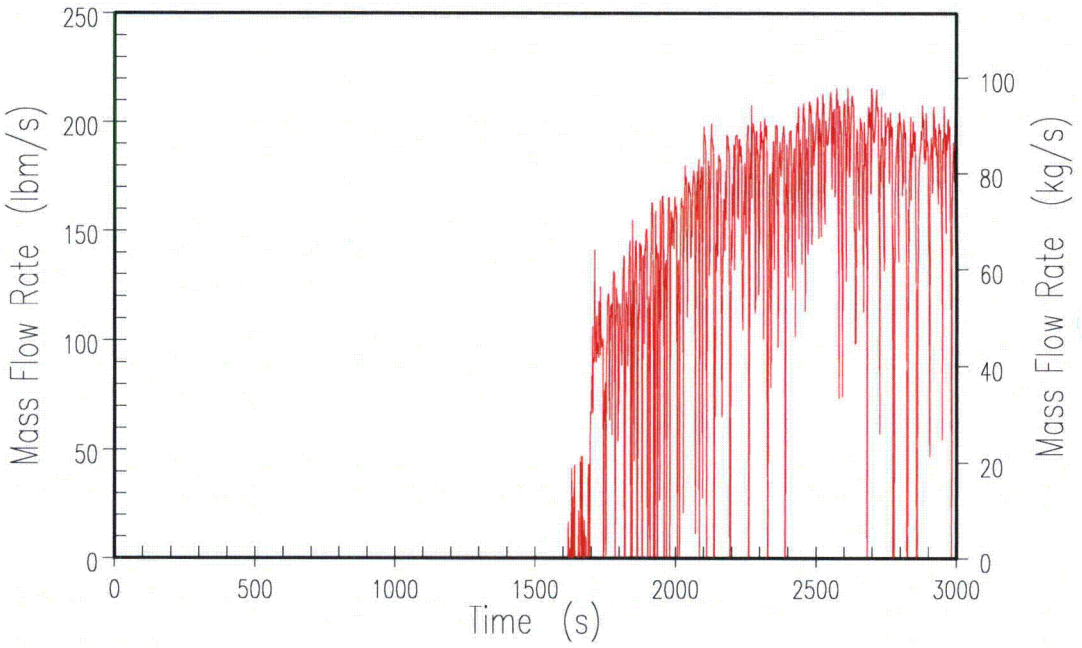

Figure 15.6.5.4B-65

10-Inch Cold Leg Break – IRWST-1 Injection Rate

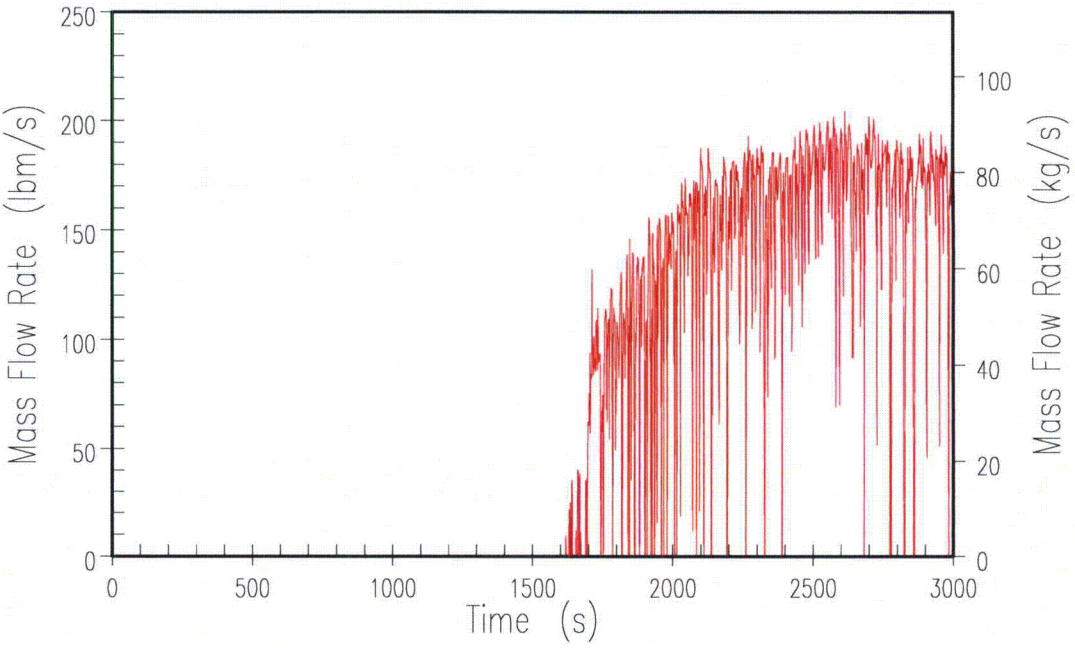

Figure 15.6.5.4B-66

10-Inch Cold Leg Break – IRWST-2 Injection Rate

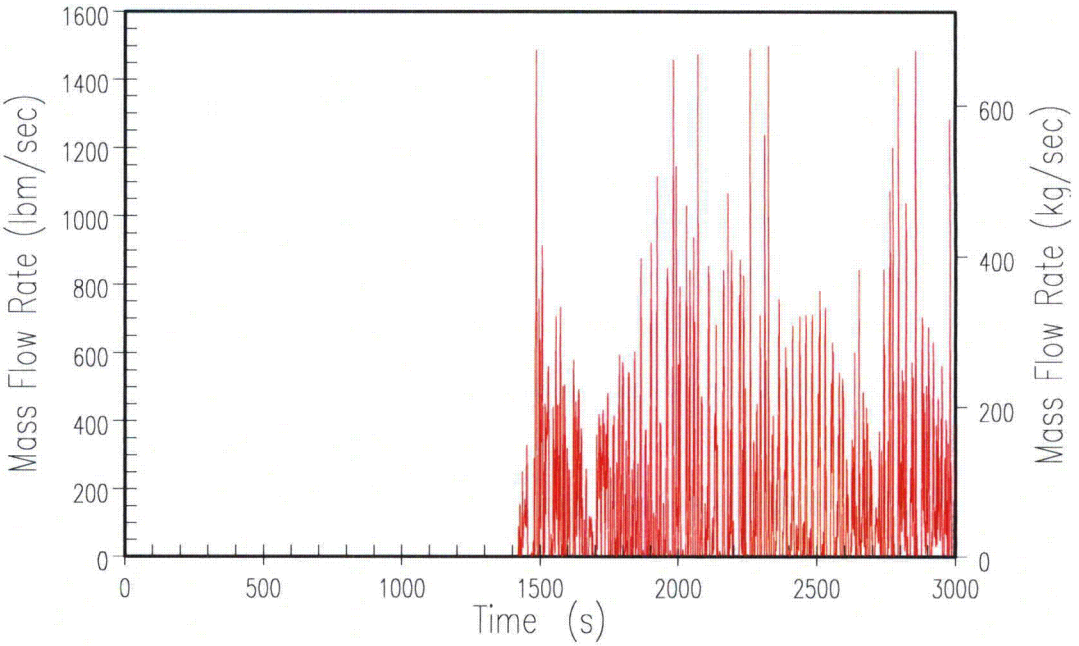

Figure 15.6.5.4B-67(a)

10-Inch Cold Leg Break – ADS-4 Liquid Discharge

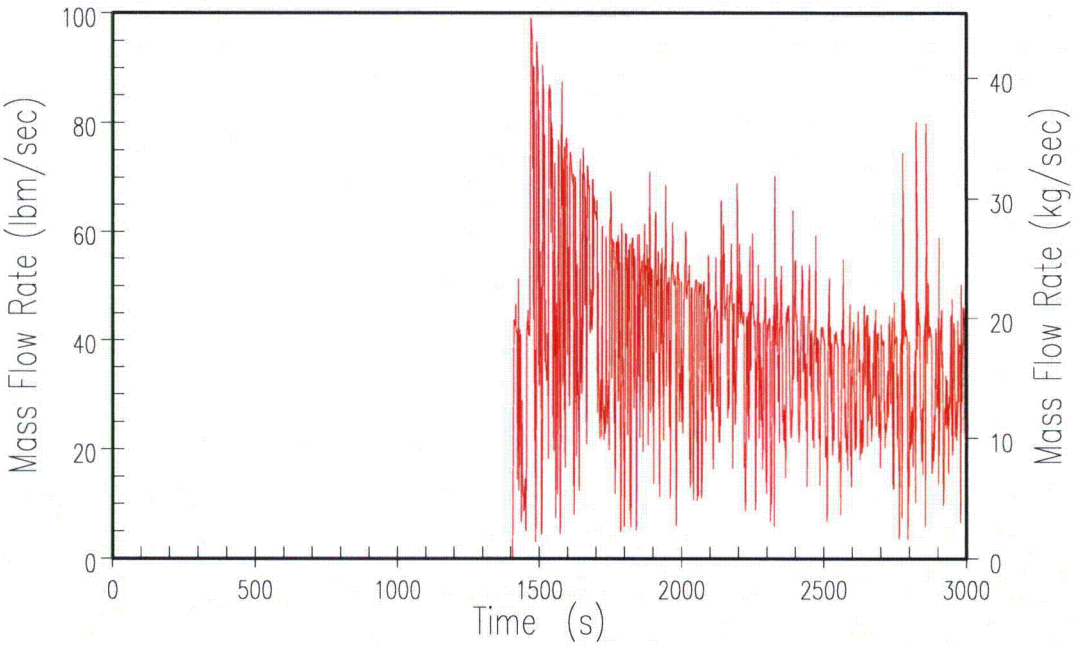

Figure 15.6.5.4B-67(b)

10-Inch Cold Leg Break – ADS-4 Vapor Discharge

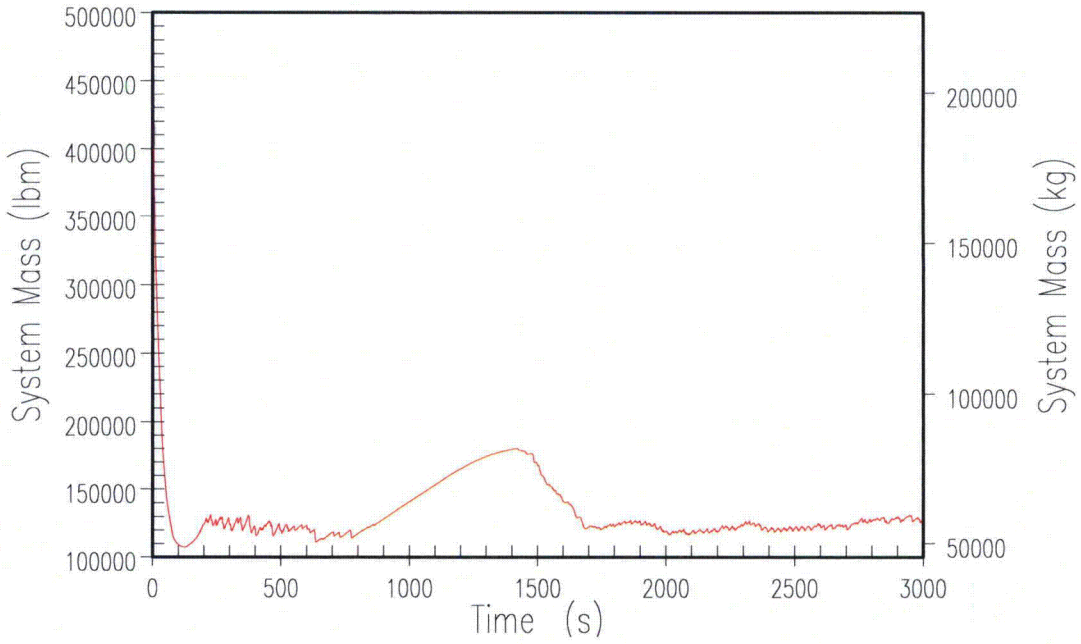

Figure 15.6.5.4B-68(a)

10-Inch Cold Leg Break – RCS System Inventory

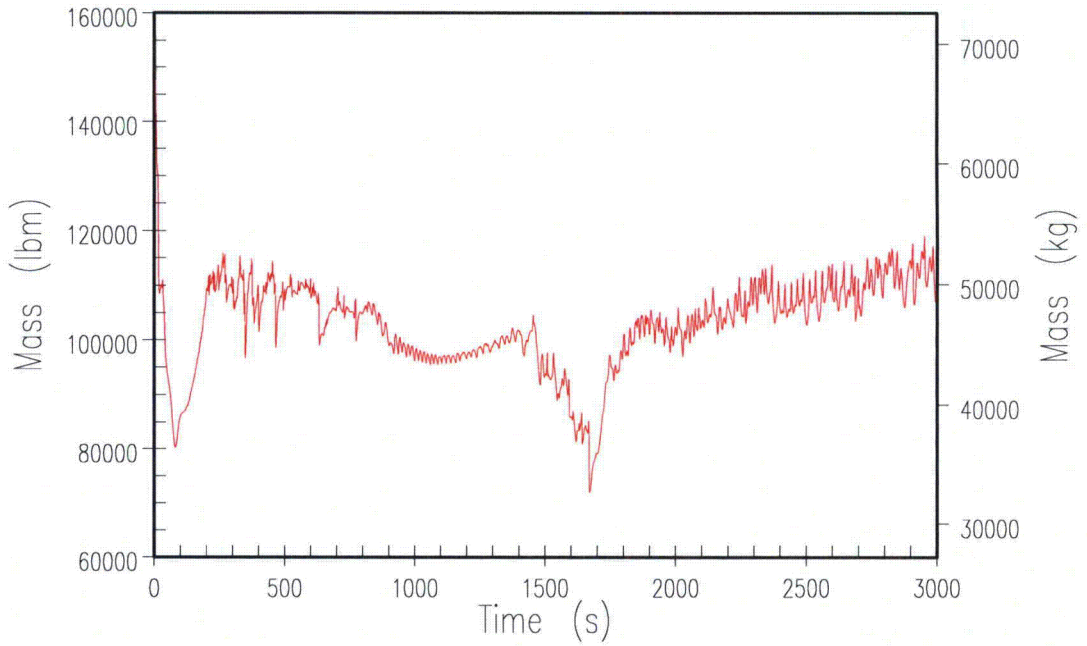

Figure 15.6.5.4B-68(b)

10-Inch Cold Leg Break – Reactor Vessel Mixture Inventory

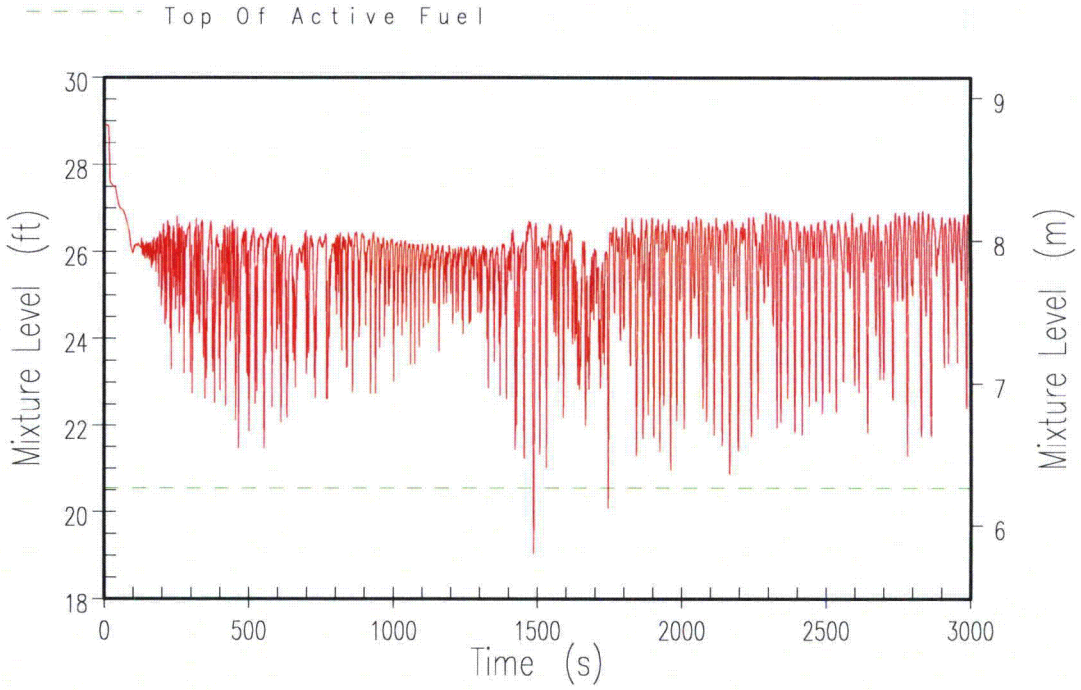

Figure 15.6.5.4B-69

10-Inch Cold Leg Break – Core/Upper Plenum Mixture Level

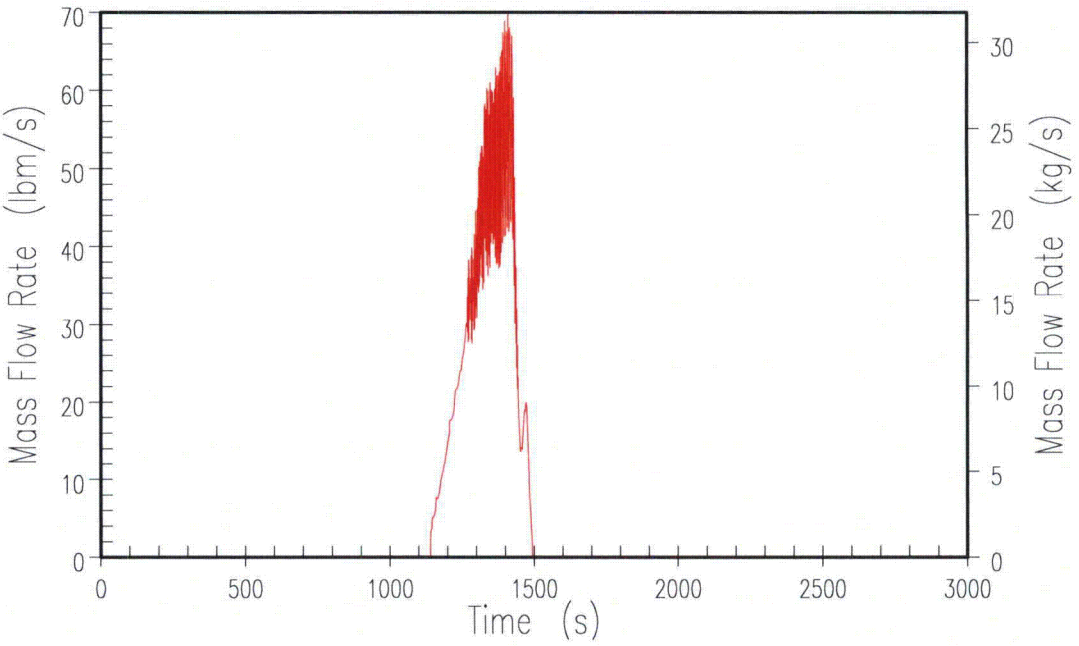

Figure 15.6.5.4B-70(a)

10-Inch Cold Leg Break – ADS 1-3 Liquid Discharge

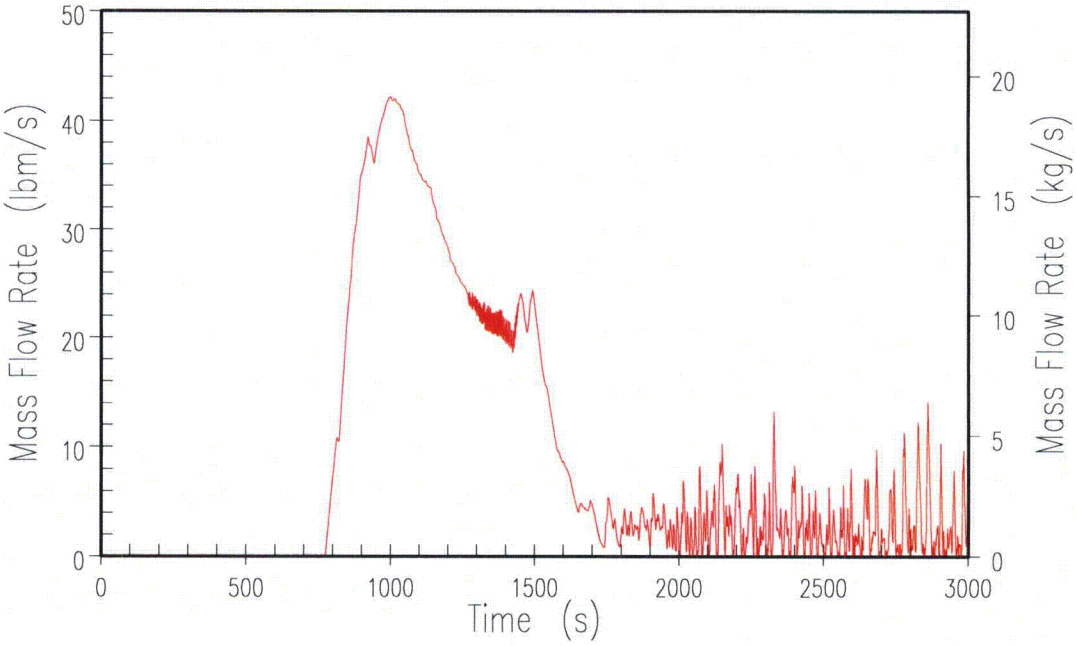

Figure 15.6.5.4B-70(b)

10-Inch Cold Leg Break – ADS 1-3 Vapor Discharge

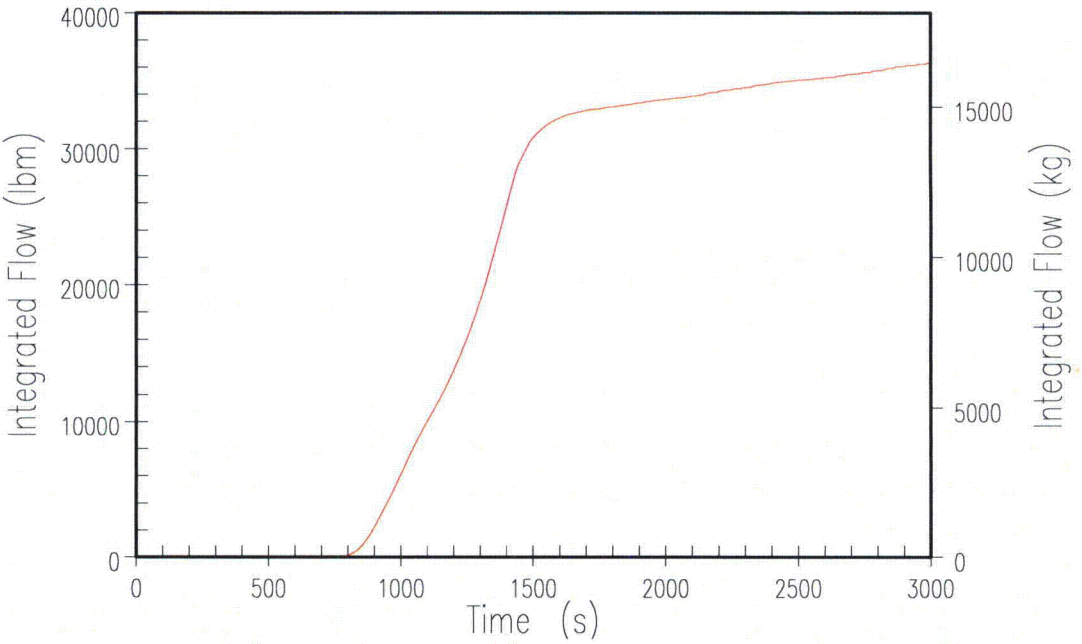

Figure 15.6.5.4B-70(c)

10-Inch Cold Leg Break – ADS 1-3 Integrated Discharge

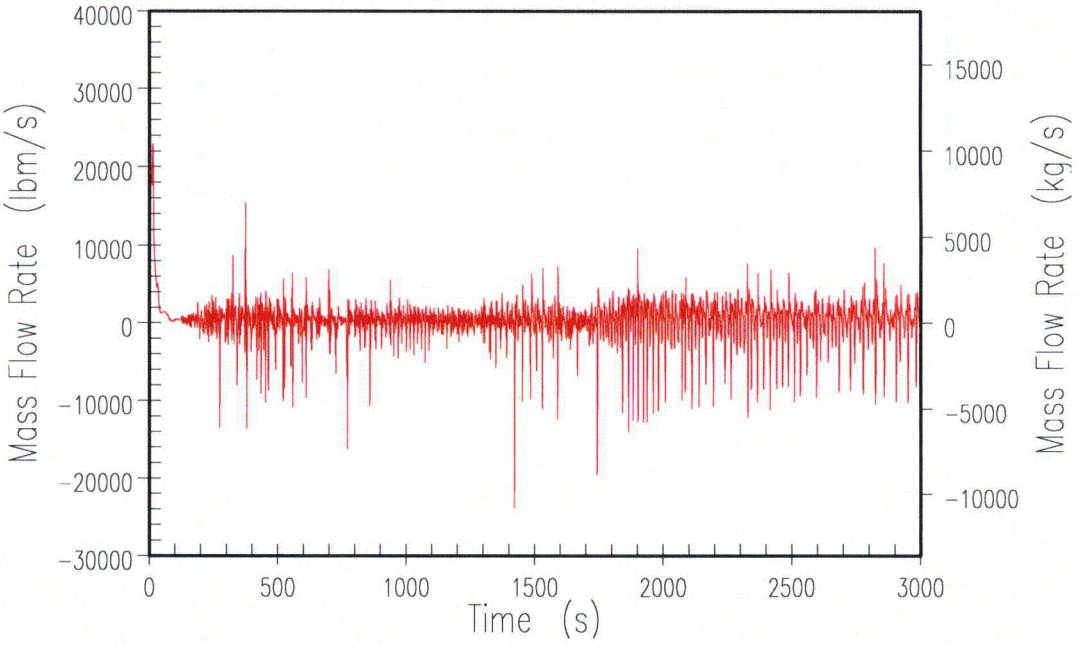

Figure 15.6.5.4B-71

10-Inch Cold Leg Break – Core Exit Liquid Flow

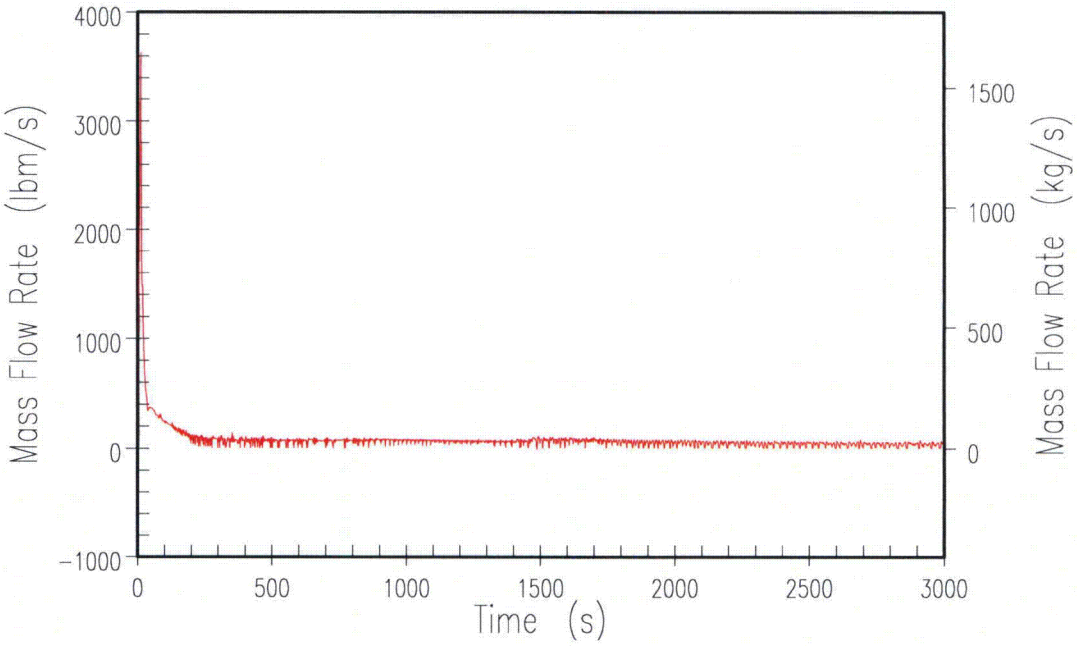

Figure 15.6.5.4B-72

10-Inch Cold Leg Break – Core Exit Vapor Flow

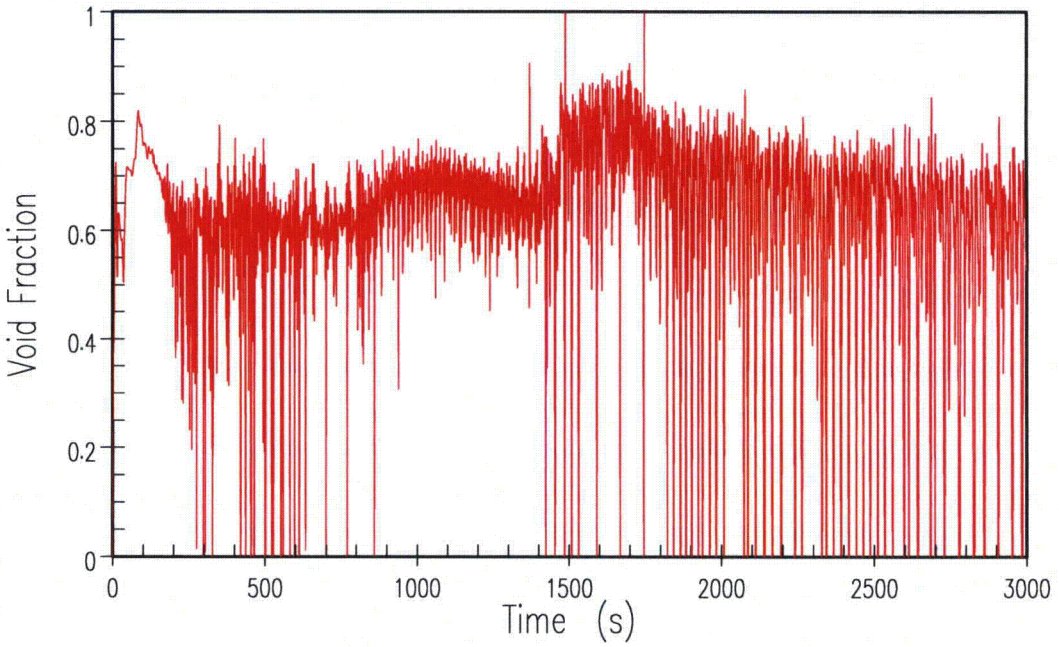

Figure 15.6.5.4B-73

10-Inch Cold Leg Break – Core Exit Void Fraction

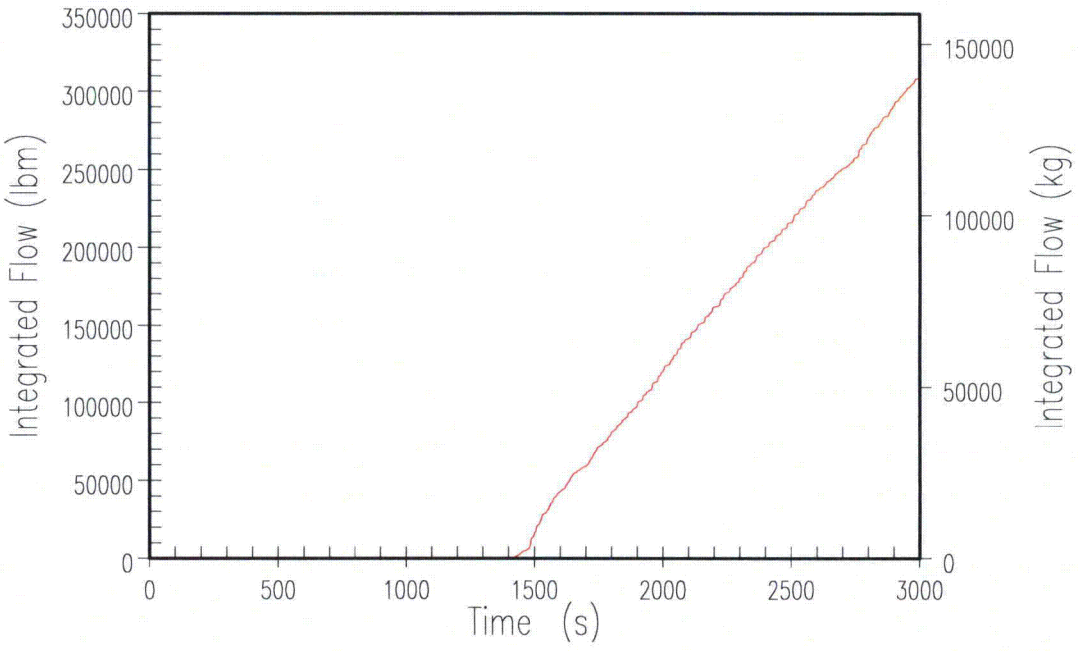

Figure 15.6.5.4B-74

10-Inch Cold Leg Break – ADS-4 Integrated Discharge

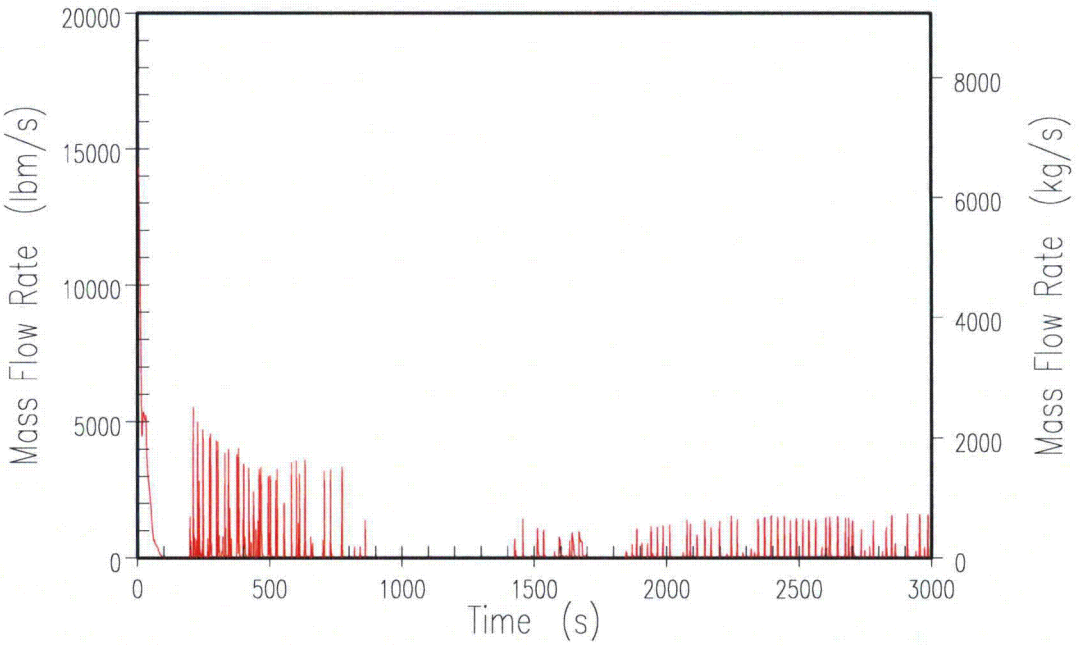

Figure 15.6.5.4B-75

10-Inch Cold Leg Break – Liquid Break Discharge

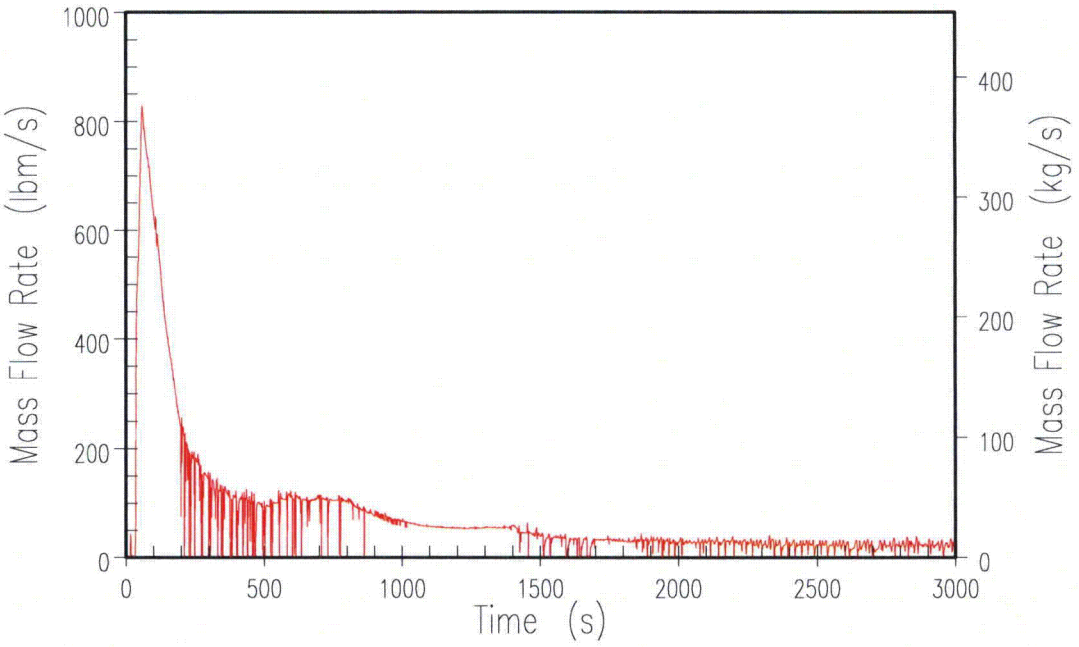

Figure 15.6.5.4B-76

10-Inch Cold Leg Break – Vapor Break Discharge

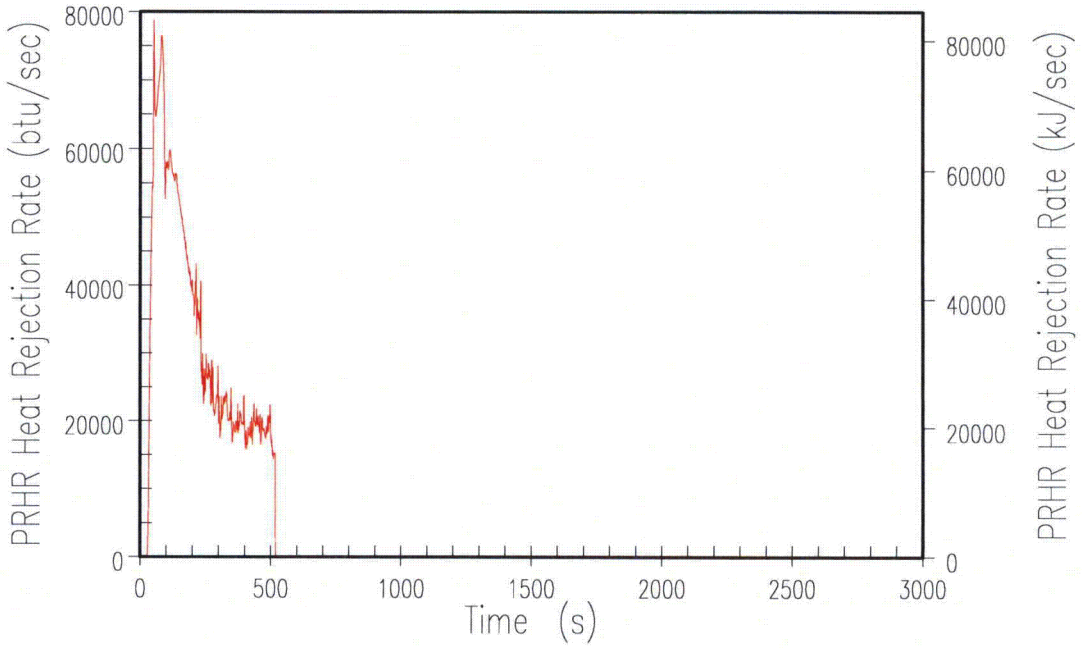

Figure 15.6.5.4B-77

10-Inch Cold Leg Break – PRHR Heat Removal Rate

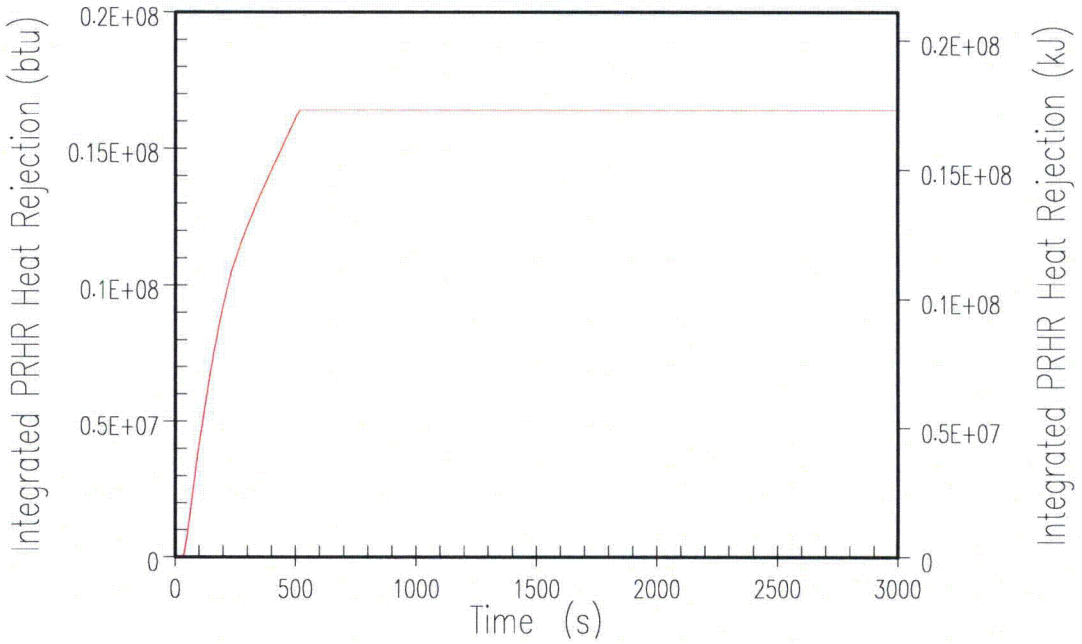

Figure 15.6.5.4B-78

10-Inch Cold Leg Break – Integrated PRHR Heat Removal

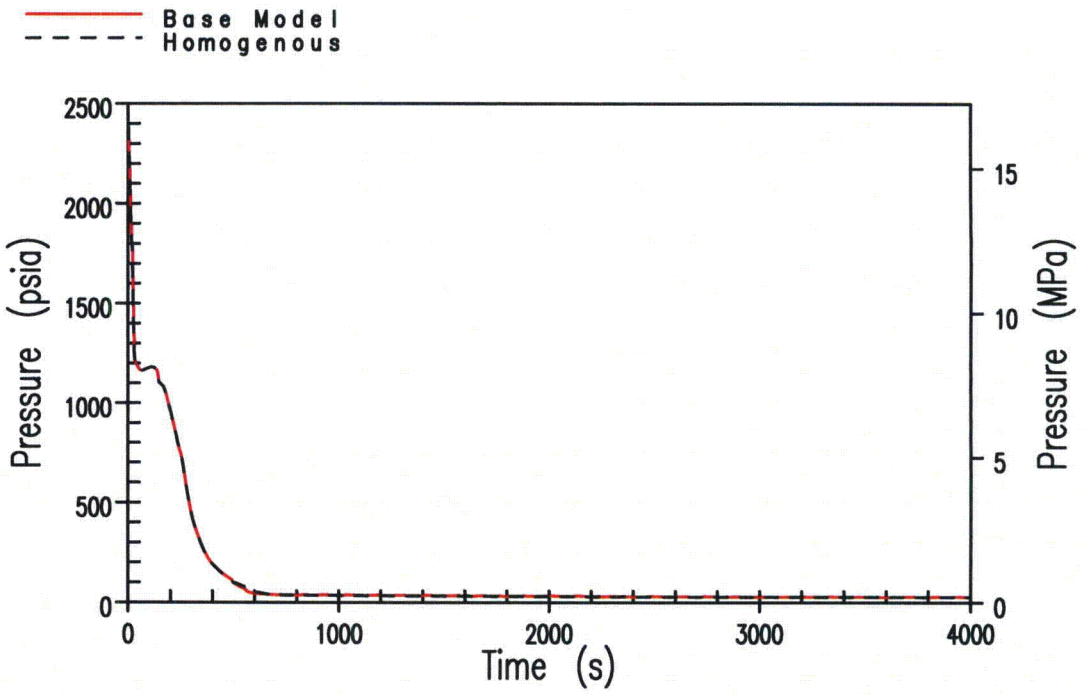

Figure 15.6.5.4B-79(a)

DEDVI Entrainment – Downcomer Pressure Comparison

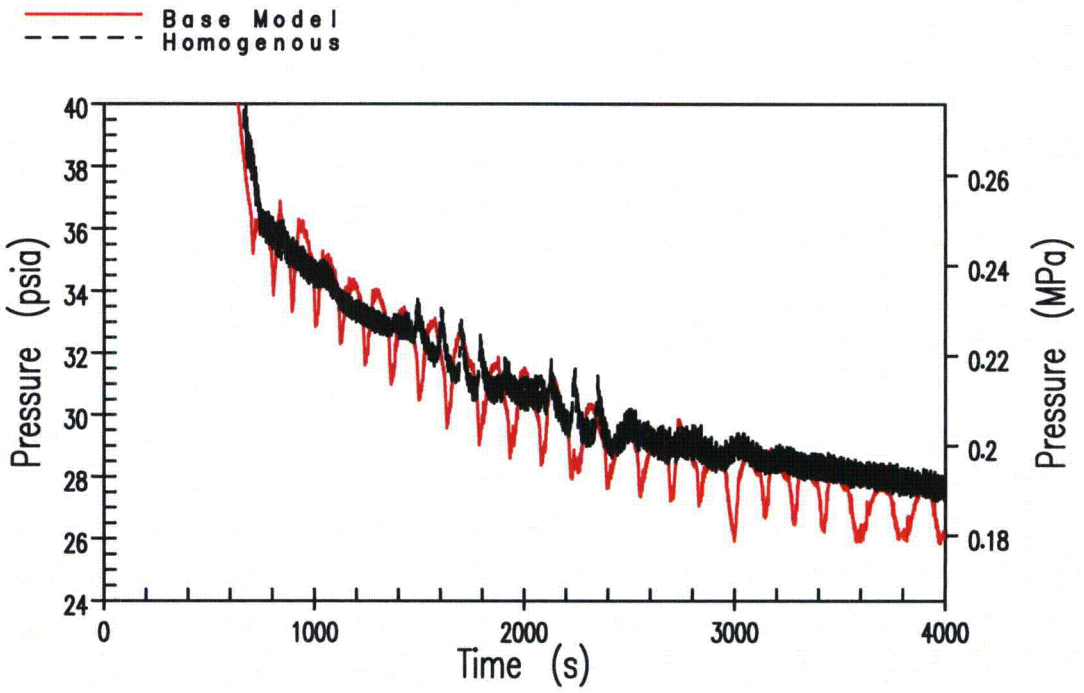

Figure 15.6.5.4B-79(b)

DEDVI Entrainment – Downcomer Pressure Comparison (Zoomed)

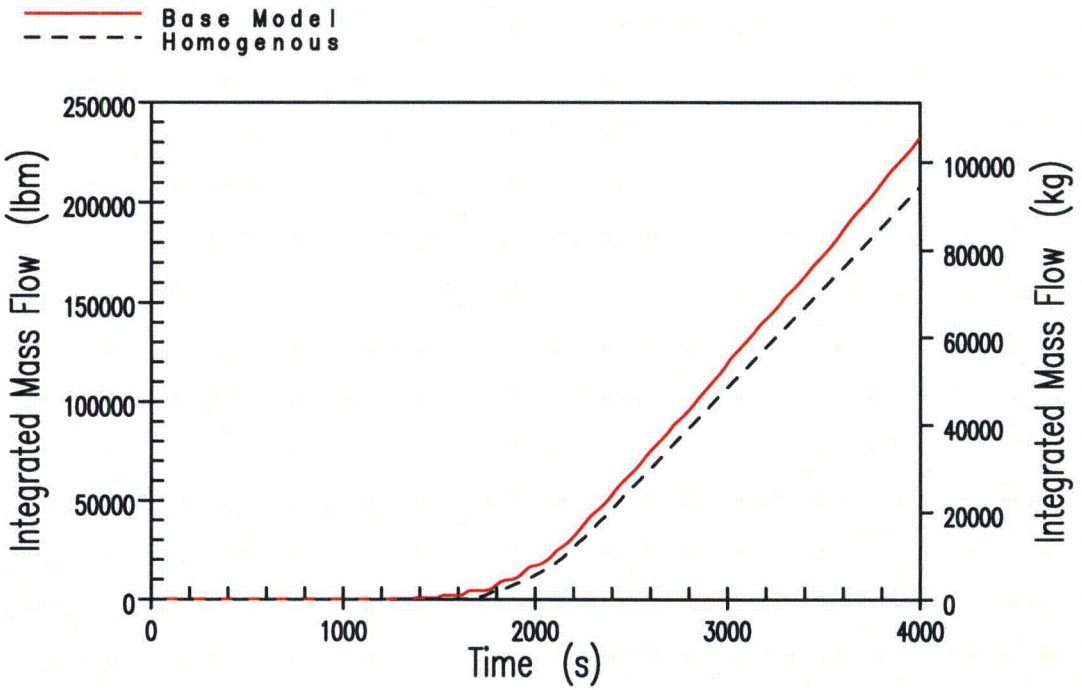

Figure 15.6.5.4B-80

DEDVI Entrainment – Intact IRWST Injection Flow

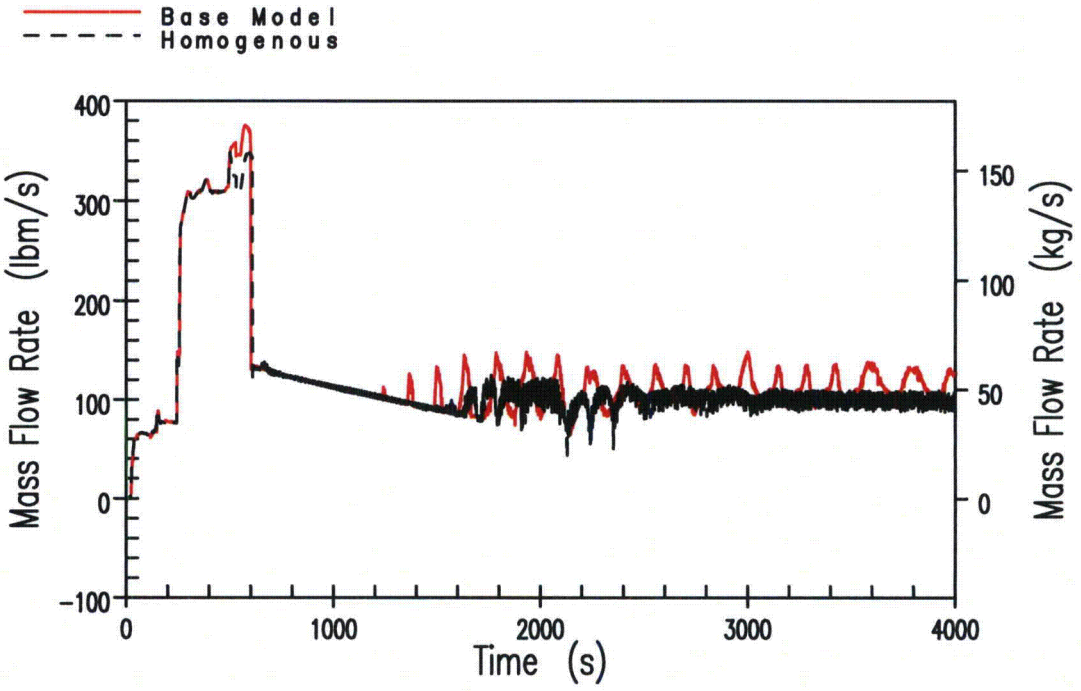

Figure 15.6.5.4B-81

DEDVI Entrainment – Intact DVI Line Injection Flow

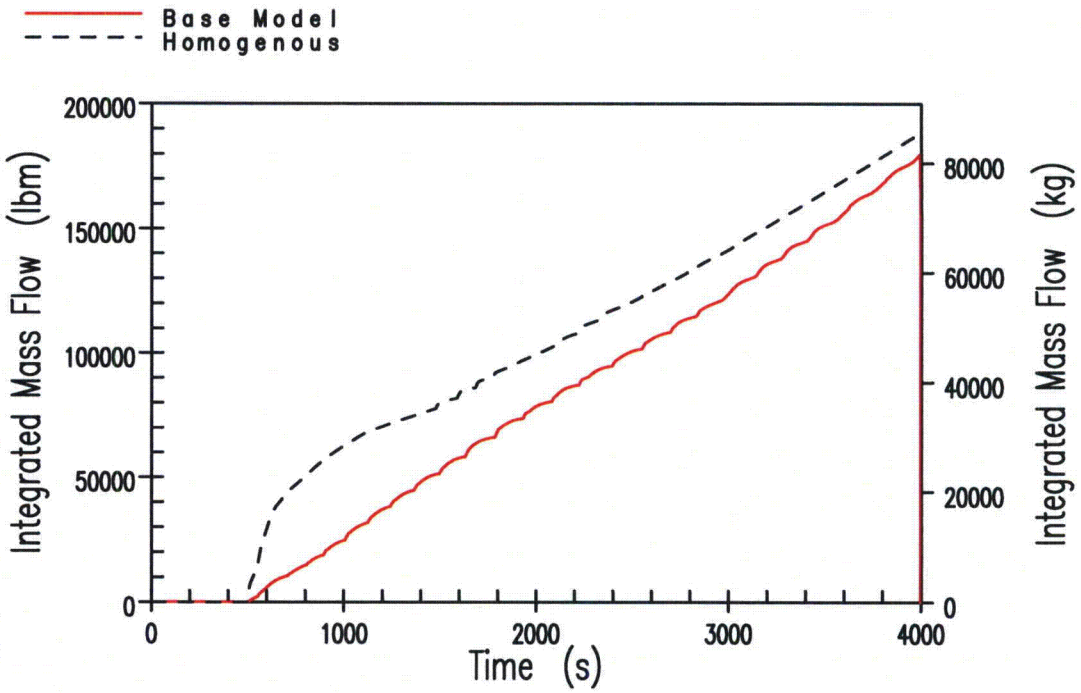

Figure 15.6.5.4B-82

DEDVI Entrainment – ADS-4 Integrated Liquid Discharge Comparison

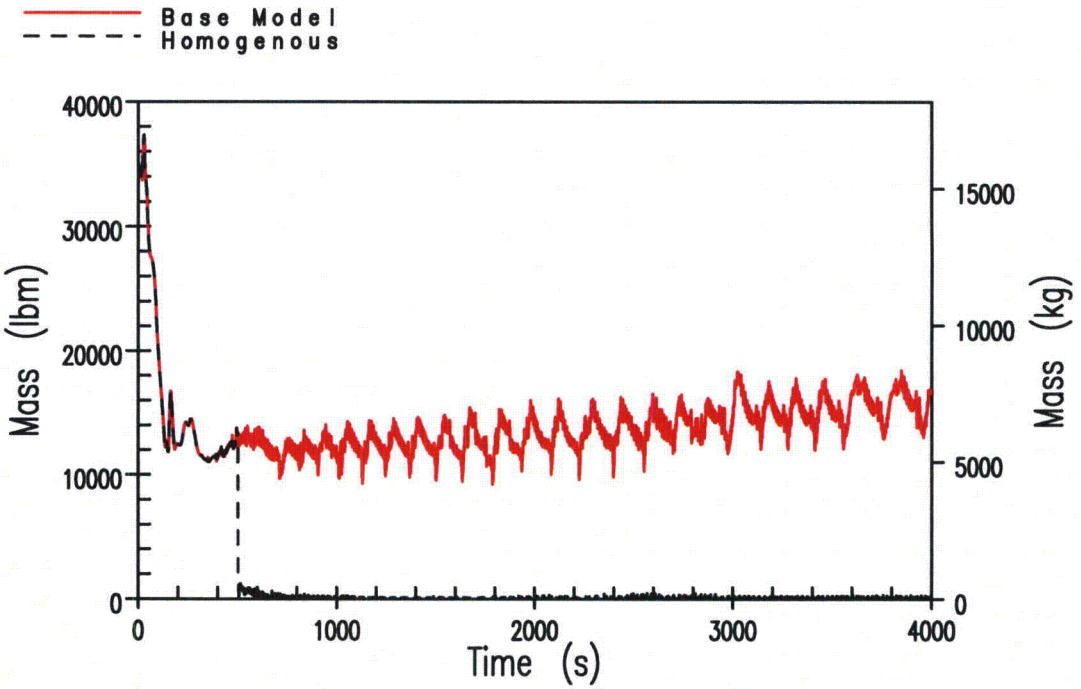

Figure 15.6.5.4B-83

DEDVI Entrainment – Upper Plenum Mixture Mass Comparison

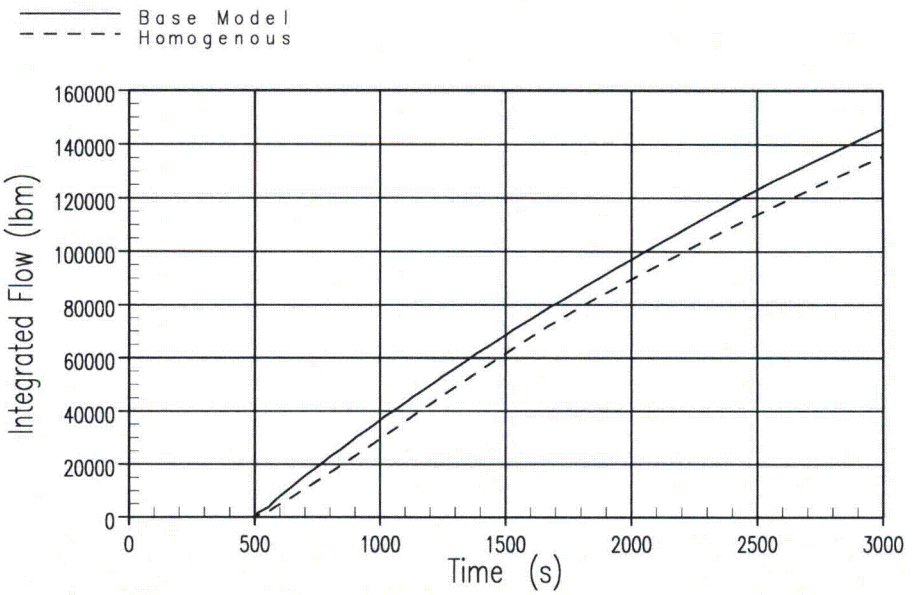

Figure 15.6.5.4B-84

DEDVI Entrainment – ADS-4 Integrated Vapor Discharge Comparison

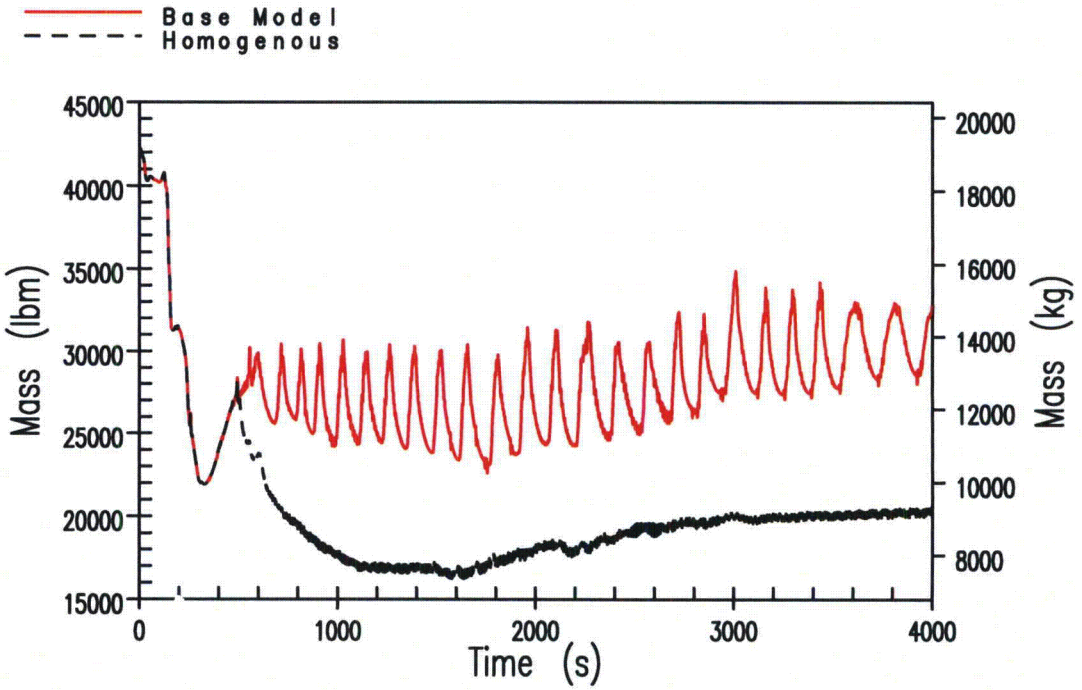

Figure 15.6.5.4B-85

DEDVI Entrainment – Downcomer Region Mass Comparison

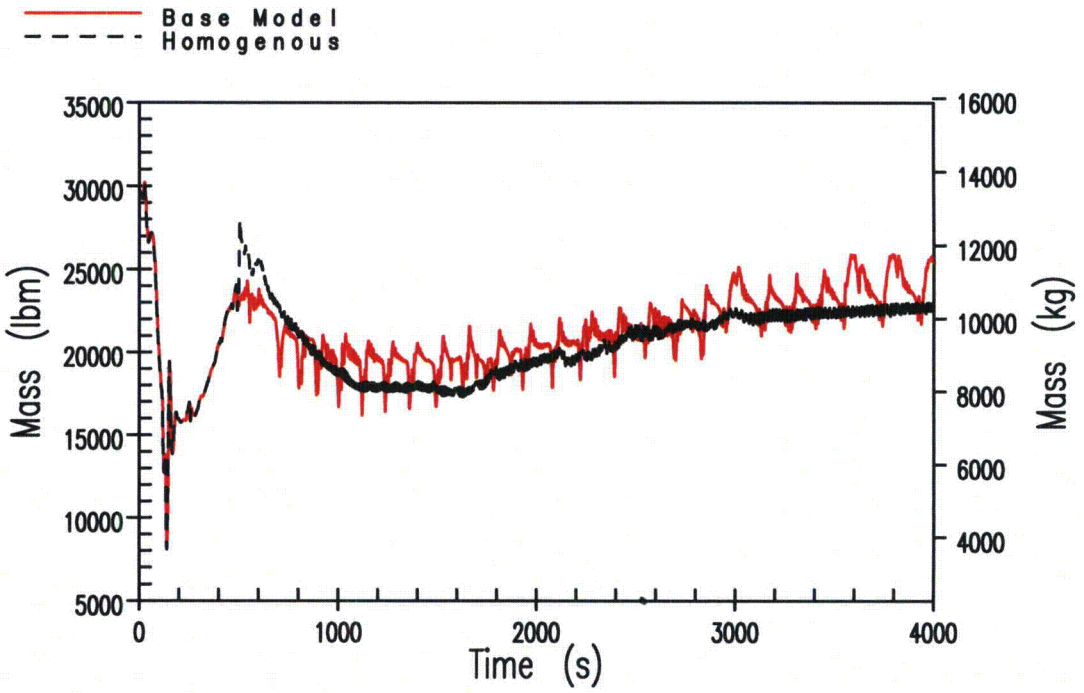

Figure 15.6.5.4B-86

DEDVI Entrainment – Core Region Mass Comparison

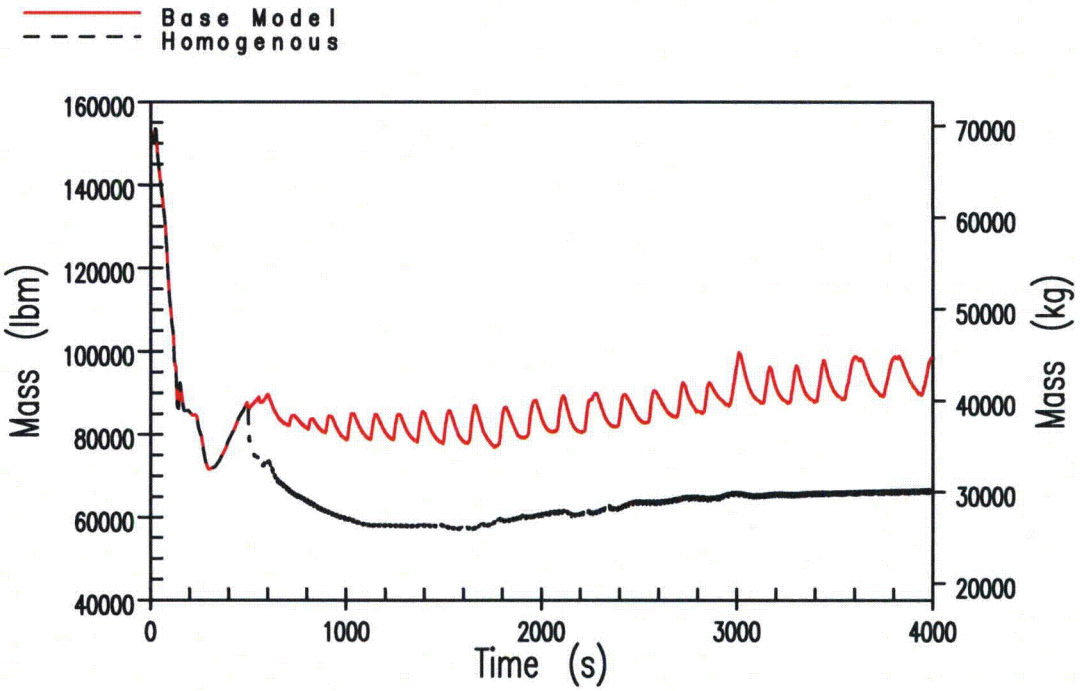

Figure 15.6.5.4B-87

DEDVI Entrainment – Vessel Mixture Mass Comparison

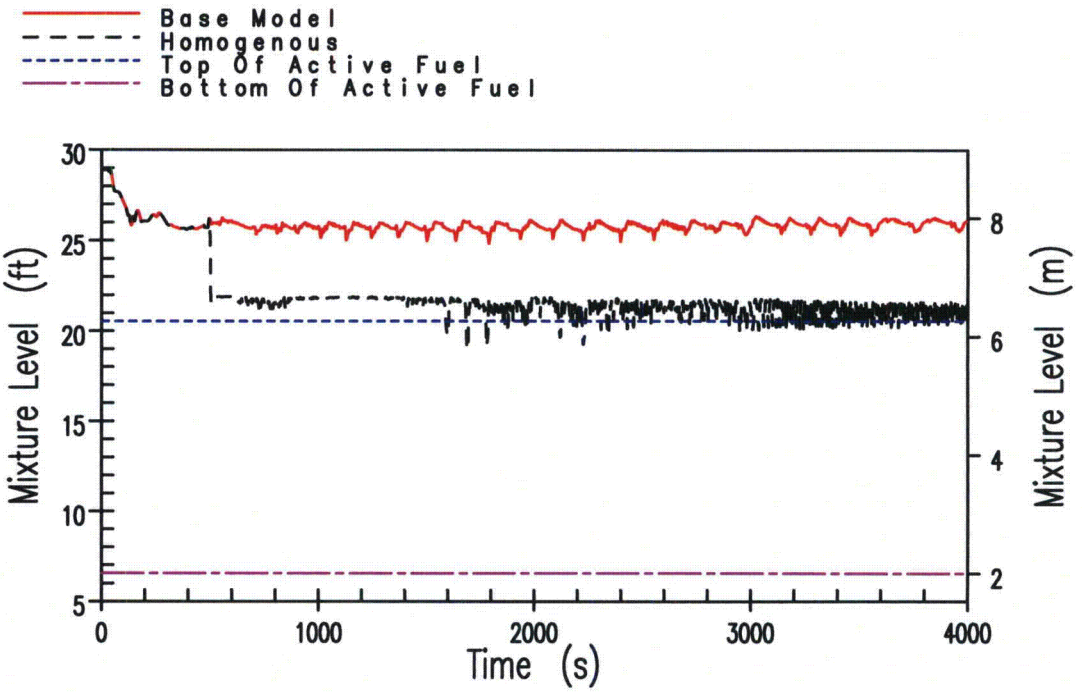

Figure 15.6.5.4B-88

DEDVI Entrainment – Core/Upper Plenum Mixture Level Comparison

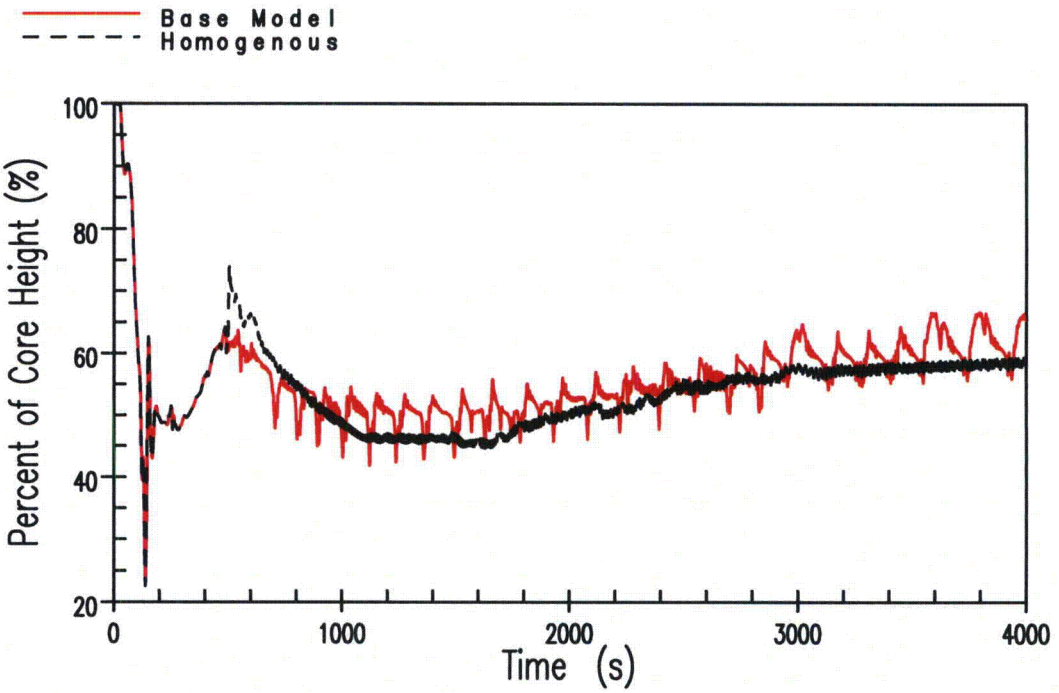

Figure 15.6.5.4B-89

DEDVI Entrainment – Core Collapsed Liquid Level Comparison

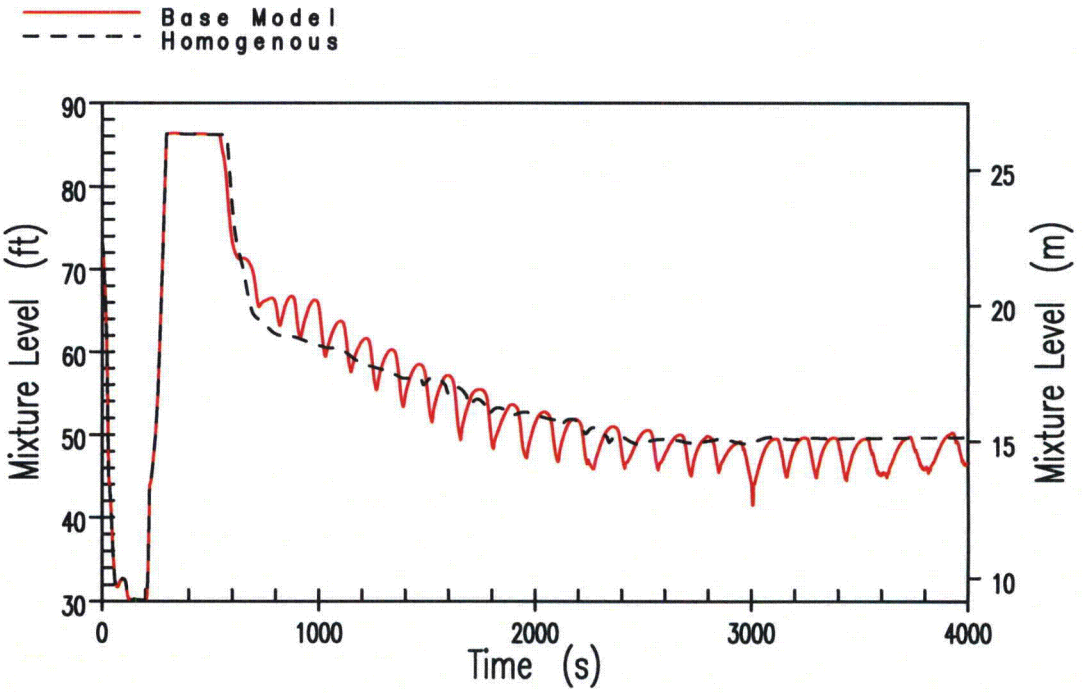

Figure 15.6.5.4B-90

DEDVI Entrainment – Pressurizer Mixture Level Comparison

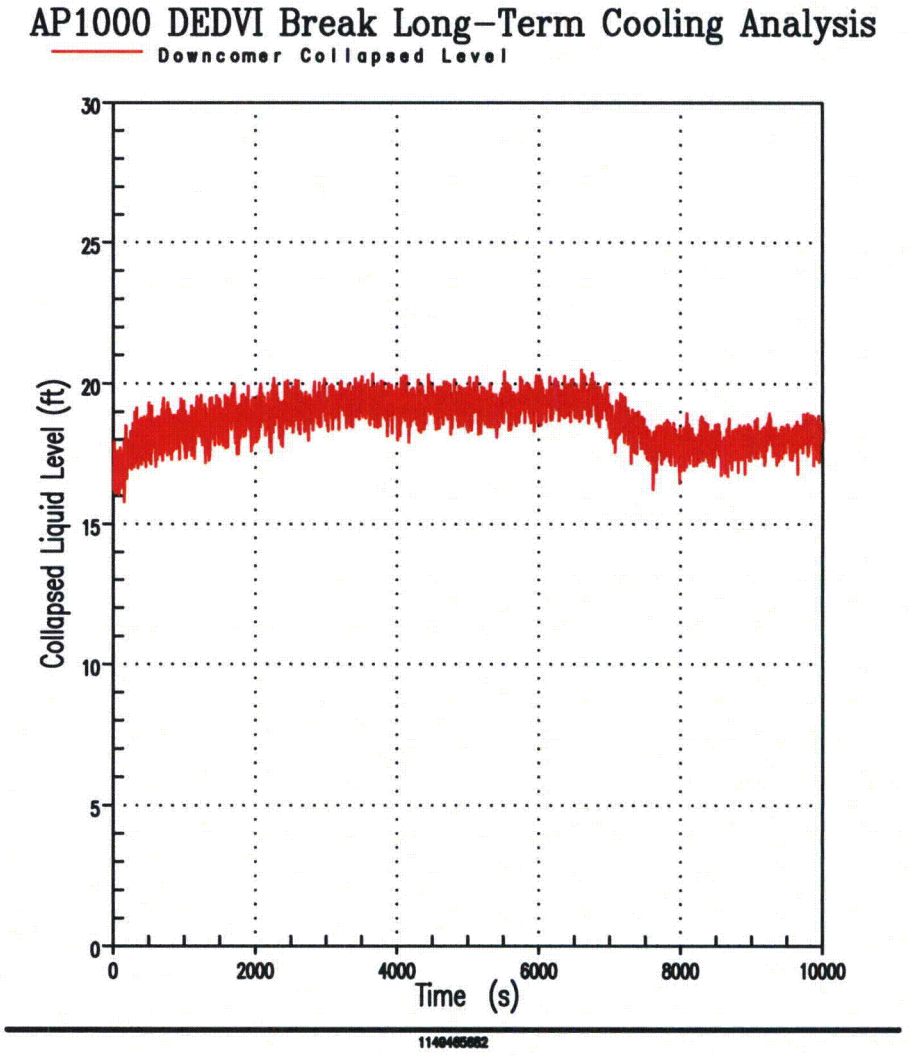

Figure 15.6.5.4C-1

Collapsed Level of Liquid in the Downcomer (DEDVI Case)

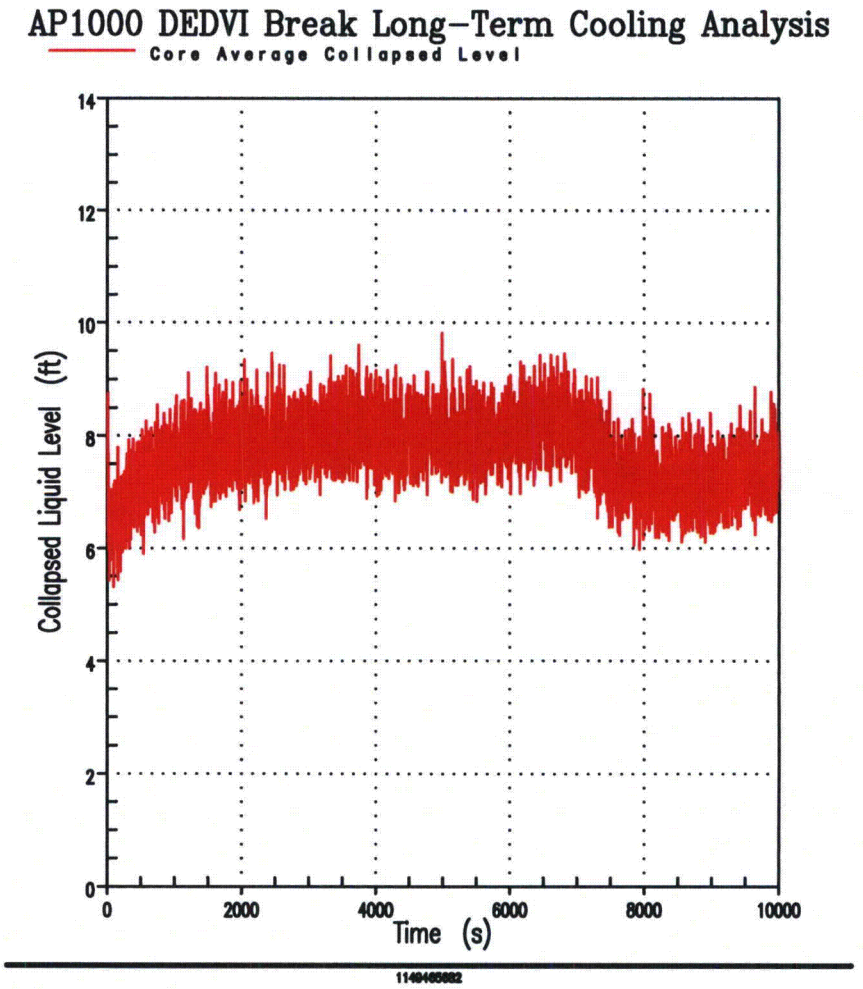

Figure 15.6.5.4C-2

Collapsed Level of Liquid over the Heated Length of the Fuel (DEDVI Case)

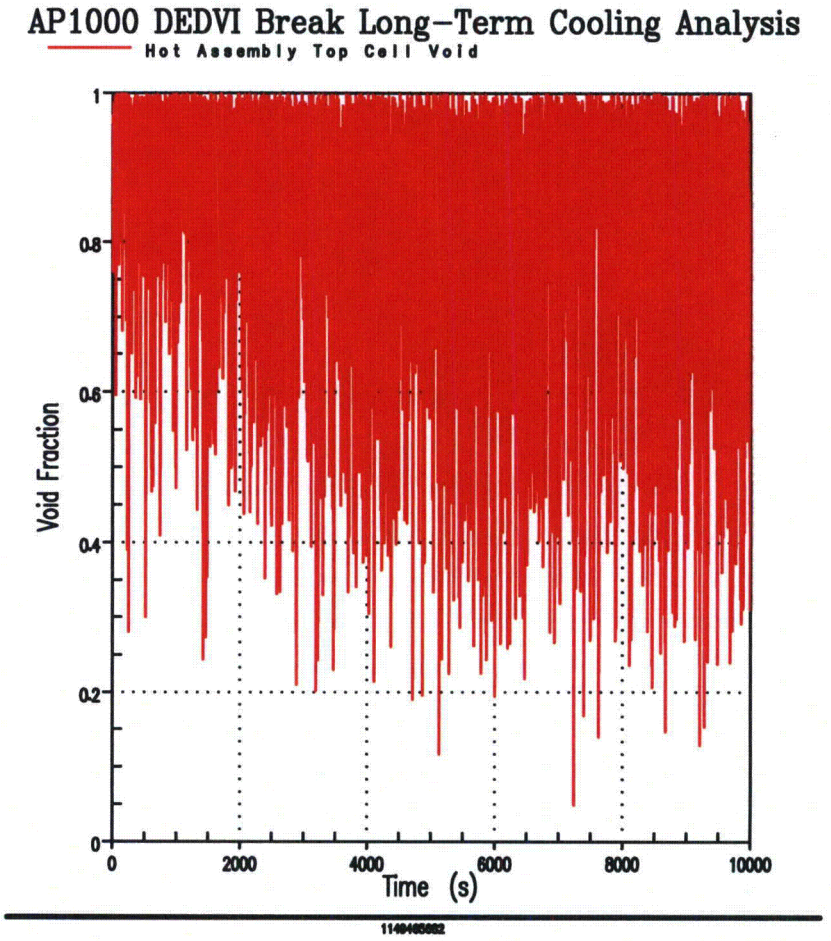

Figure 15.6.5.4C-3

Void Fraction in Core Hot Assembly Top Cell (DEDVI Case)

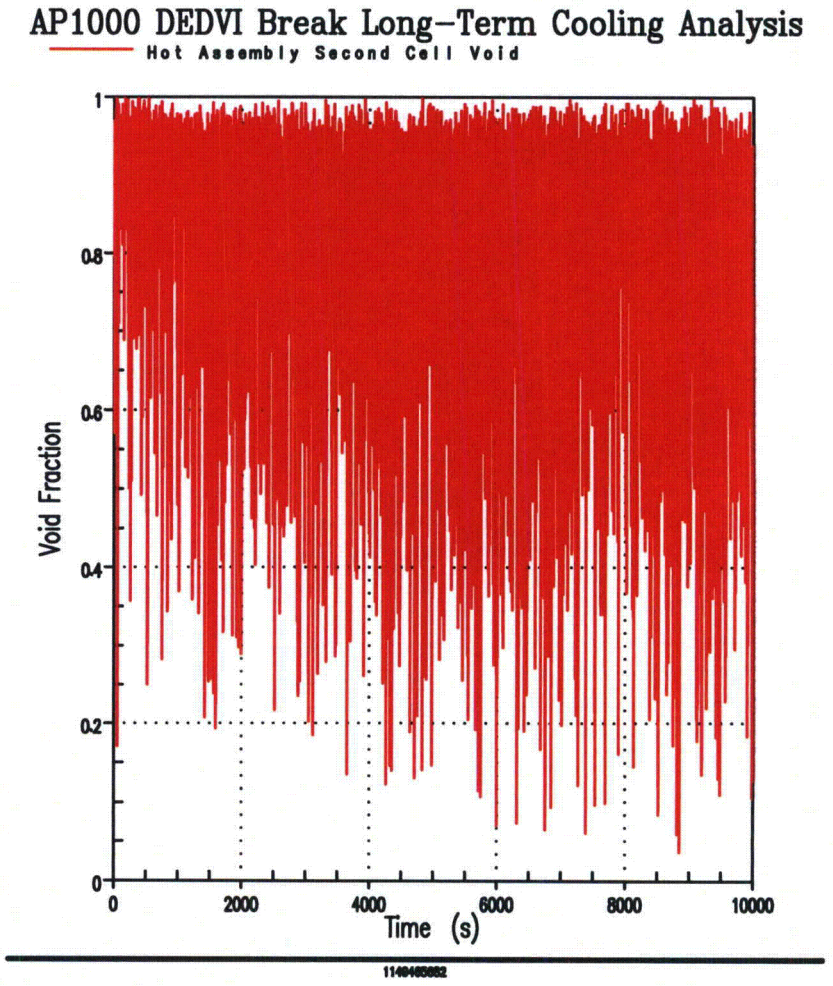

Figure 15.6.5.4C-4

Void Fraction in Core Hot Assembly Second from Top Cell (DEDVI Case)

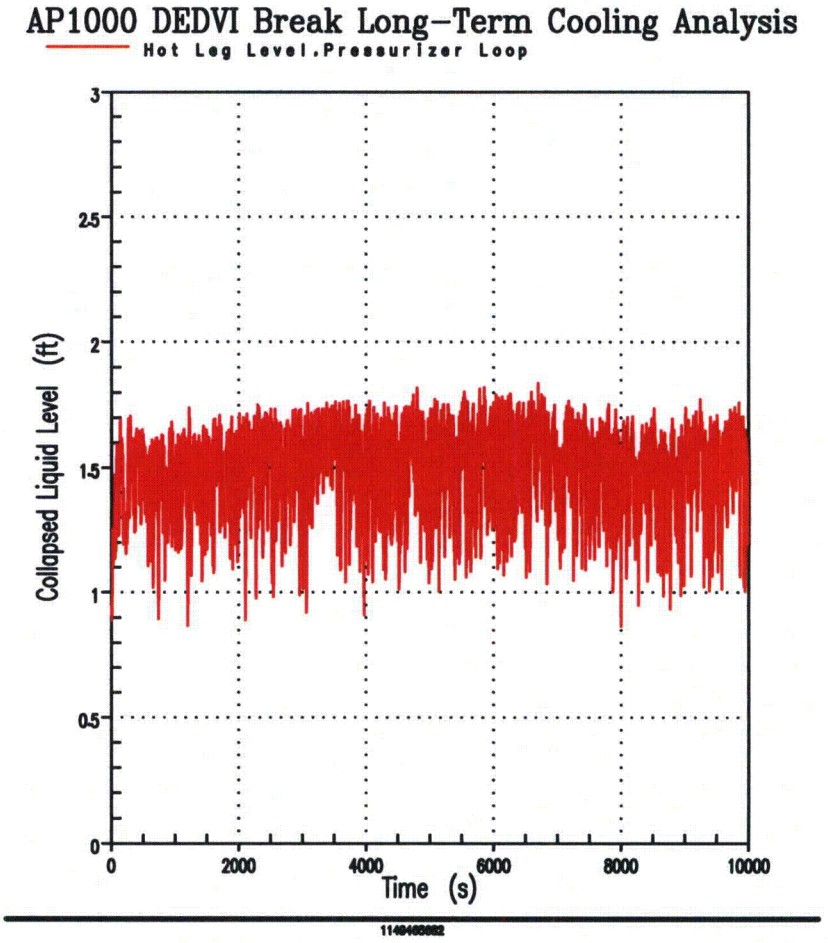

Figure 15.6.5.4C-5

Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (DEDVI Case)

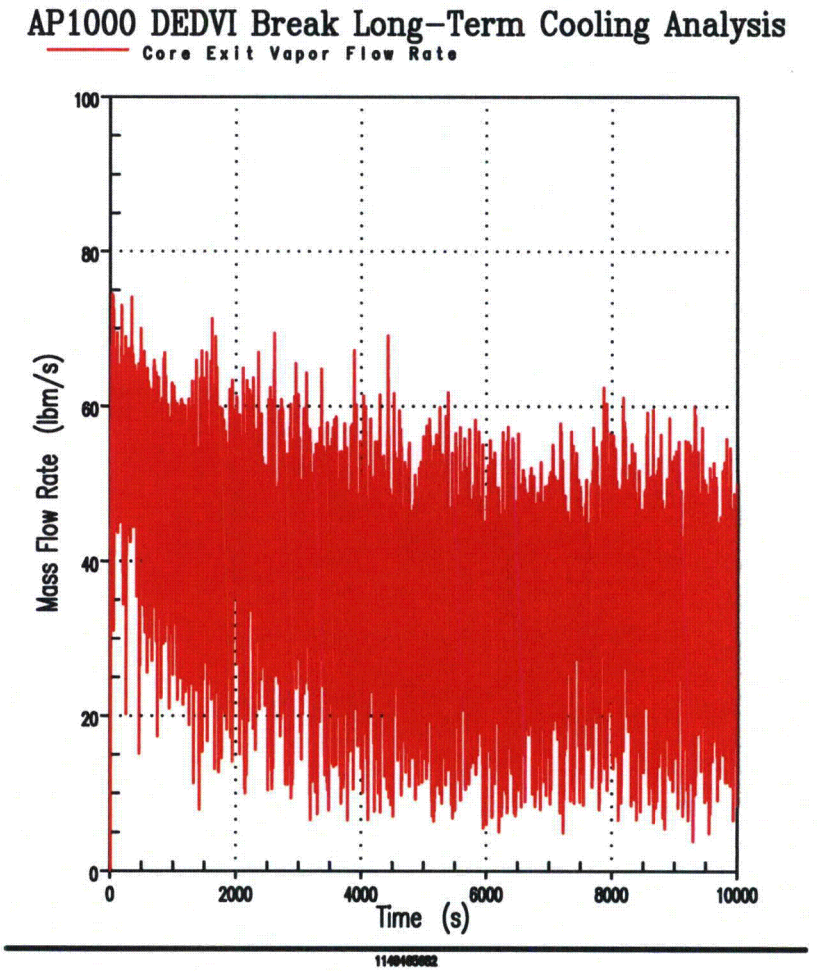

Figure 15.6.5.4C-6

Vapor Rate out of the Core (DEDVI Case)

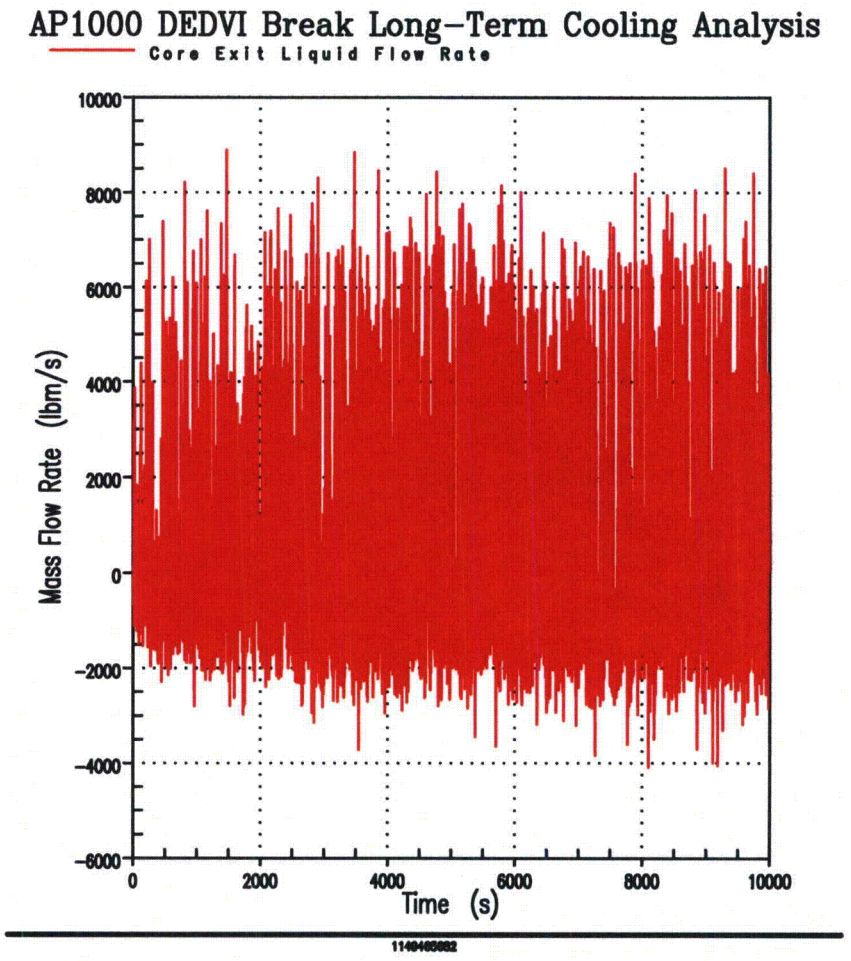

Figure 15.6.5.4C-7

Liquid Flow Rate out of the Core (DEDVI Case)

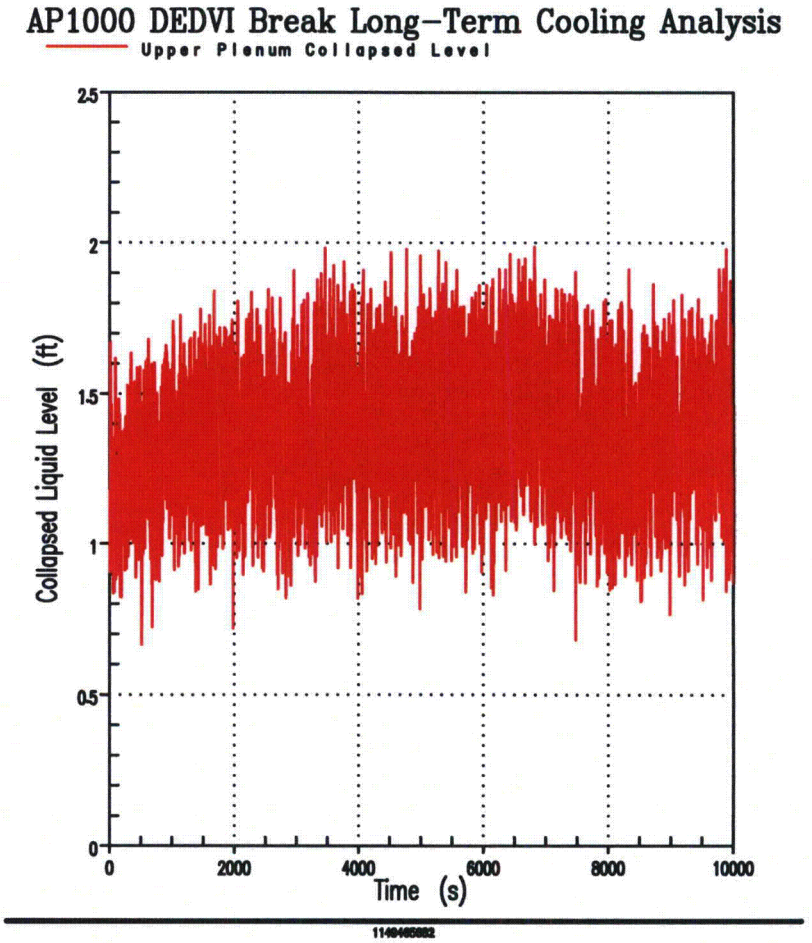

Figure 15.6.5.4C-8

Collapsed Liquid Level in the Upper Plenum (DEDVI Case)

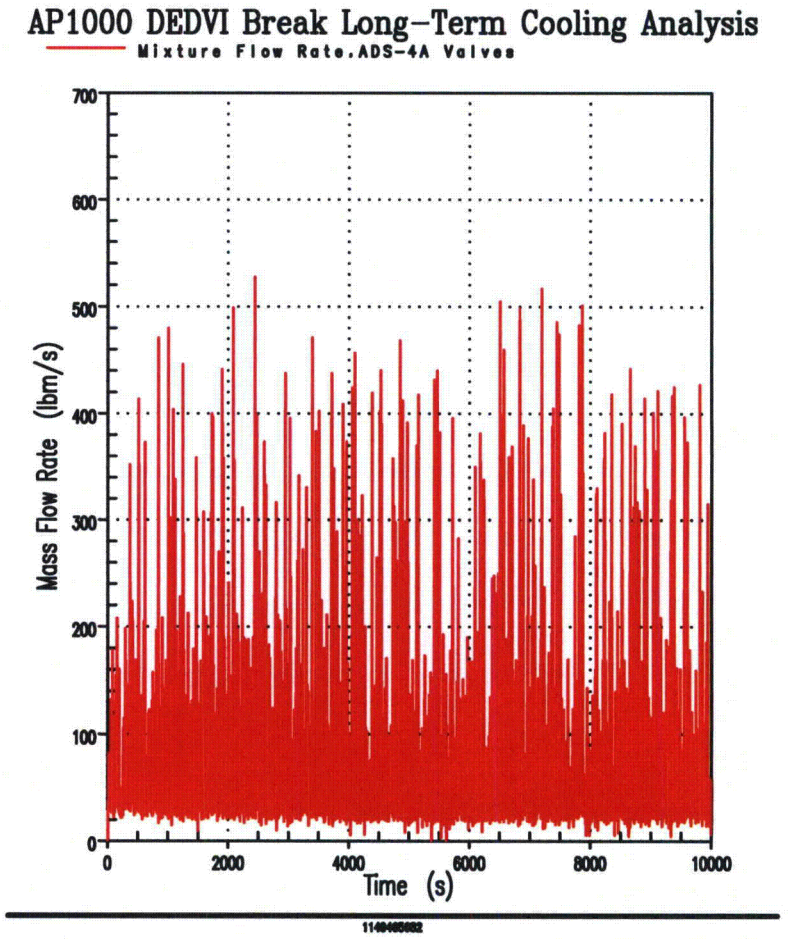

Figure 15.6.5.4C-9

Mixture Flow Rate Through ADS Stage 4A Valves (DEDVI Case)

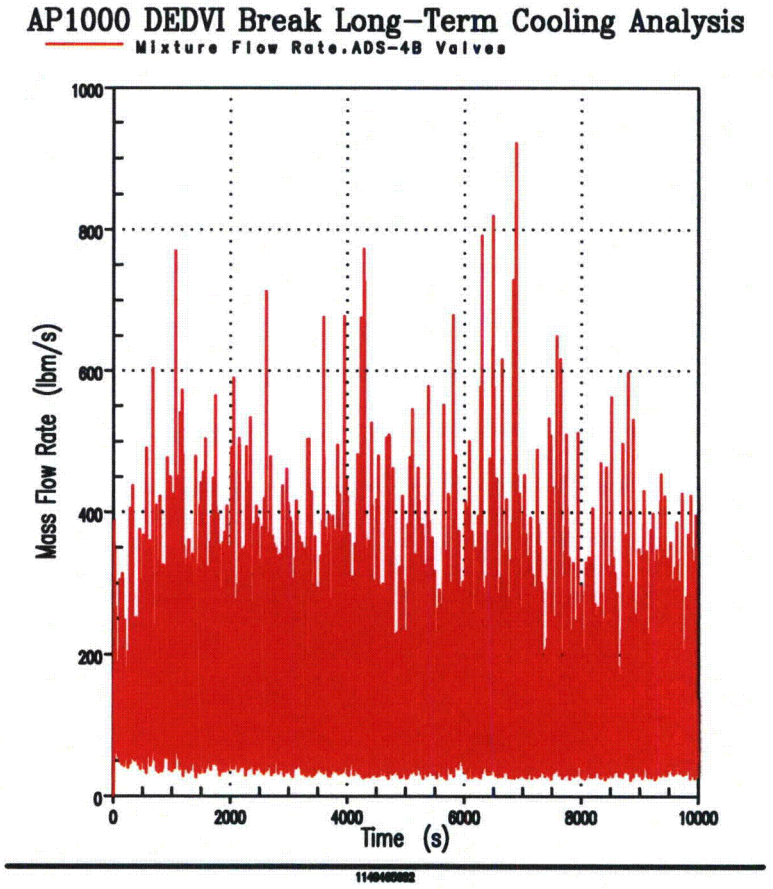

Figure 15.6.5.4C-10

Mixture Flow Rate Through ADS Stage 4B Valves (DEDVI Case)

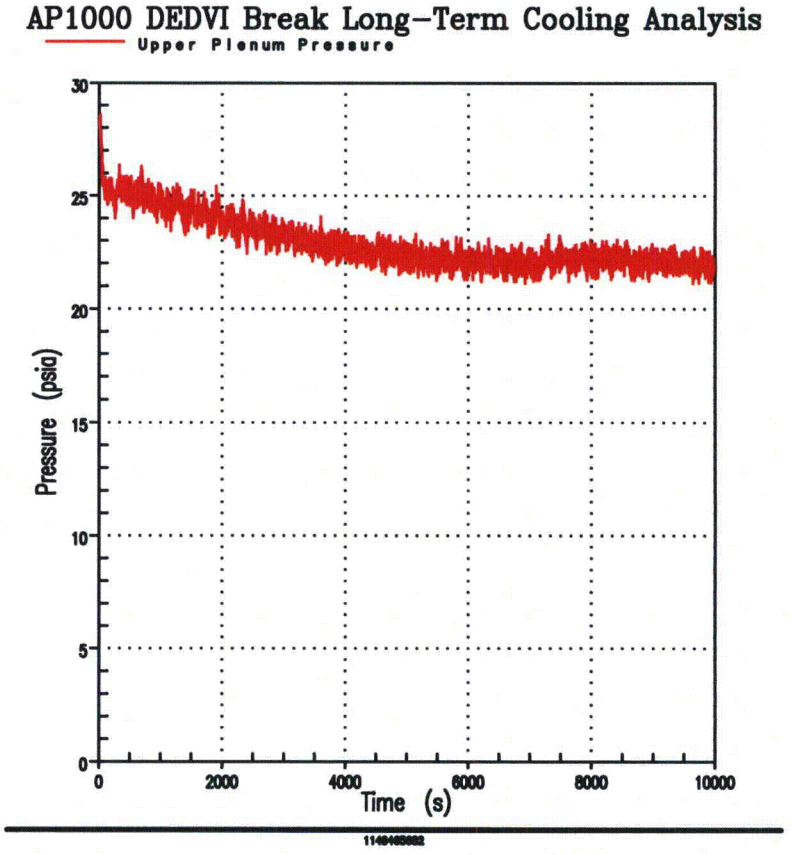

Figure 15.6.5.4C-11

Upper Plenum Pressure (DEDVI Case)

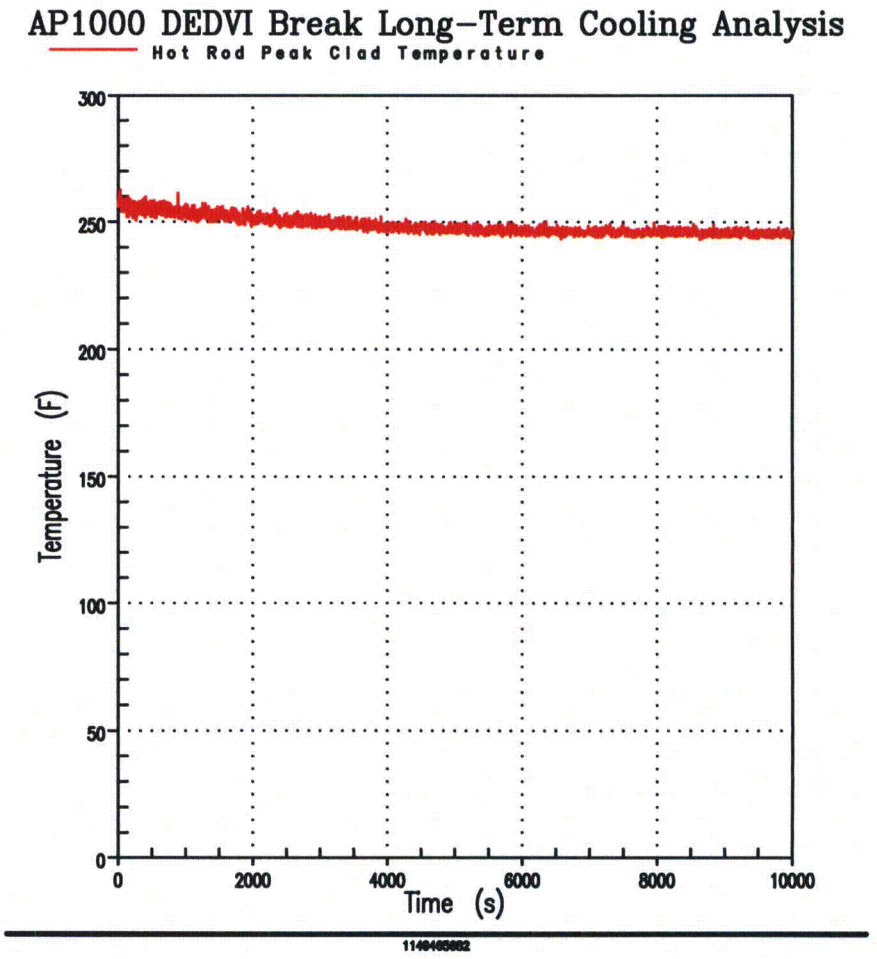

Figure 15.6.5.4C-12

Peak Cladding Temperature (DEDVI Case)

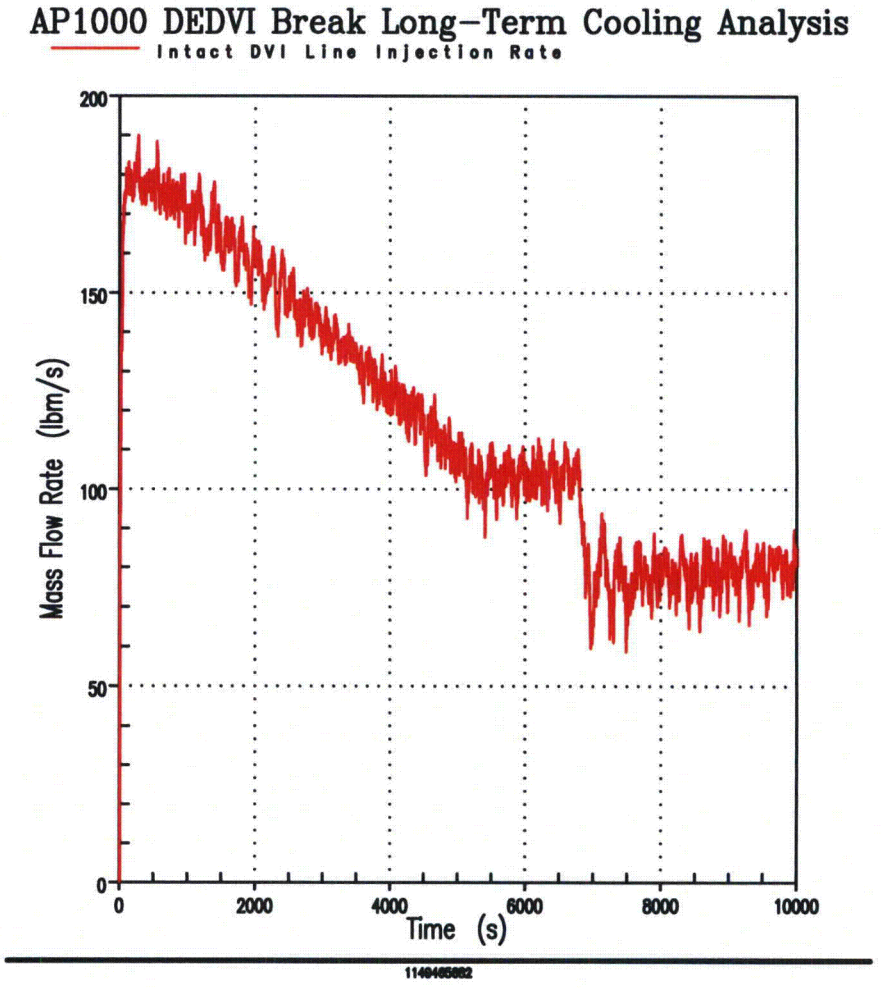

Figure 15.6.5.4C-13

DVI–A Mixture Flow Rate
(DEDVI Case)

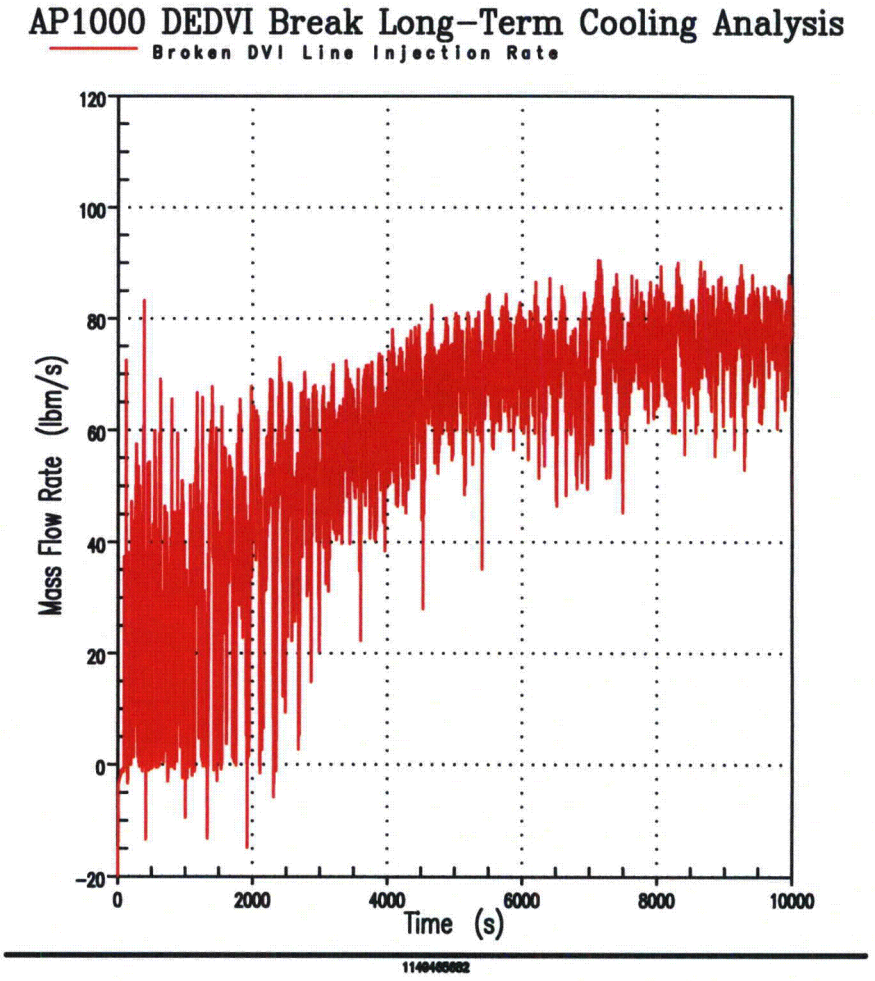

Figure 15.6.5.4C-14

DVI–B Mixture Flow Rate (DEDVI Case)

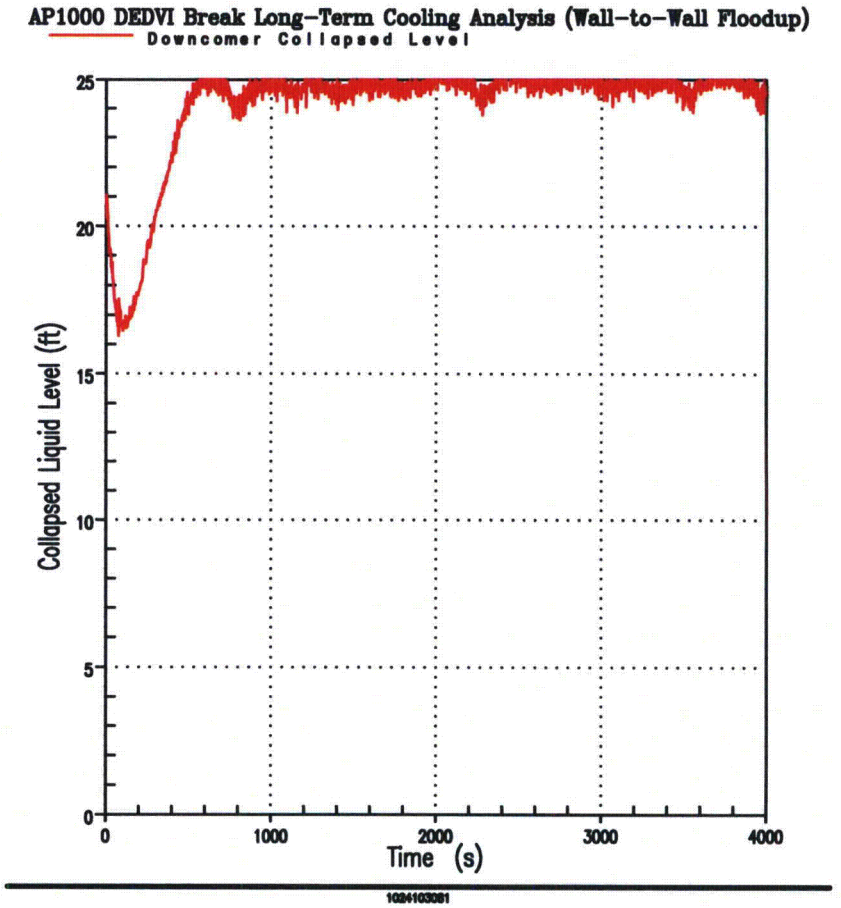

Figure 15.6.5.4C-15

Collapsed Level of Liquid in the Downcomer (Wall-to-Wall Floodup Case) – 14.7 psia

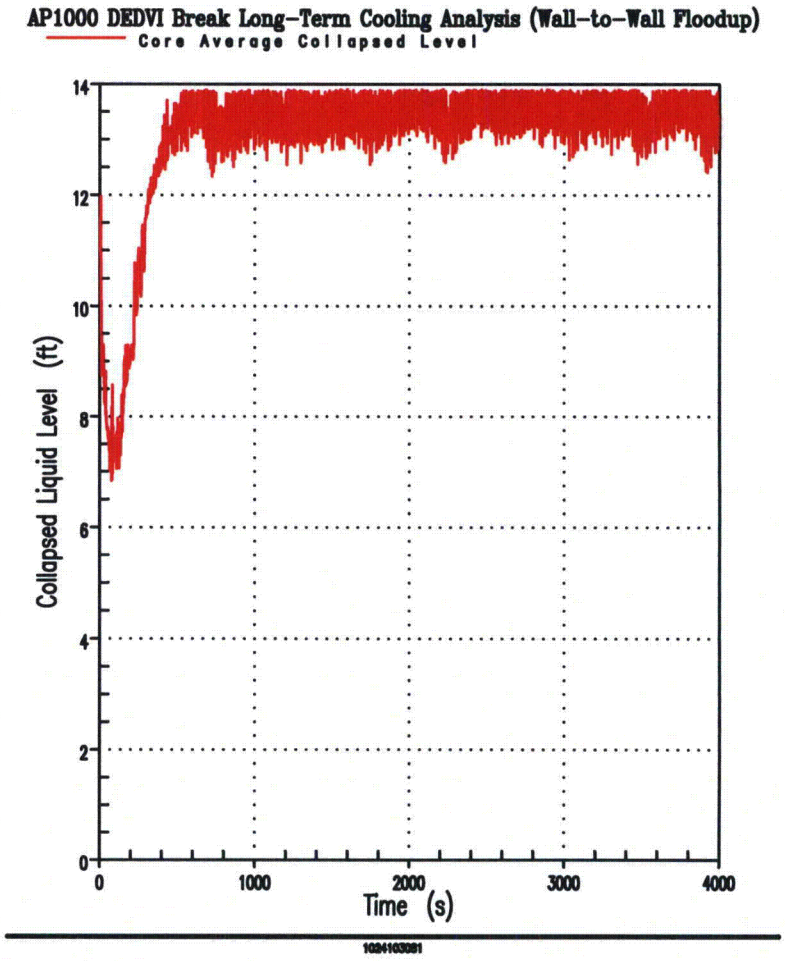

Figure 15.6.5.4C-16

Collapsed Level of Liquid Over the Heated Length of the Fuel (Wall-to-Wall Floodup Case) – 14.7 psia

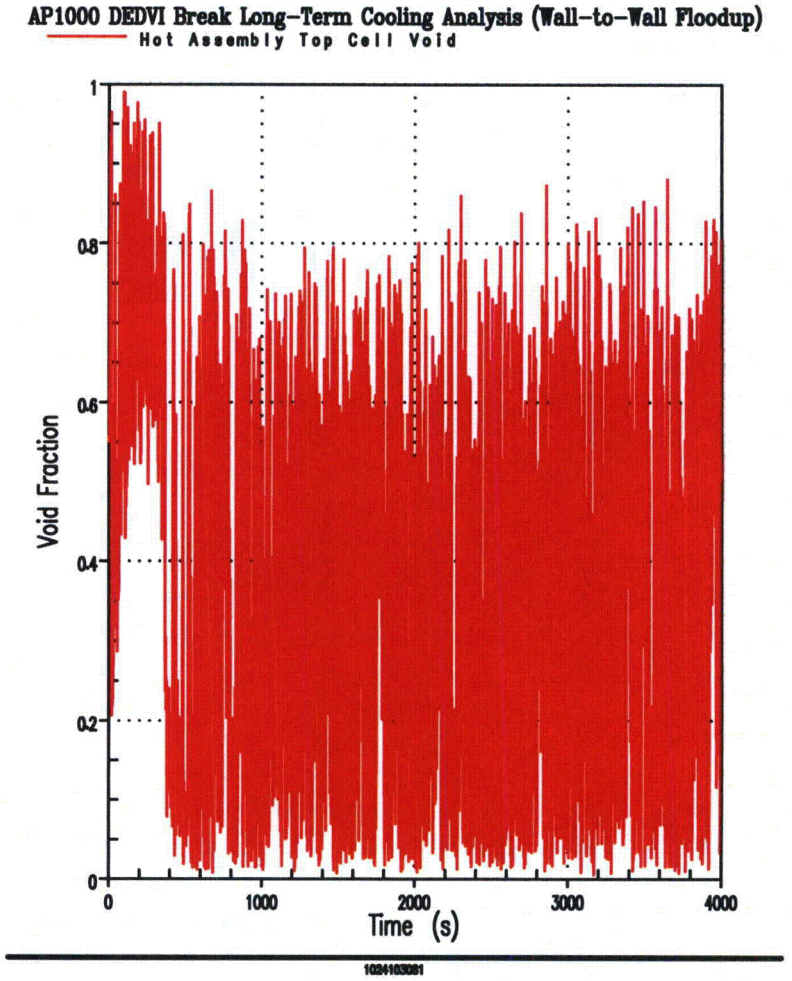

Figure 15.6.5.4C-17

Void Fraction in Core Hot Assembly Top Cell (Wall-to-Wall Floodup Case) – 14.7 psia

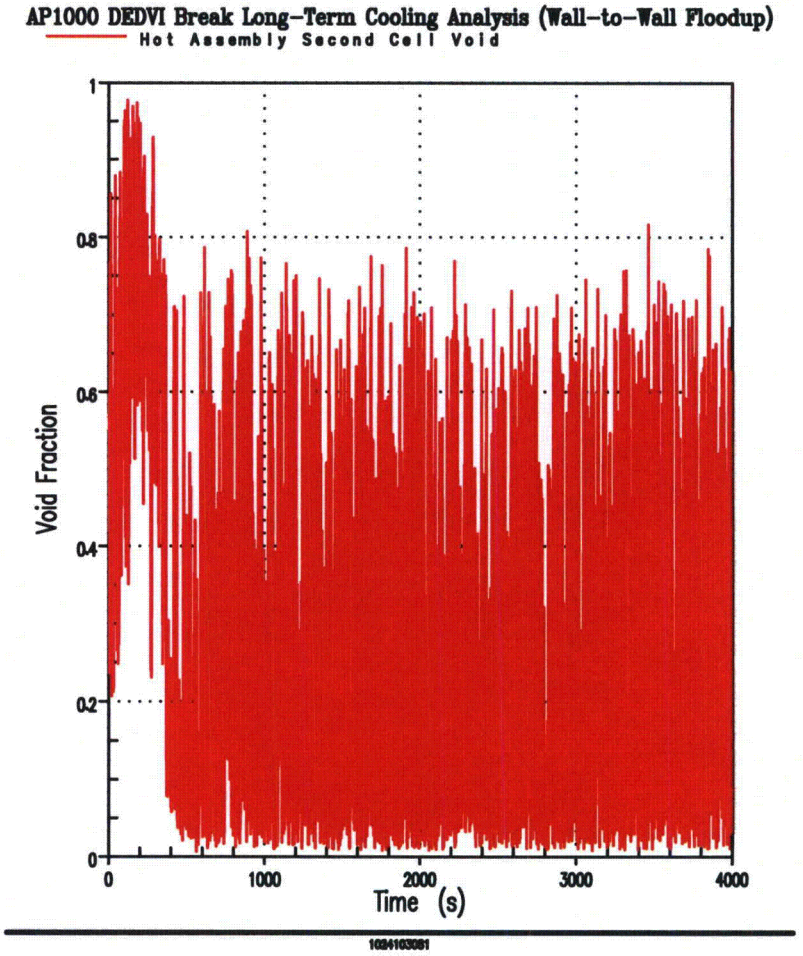

Figure 15.6.5.4C-18

Void Fraction in Core Hot Assembly Second from Top Cell (Wall-to-Wall Floodup Case) – 14.7 psia

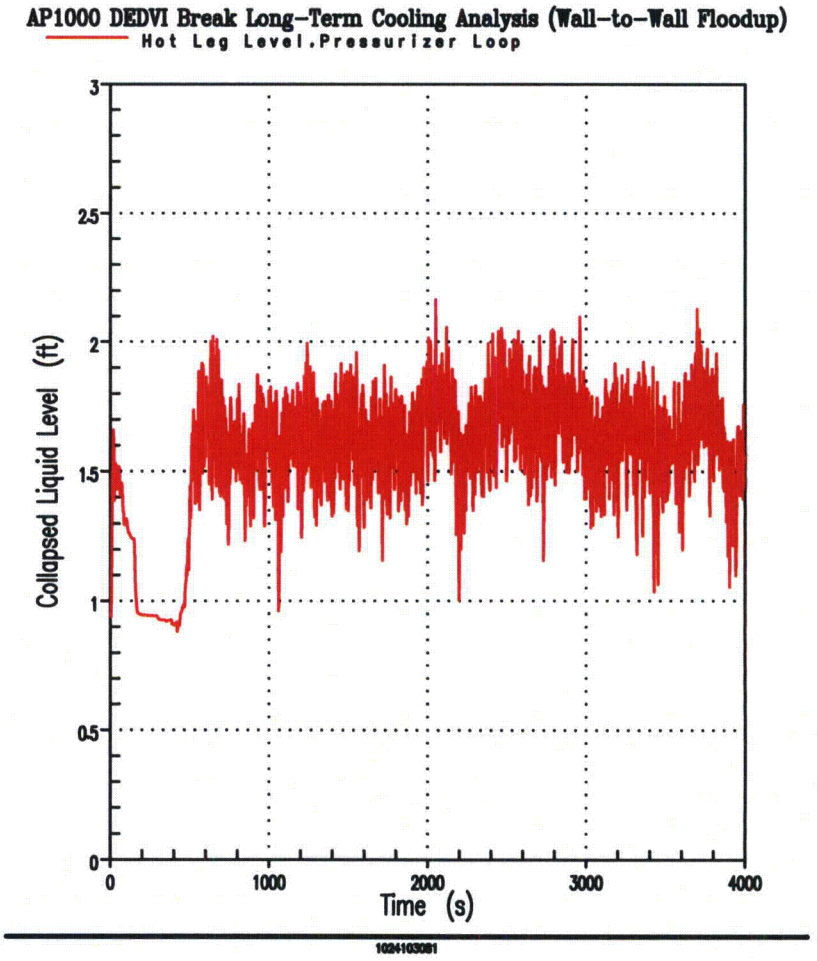

Figure 15.6.5.4C-19

Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (Wall-to-Wall Floodup Case) – 14.7 psia

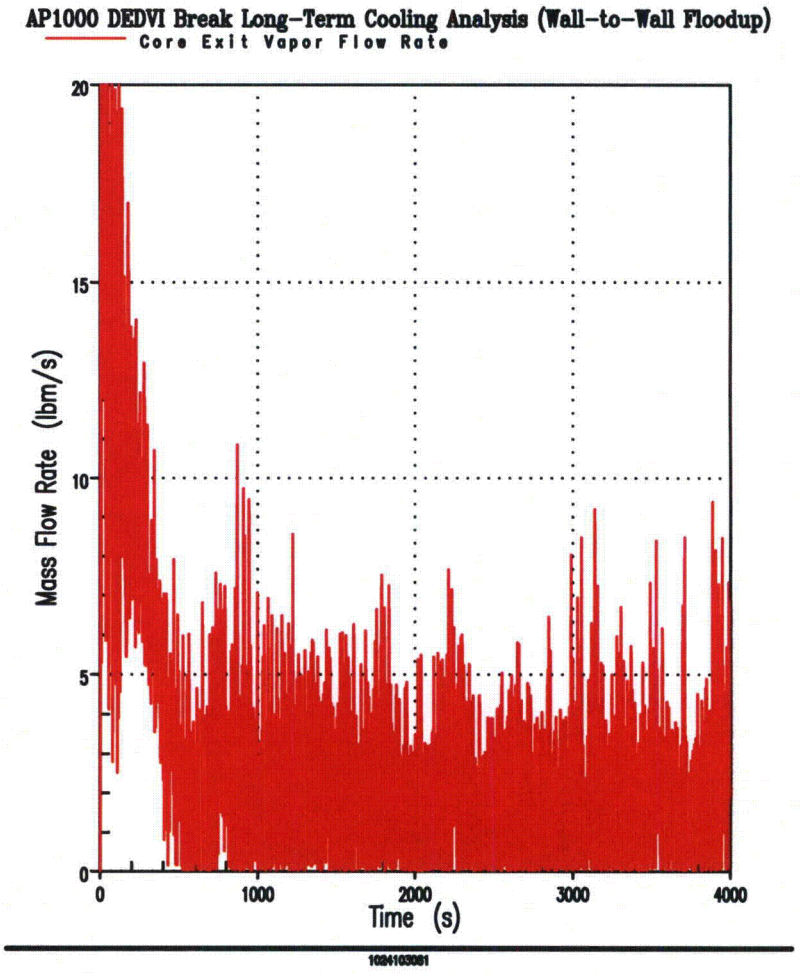

Figure 15.6.5.4C-20

Vapor Rate out of the Core
(Wall-to-Wall Floodup Case) – 14.7 psia

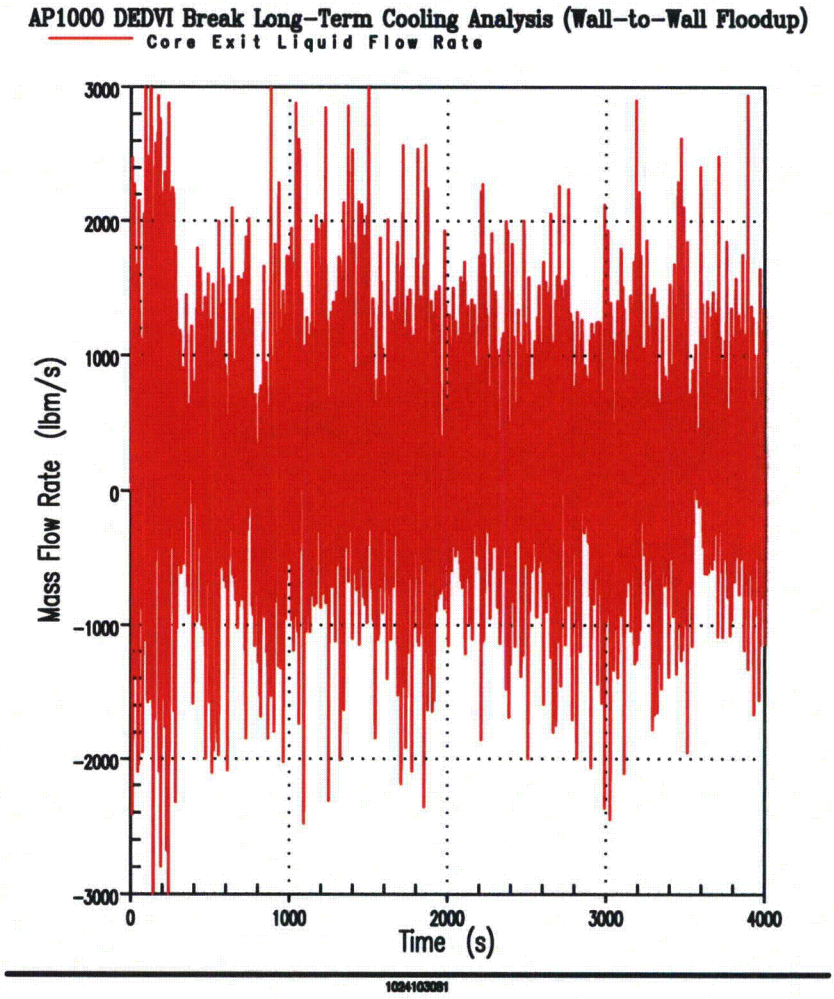

Figure 15.6.5.4C-21

Liquid Flow Rate out of the Core (Wall-to-Wall Floodup Case) – 14.7 psia

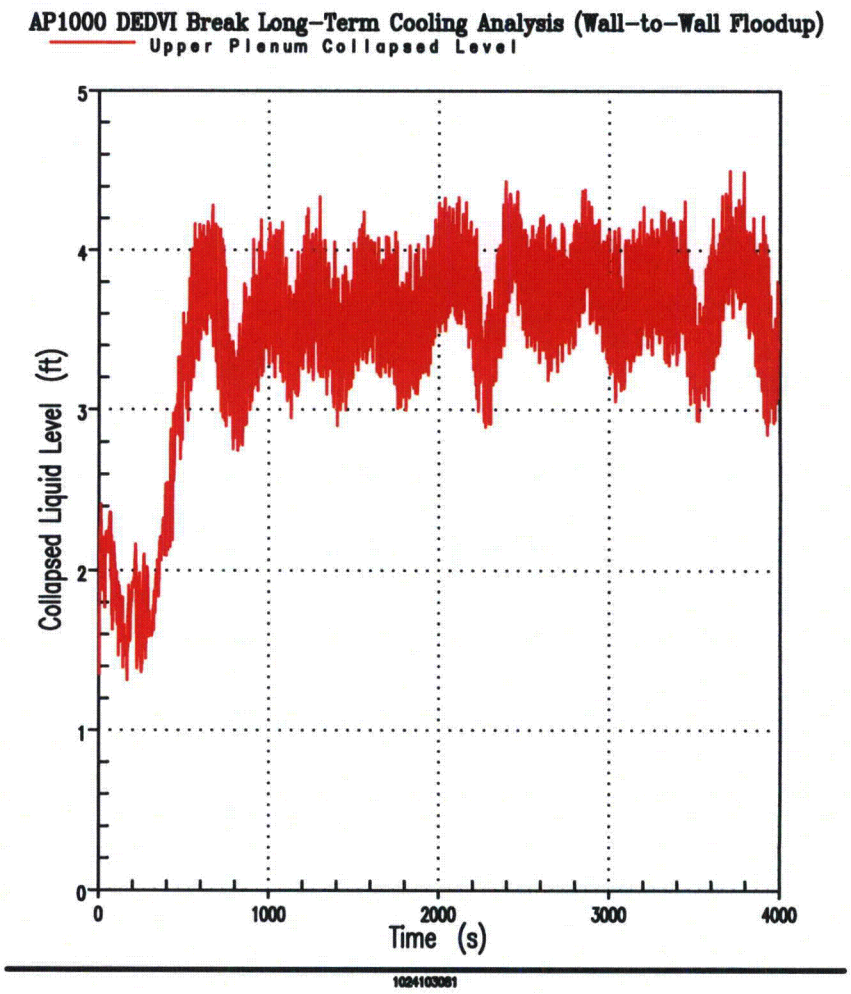

Figure 15.6.5.4C-22

Collapsed Liquid Level in the Upper Plenum (Wall-to-Wall Floodup Case) – 14.7 psia

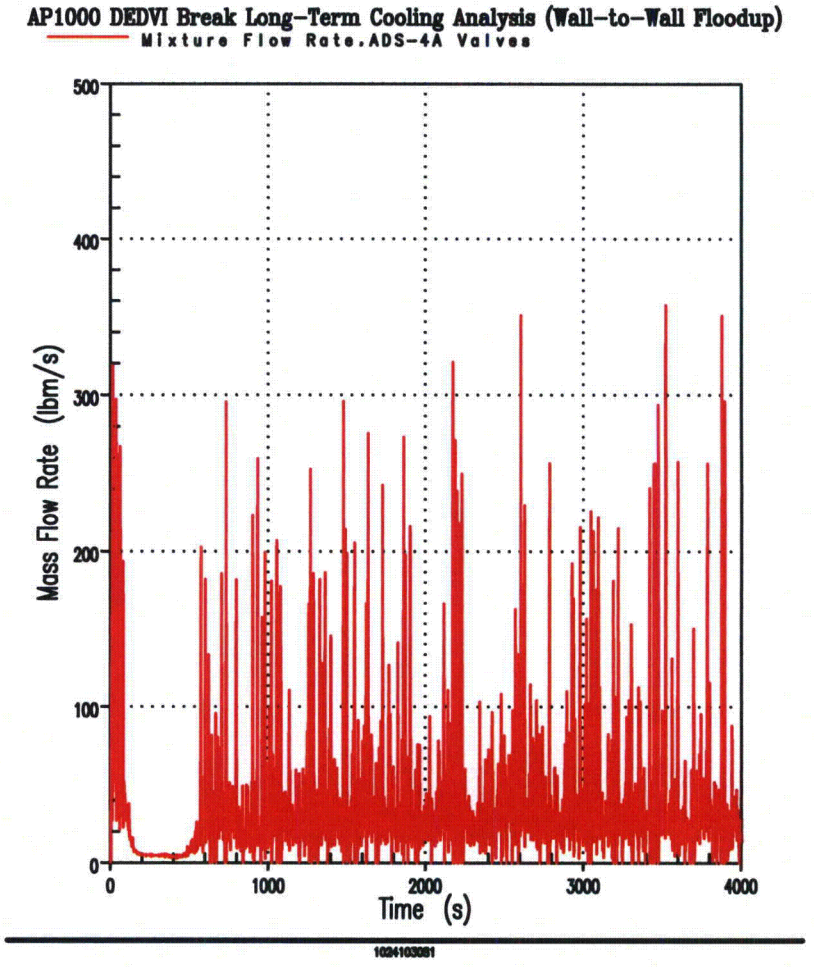

Figure 15.6.5.4C-23

Mixture Flow Rate Through ADS Stage 4A Valves (Wall-to-Wall Floodup Case) – 14.7 psia

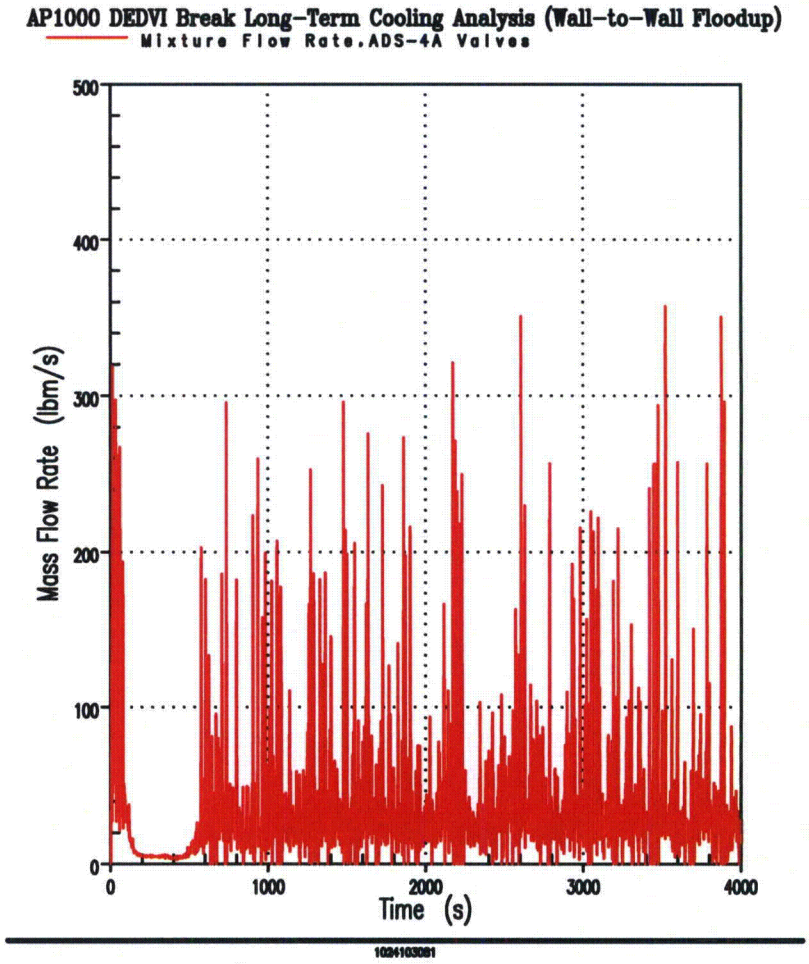

Figure 15.6.5.4C-24

Mixture Flow Rate Through ADS Stage 4B Valves (Wall-to-Wall Floodup Case) – 14.7 psia

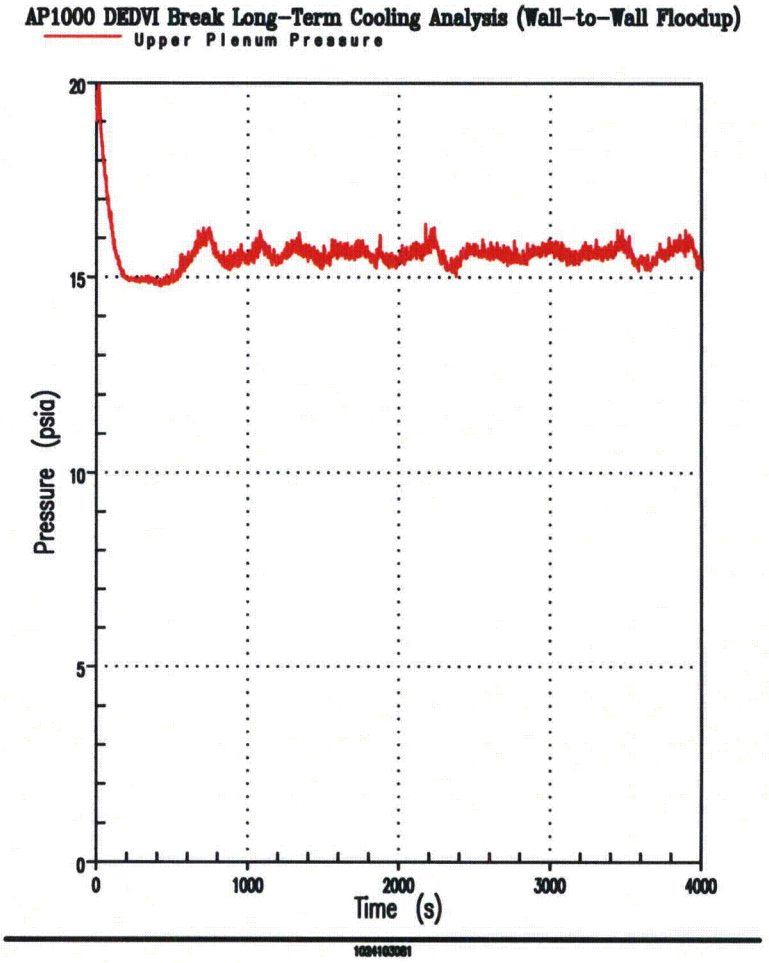

Figure 15.6.5.4C-25

Upper Plenum Pressure
(Wall-to-Wall Floodup Case) – 14.7 psia

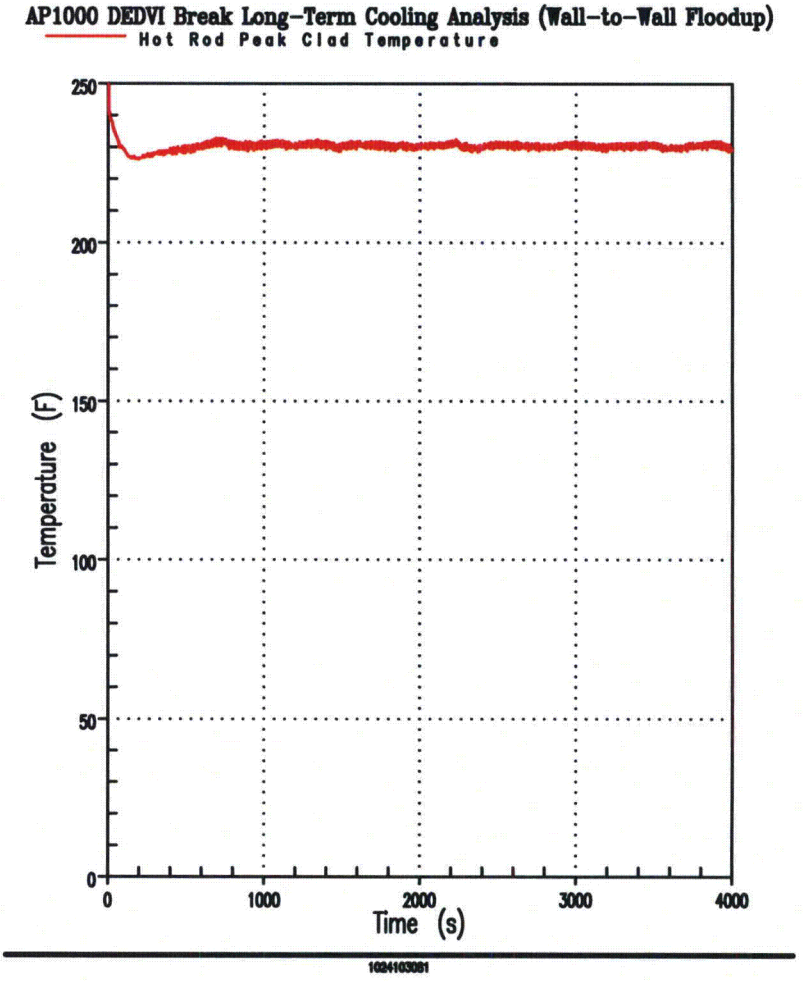

Figure 15.6.5.4C-26

Hot Rod Cladding Temperature Near Top of Core (Wall-to-Wall Floodup Case) – 14.7 psia

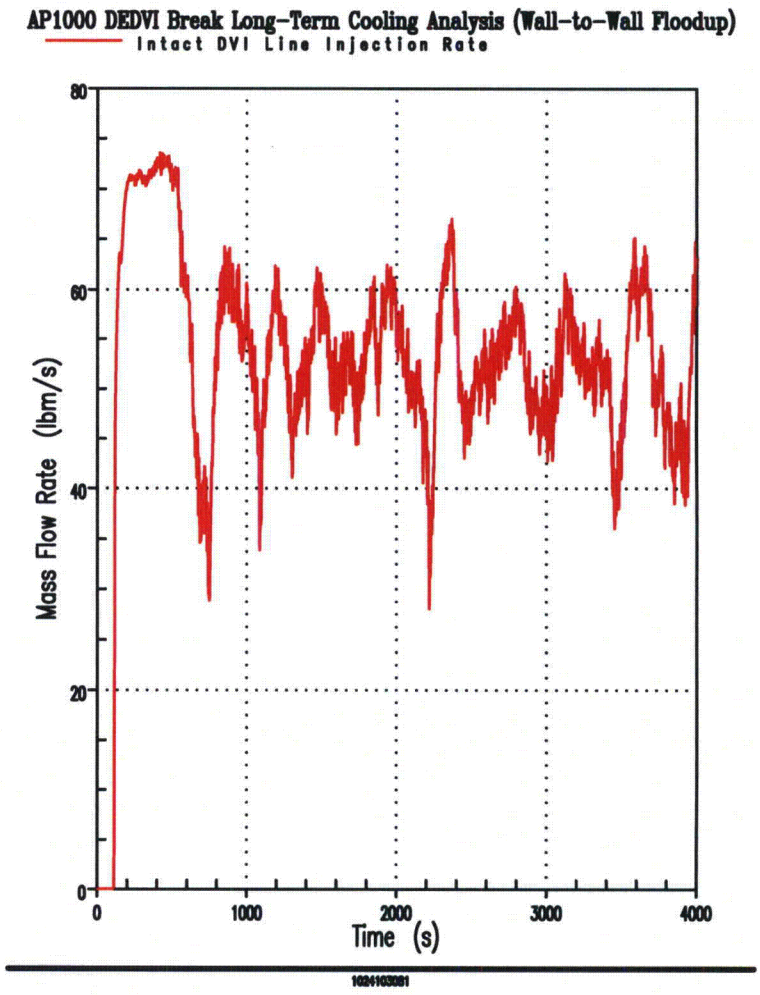

Figure 15.6.5.4C-27

DVI-A Mixture Flow Rate
(Wall-to-Wall Floodup Case) – 14.7 psia

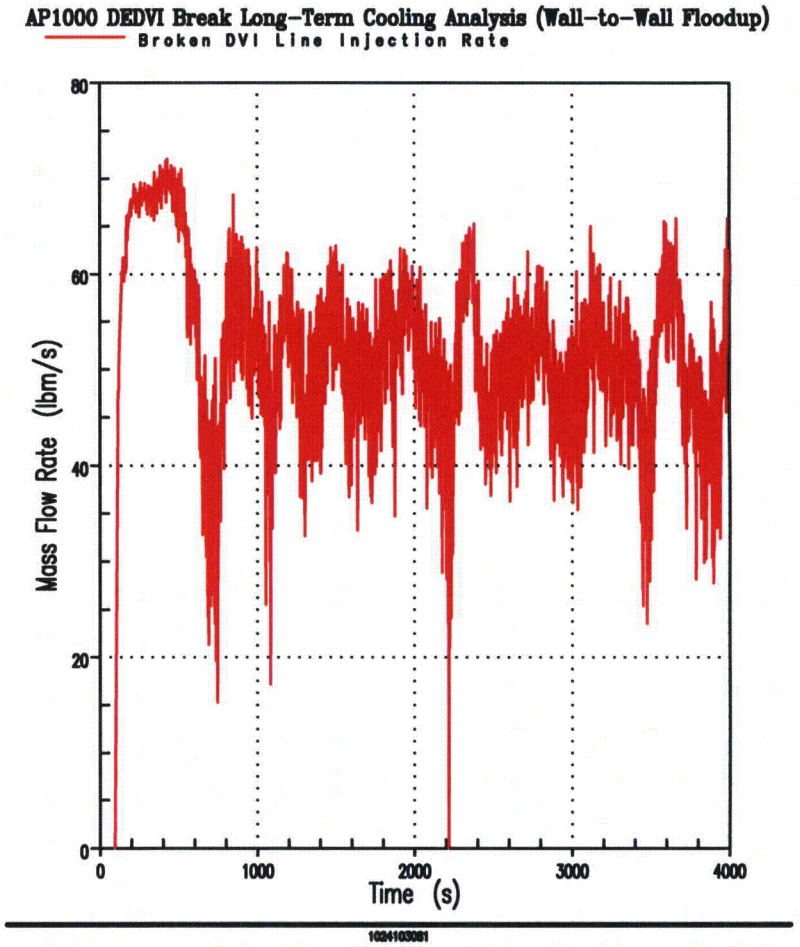

Figure 15.6.5.4C-28

DVI-B Mixture Flow Rate
(Wall-to-Wall Floodup Case) – 14.7 psia

15.7 Radioactive Release from a Subsystem or Component

This group of events includes the following:

- Gas waste management system leak or failure
- Liquid waste management system leak or failure (atmospheric release)
- Release of radioactivity to the environment via liquid pathways
- Fuel handling accident
- Spent fuel cask drop accident

15.7.1 Gas Waste Management System Leak or Failure

The AP1000 gaseous radwaste system is a low-pressure, low-flow charcoal delay process. Failure of the gaseous radwaste system results in a minor release of activity that is not significant. The Standard Review Plan no longer includes this event as part of the review. Therefore, no analysis is provided.

15.7.2 Liquid Waste Management System Leak or Failure (Atmospheric Release)

The AP1000 liquid radwaste system tanks do not contain significant levels of gaseous activity because liquids expected to contain gaseous radioactivity are processed by a gas stripper before being directed to storage. The tanks are open to the atmosphere so that any evolution of gaseous activity is continually released through the monitored plant vent. The Standard Review Plan no longer includes this event as part of the review. Therefore, no analysis is provided.

15.7.3 Release of Radioactivity to the Environment Due to a Liquid Tank Failure

Tanks containing radioactive fluids are located inside plant structures.

In the event of a tank failure, the liquid would be drained by the floor drains to the auxiliary building sump. From the sump, the water would be directed to the waste holdup tank. The basemat of the auxiliary building is 6-feet thick, the exterior walls are 3-feet thick, and the building is seismic Category I. The exterior walls are sealed to prevent leakage. Thus, it is assumed that there is no release of the spilled liquid waste to the environment. However, the Standard Review Plan states that credit cannot be taken for liquid retention by unlined building foundations. Analysis of the impact of this event will be performed as discussed in subsection 15.7.6. This analysis should include consideration of tank liquid level, processing and decay of tank contents, potential paths of spilled waste to the environment, as well as other pertinent factors.

15.7.4 Fuel Handling Accident

A fuel handling accident can be postulated to occur either inside the containment or in the fuel handling area inside the auxiliary building. The fuel handling accident is defined as the dropping of a spent fuel assembly such that every rod in the dropped assembly has its cladding breached so that the activity in the fuel/cladding gap is released.

The possibility of a fuel handling accident is remote because of the many administrative controls and the equipment operating limits that are incorporated in the fuel handling operations (see subsection 9.1.4). Only one spent fuel assembly is lifted at a time, and the fuel is moved at low speeds, exercising caution that the fuel assembly not strike anything during movement. The containment, auxiliary building, refueling pool, and spent fuel pool are designed to seismic Category I requirements to thus provide their integrity in the event of a safe shutdown earthquake. The spent fuel storage racks are located to prevent a credible external missile from reaching the stored fuel assemblies. The fuel handling equipment is designed to prevent the handling equipment from falling onto the fuel in the reactor vessel or that stored in the spent fuel pool. The facility is designed so that heavy objects, such as the spent fuel shipping cask, cannot be carried over or tipped into the spent fuel pool.

15.7.4.1 Source Term

The inventory of fission products available for release at the time of the accident is dependent on a number of factors, such as the power history of the fuel assembly, the time delay between reactor shutdown and the beginning of fuel handling operations, and the volatility of the nuclides.

The fuel handling accident source term is derived from the core source term detailed in Appendix 15A by taking into account the factors below. The assumptions used to define the fuel handling accident initial airborne release source term are provided in Table 15.7-1 along with the derived source term.

15.7.4.1.1 Fission Product Gap Fraction

During power operation, a portion of the fission products generated in the fuel pellet matrix diffuses into the fuel/cladding gap. The fraction of the assembly fission products found in the gap depends on the rate of diffusion for the nuclide in question as well as the rate of radioactive decay. In the event of a fuel handling accident, the gaseous and volatile radionuclides contained in the fuel/cladding gap are free to escape from the fuel assembly. The radionuclides of concern are the noble gases (kryptons and xenons) and iodines. Based on NUREG-1465 (Reference 1), the fission product gap fraction is 3-percent of fuel inventory. For this analysis, the gap fractions are increased to be consistent with the guidance of Regulatory Guide 1.183 (Reference 2). The gap fractions are listed in Table 15.7-1.

15.7.4.1.2 Iodine Chemical Form

Consistent with NUREG-1465 guidance, the iodine released from the damaged fuel rods is assumed to be 95-percent cesium iodide, 4.85-percent elemental iodine, and 0.15-percent organic iodine.

Cesium iodide is nonvolatile, and the iodine in this form dissolves in water but does not readily become airborne. However, consistent with the guidance in Regulatory Guide 1.183, it is assumed that the cesium iodide is instantaneously converted to the elemental form when released from the fuel into the low pH water pool.

15.7.4.1.3 Assembly Power Level

All fuel assemblies are assumed to be handled inside the containment during the core shuffle so a peak power assembly is considered for the accident. Any fuel assembly can be transferred to the spent fuel pool; during a core off-load, all fuel assemblies are discharged to the spent fuel pool. To obtain a bounding condition for the fuel handling accident analysis, it is assumed that the accident involves a fuel assembly that operated at the maximum rated fuel rod peaking factor. This is conservative because the entire fuel assembly does not operate at this level.

15.7.4.1.4 Radiological Decay

The fission product decay time experienced prior to the fuel handling accident is at least 48 hours.

15.7.4.2 Release Pathways

The spent fuel handling operations take place underwater. Thus, activity releases are first scrubbed by the column of water 23 feet in depth. This has no effect on the releases of noble gases or organic iodine but there is a significant removal of elemental iodine. Consistent with the guidance in Regulatory Guide 1.183, the overall pool scrubbing decontamination factor for iodine is assumed to be 200.

After the activity escapes from the water pool, it is assumed that it is released directly to the environment within a 2-hour period without credit for any additional iodine removal process.

If the fuel handling accident occurs in the containment, the release of activity can be terminated by closure of the containment purge lines on detection of high radioactivity. No credit is taken for this in the analysis. Additionally, no credit is taken for removal of airborne iodine by the filters in the containment purge lines.

For the fuel handling accident postulated to occur in the spent fuel pool, there is assumed to be no filtration in the release pathway. Activity released from the pool is assumed to pass directly to the environment with no credit for holdup or delay of release in the building.

15.7.4.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

Table 15.7-1 lists the assumptions used in the analysis. The guidance of Regulatory Guide 1.183 is reflected in the analysis assumptions.

15.7.4.4 Identification of Conservatisms

The fuel handling accident dose analysis assumptions contain a number of conservatisms. Some of these conservatisms are described in the following subsections.

15.7.4.4.1 Fuel Assembly Power Level

The source term is based on the assumption that all of the fuel rods in the damaged assembly have been operating at the maximum fuel rod radial peaking factor. In actuality, this is true for only a small fraction of the fuel rods in any assembly. The overall assembly power level is less than the maximum radial peaking factor.

15.7.4.4.2 Fission Product Gap Fraction

The assumption of Regulatory Guide 1.183 gap fractions for the short-lived nuclides is conservative by a factor of 2 or more, depending on the nuclide.

15.7.4.4.3 Amount of Fuel Damage

It is assumed that all fuel rods in a fuel assembly are damaged so as to release the fission product inventory in the fuel/cladding gap. In an actual fuel handling accident, it is expected that there would be few rods damaged to this extent.

15.7.4.4.4 Iodine Plateout on Fuel Cladding

Although it is expected that virtually all elemental iodine plates out on the fuel cladding and is unavailable for atmospheric release, no credit is taken for plateout.

15.7.4.4.5 Presence of Organic Iodine

Although 0.15% of the iodine is assumed to be in the organic form (and thus not subject to scrubbing removal in the water pool), there would be no organic iodine in the fuel rods. Any

formation of organic iodine would occur gradually and would not contribute to early releases of activity.

15.7.4.4.6 Conversion of Cesium Iodide to Form Elemental Iodine

The analysis assumes that all of the cesium iodide converts immediately to the elemental iodine form after release to the water pool and is treated in the same manner as the iodine initially in the elemental form. While the low pH solution does support conversion to the elemental form, the conversion would not occur unless the cesium iodide was dissolved in the water. The elemental iodine that is formed would thus be in the water solution and not in the bubbles of gas released from the damaged fuel. Additionally, conversion of cesium iodide would occur slowly and the elemental iodine formed would not be immediately available for release.

15.7.4.4.7 Meteorology

It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

15.7.4.4.8 Time Available for Radioactive Decay

The dose analysis assumes that the fuel handling accident involves one of the first fuel assemblies handled. If it were one of the later fuel handling operations, there is additional decay and a reduction in the source term.

The dose evaluation was performed assuming 48 hours decay.

15.7.4.5 Offsite Doses

Using the assumptions from Table 15.7-1, the calculated doses from the initial releases are determined to be 2.8 rem TEDE at the site boundary and 1.2 rem TEDE at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase "well within" is taken as meaning 25 percent or less.

15.7.5 Spent Fuel Cask Drop Accident

The spent fuel cask handling crane is prevented from travelling over the spent fuel. No radiological consequences analysis is necessary for the dropped cask event.

15.7.6 Combined License Information

Combined License applicant referencing the AP1000 certified design will perform an analysis of the consequences of potential release of radioactivity to the environment due to a liquid tank failure as outlined in subsection 15.7.3.

15.7.7 References

1. Sofer, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.

2. U. S. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, " July 2000.

Table 15.7-1

ASSUMPTIONS USED TO DETERMINE FUEL HANDLING ACCIDENT RADIOLOGICAL CONSEQUENCES

Source term assumptions	
– Core power (MWt)	3434[(1)]
– Decay time (hr)	48
Core source term after 48 hours decay (Ci)	
I-130	1.28 E+05
I-131	8.18 E+07
I-132	9.10 E+07
I-133	4.06 E+07
I-135	1.17 E+06
Kr-85m	1.52 E+04
Kr-85	1.07 E+06
Kr-88	5.45 E+02
Xe-131m	1.02 E+06
Xe-133m	4.47 E+06
Xe-133	1.70 E+08
Xe-135m	1.91 E+05
Xe-135	1.04 E+07
Number of fuel assemblies in core	157
Amount of fuel damage	One assembly
Maximum rod radial peaking factor	1.75
Percentage of fission products in gap	
I-131	8
Other iodines	5
Kr-85	10
Other noble gases	5
Pool decontamination factor for iodine	200
Activity release period (hr)	2
Atmospheric dispersion factors	See Table 15A-5 in Appendix 15A
Breathing rates (m^3/sec)	3.5 E-4
Nuclide data	See Appendix 15A

Note:

1. The main feedwater flow measurement supports a 1-percent power uncertainty.

15.8 Anticipated Transients Without Scram

15.8.1 General Background

An anticipated transient without scram (ATWS) is an anticipated operational occurrence during which an automatic reactor scram is required but fails to occur due to a common mode fault in the reactor protection system. Under certain circumstances, failure to execute a required scram during an anticipated operational occurrence could transform a relatively minor transient into a more severe accident. ATWS events are not considered to be in the design basis for Westinghouse plants.

15.8.2 Anticipated Transients Without Scram in the AP1000

For Westinghouse plants, the ATWS rule (10 CFR 50.62) requires the installation of equipment that is diverse from the reactor protection system to automatically trip the turbine and initiate decay heat removal. This equipment must be designed to perform its function in a reliable manner and be independent from sensor output to final actuation device from the existing reactor protection system.

The basis for the ATWS rule requirements, as outlined in SECY-83-293 (Reference 1), is to reduce the risk of core damage because of ATWS to less than 10^{-5} per reactor year.

The AP1000 includes a diverse actuation system, which provides the AMSAC protection features mandated for Westinghouse plants by 10 CFR 50.62, plus a diverse reactor scram (see Section 7.7). Thus, the ATWS rule is met.

15.8.3 Conclusion

The AP1000 is equipped with a diverse actuation system, which provides the functions required by the ATWS rule (10 CFR 50.62). The ATWS core damage frequency for the AP1000 is well below the SECY-83-293 goal of 10^{-5} per reactor year. The AP1000 ATWS core damage frequency is discussed in Chapter 33 of the Probabilistic Risk Assessment (PRA). The AP1000 design meets the ATWS rule (10 CFR 50.62) and its ATWS core damage frequency safety goal basis.

15.8.4 Combined License Information

This section has no requirement for additional information to be provided in support of the Combined License application.

15.8.5 References

1. Dircks, W. J., "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," SECY-83-293, U.S. NRC, July 19, 1983.

APPENDIX 15A

EVALUATION MODELS AND PARAMETERS FOR ANALYSIS OF RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

This appendix contains the parameters and models that form the basis of the radiological consequences analyses for the various postulated accidents.

15A.1 Offsite Dose Calculation Models

Radiological consequences analyses are performed to determine the total effective dose equivalent (TEDE) doses associated with the postulated accident. The determination of TEDE doses takes into account the committed effective dose equivalent (CEDE) dose resulting from the inhalation of airborne activity (that is, the long-term dose accumulation in the various organs) as well as the effective dose equivalent (EDE) dose resulting from immersion in the cloud of activity.

15A.1.1 Immersion Dose (Effective Dose Equivalent)

Assuming a semi-infinite cloud, the immersion doses are calculated using the equation:

$$D_{im} = \sum_i DCF_i \sum_j R_{ij} (\chi/Q)_j$$

where:

D_{im} = Immersion (EDE) dose (rem)

DCF_i = EDE dose conversion factor for isotope i (rem-m^3/Ci-s)

R_{ij} = Amount of isotope i released during time period j (Ci)

$(\chi/Q)_j$ = Atmospheric dispersion factor during time period j (s/m^3)

15A.1.2 Inhalation Dose (Committed Effective Dose Equivalent)

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_i DCF_i \sum_j R_{ij} (BR)_j (\chi/Q)_j$$

where:

D_{CEDE} = CEDE dose (rem)

DCF_i = CEDE dose conversion factor (rem per curie inhaled) for isotope i

R_{ij} = Amount of isotope i released during time period j (Ci)

$(BR)_j$ = Breathing rate during time period j (m^3/s)

$(\chi/Q)_j$ = Atmospheric dispersion factor during time period j (s/m^3)

15A.1.3 Total Dose (Total Effective Dose Equivalent)

The TEDE doses are the sum of the EDE and the CEDE doses.

15A.2 Main Control Room Dose Models

Radiological consequences analyses are performed to determine the TEDE doses associated with the postulated accident. The determination of TEDE doses takes into account the CEDE dose resulting from the inhalation of airborne activity (that is, the long-term dose accumulation in the various organs) as well as the EDE dose resulting from immersion in the cloud of activity.

15A.2.1 Immersion Dose Models

Due to the finite volume of air contained in the main control room, the immersion dose for an operator occupying the main control room is substantially less than it is for the case in which a semi-infinite cloud is assumed. The finite cloud doses are calculated using the geometry correction factor from Murphy and Campe (Reference 1).

The equation is:

$$D_{im} = \frac{1}{GF} \sum_i DCF_i \sum_j (IAR)_{ij} O_j$$

where:

D_{im} = Immersion (EDE) dose (rem)

GF = Main control room geometry factor
= $1173/V^{0.338}$

V = Volume of the main control room (ft^3)

DCF_i = EDE dose conversion factor for isotope i (rem-m^3/Ci-s)

$(IAR)_{ij}$ = Integrated activity for isotope i in the main control room during time period j ($Ci\text{-}s/m^3$)

O_j = Fraction of time period j that the operator is assumed to be present

15A.2.2 Inhalation Dose

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_i DCF_i \sum_j (IAR)_{ij} (BR)_j O_j$$

where:

D_{CEDE} = CEDE dose (rem)

DCF_i = CEDE dose conversion factor (rem per curie inhaled) for isotope i

$(IAR)_{ij}$ = Integrated activity for isotope i in the main control room during time period j ($Ci\text{-}s/m^3$)

$(BR)_j$ = Breathing rate during time period j (m^3/s)

O_j = Fraction of time period j that the operator is assumed to be present

15A.2.3 Total Dose (Total Effective Dose Equivalent)

The TEDE doses are the sum of the EDE and the CEDE doses.

15A.3 General Analysis Parameters

15A.3.1 Source Terms

The sources of radioactivity for release are dependent on the specific accident. Activity may be released from the primary coolant, from the secondary coolant, and from the core if the accident involves fuel failures. The radiological consequences analyses use conservative design basis source terms.

15A.3.1.1 Primary Coolant Source Term

The design basis primary coolant source terms are listed in Table 11.1-2. These source terms are based on continuous plant operation with 0.25-percent fuel defects. The remaining assumptions used in determining the primary coolant source terms are listed in Table 11.1-1.

The accident dose analyses take into account increases in the primary coolant source terms for iodines and noble gases above those listed in Table 11.1-2, consistent with the Tech Spec limits of 1.0 μCi/g dose equivalent I-131 for the iodines and 280 μCi/g dose equivalent Xe-133 for the noble gases.

The radiological consequences analyses for certain accidents also take into account the phenomenon of iodine spiking, which causes the concentration of radioactive iodines in the primary coolant to increase significantly. Table 15A-1 lists the concentrations of iodine isotopes associated with a pre-existing iodine spike. This is an iodine spike that occurs prior to the accident and for which the peak primary coolant activity is reached at the time the accident is assumed to occur. These isotopic concentrations are also defined as 60 μCi/g dose equivalent I-131. The probability of this adverse timing of the iodine spike and accident is small.

Although it is unlikely for an accident to occur at the same time that an iodine spike is at its maximum reactor coolant concentration, for many accidents it is expected that an iodine spike would be initiated by the accident or by the reactor trip associated with the accident. Table 15A-2 lists the iodine appearance rates (rates at which the various iodine isotopes are transferred from the core to the primary coolant by way of the assumed cladding defects) for normal operation. The iodine spike appearance rates are assumed to be as much as 500 times the normal appearance rates.

15A.3.1.2 Secondary Coolant Source Term

The secondary coolant source term used in the radiological consequences analyses is conservatively assumed to be 10 percent of the primary coolant equilibrium source term. This is more conservative than using the design basis secondary coolant source terms listed in Table 11.1-5.

Because the iodine spiking phenomenon is short-lived and there is a high level of conservatism for the assumed secondary coolant iodine concentrations, the effect of iodine spiking on the secondary coolant iodine source terms is not modeled.

There is assumed to be no secondary coolant noble gas source term because the noble gases entering the secondary side due to primary-to-secondary leakage enter the steam phase and are discharged via the condenser air removal system.

15A.3.1.3 Core Source Term

Table 15A-3 lists the core source terms at shutdown for an assumed three-region equilibrium cycle at end of life after continuous operation at 1 percent above full core thermal power. The main feedwater flow measurement supports a 1-percent power uncertainty In addition to iodines and noble gases, the source terms listed include nuclides that are identified as

potentially significant dose contributors in the event of a degraded core accident. The design basis loss-of-coolant accident analysis is

not expected to result in significant core damage, but the radiological consequences analysis assumes severe core degradation.

15A.3.2 Nuclide Parameters

The radiological consequence analyses consider radioactive decay of the subject nuclides prior to their release, but no additional decay is assumed after the activity is released to the environment. Table 15A-4 lists the decay constants for the nuclides of concern.

Table 15A-4 also lists the dose conversion factors for calculation of the CEDE doses due to inhalation of iodines and other nuclides and EDE dose conversion factors for calculation of the dose due to immersion in a cloud of activity. The CEDE dose conversion factors are from EPA Federal Guidance Report No. 11 (Reference 2) and the EDE dose conversion factors are from EPA Federal Guidance Report No. 12 (Reference 3).

15A.3.3 Atmospheric Dispersion Factors

Subsection 2.3.4 lists the off-site short-term atmospheric dispersion factors (χ/Q) for the reference site. Table 15A-5 (Sheet 1 of 2) reiterates these χ/Q values.

The atmospheric dispersion factors (χ/Q) to be applied to air entering the main control room following a design basis accident are specified at the HVAC intake and at the annex building entrance (which would be the air pathway to the main control room due to ingress/egress). A set of reference **AP1000** χ/Q values is identified for each potential activity release location that has been identified and the two control room receptor locations. These reference **AP1000** χ/Q values are listed in Table 15A-6 and are provided in Table 2-1 (Sheet 3 of 3).

The site-specific control room χ/Q values shall be bounded by the values in Table 15A-6.For a site selected that has χ/Q values that exceed the values in Table 15A-6, how the radiological consequences associated with the controlling design basis accident continue to meet the control room operator dose limits given in General Design Criteria 19 using site-specific χ/Q values should be addressed. Topographical characteristics in the vicinity of the site for restrictions of horizontal and/or vertical plume spread, channeling or other changes in airflow trajectories, and other unusual conditions affecting atmospheric transport and diffusion between the source and the receptors should be considered. No further action is required for sites within the bounds of the site parameters for atmospheric dispersion.

Table 15A-7 identifies the AP1000 source and receptor data to be used when determining the site-specific control room χ/Q values using the ARCON96 code (References 4 and 5).

The main control room reference **AP1000** χ/Q values do not incorporate occupancy factors.

The locations of the potential release points and their relationship to the main control room air intake and the annex building access door are shown in Figure 15A-1.

15A.4 References

1. Murphy, K. G., Campe, K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," paper presented at the 13th AEC Air Cleaning Conference.

2. EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, September 1988.

3. EPA Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," EPA 402-R-93-081, September 1993.

4. NUREG/CR-6331, Ramsdell, J. V. and Simonen, C. A., "Atmospheric Relative Concentrations in Building Wakes," Revision 1, May 1997.

5. Regulatory Guide 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, June 2003.

Table 15A-1

REACTOR COOLANT IODINE CONCENTRATIONS FOR MAXIMUM IODINE SPIKE OF 60 μCi/g DOSE EQUIVALENT I-131

Nuclide	μCi/g
I-130	0.38
I-131	42.8
I-132	61.8
I-133	81.6
I-134	15.7
I-135	51.9

Table 15A-2

IODINE APPEARANCE RATES IN THE REACTOR COOLANT

Nuclide	Equilibrium Appearance Rate (Ci/min)
I-130	4.05E-03
I-131	3.34E-01
I-132	1.50E+00
I-133	7.70 E-01
I-134	8.07E-01
I-135	6.96E-01

Table 15A-3 (Sheet 1 of 2)

REACTOR CORE SOURCE TERM[1]

	Nuclide	Inventory (Ci)		Nuclide	Inventory (Ci)
Iodines	I-130	1.87E+06	**Noble Gases**	Kr-85m	2.52E+07
	I-131	9.48E+07		Kr-85	1.07E+06
	I-132	1.38E+08		Kr-87	4.96E+07
	I-133	1.95E+08		Kr-88	6.66E+07
	I-134	2.19E+08		Xe-131m	1.03E+06
	I-135	1.85E+08		Xe-133m	6.07E+06
Cs Group	Cs-134	1.65E+07		Xe-133	1.92E+08
	Cs-136	3.87E+06		Xe-135m	4.08E+07
	Cs-137	1.09E+07		Xe-135	3.87E+07
	Cs-138	1.82E+08		Xe-138	1.66E+08
	Rb-86	1.80E+05	**Sr & Ba**	Sr-89	9.48E+07
Te Group	Te-127m	1.37E+06		Sr-90	8.34E+06
	Te-127	8.44E+06		Sr-91	1.17E+08
	Te-129m	4.74E+06		Sr-92	1.25E+08
	Te-129	2.49E+07		Ba-139	1.73E+08
	Te-131m	1.88E+07		Ba-140	1.68E+08
	Te-132	1.35E+08	**Ce Group**	Ce-141	1.59E+08
	Sb-127	8.71E+06		Ce-143	1.48E+08
	Sb-129	2.67E+07		Ce-144	1.22E+08
Ru Group	Ru-103	1.44E+08		Pu-238	2.22E+05
	Ru-105	9.92E+07		Pu-239	2.53E+04
	Ru-106	4.54E+07		Pu-240	3.96E+04
	Rh-105	9.21E+07		Pu-241	1.01E+07
	Mo-99	1.77E+08		Np-239	1.99E+09
	Tc-99m	1.56E+08			

Note:

1. The following assumptions apply:
   - Core thermal power of 3434 MWt (1 percent above the design core power of 3400 MWt). The main feedwater flow measurement supports a 1-percent power uncertainty.
   - Three-region equilibrium cycle core at end of life

Table 15A-3 (Sheet 2 of 2)

REACTOR CORE SOURCE TERM[1]

	Nuclide	Inventory (Ci)
La Group	Y-90	8.73E+06
	Y-91	1.23E+08
	Y-92	1.26E+08
	Y-93	1.42E+08
	Nb-95	1.65E+08
	Zr-95	1.63E+08
	Zr-97	1.62E+08
	La-140	1.77E+08
	La-142	1.52E+08
	Pr-143	1.44E+08
	Nd-147	6.12E+07
	Am-241	1.04E+04
	Cm-242	2.81E+06
	Cm-244	2.42E+05

Note:

1. The following assumptions apply:
   - Core thermal power of 3434 MWt (1 percent above the design core power of 3400 MWt). The main feedwater flow measurement supports a 1-percent power uncertainty.
   - Three-region equilibrium cycle core at end of life

Table 15A-4 (Sheet 1 of 4)

NUCLIDE PARAMETERS

A. HALOGENS

Isotope	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
I-130	$5.61x10^{-2}$	$1.04x10^{-13}$	$7.14x10^{-10}$
I-131	$3.59x10^{-3}$	$1.82x10^{-14}$	$8.89x10^{-9}$
I-132	$3.01x10^{-1}$	$1.12x10^{-13}$	$1.03x10^{-10}$
I-133	$3.33x10^{-2}$	$2.94x10^{-14}$	$1.58x10^{-9}$
I-134	$7.91x10^{-1}$	$1.30x10^{-13}$	$3.55x10^{-11}$
I-135	$1.05x10^{-1}$	$7.98x10^{-14}$	$3.32x10^{-10}$

B. NOBLE GASES

Isotope	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	
Kr-85m	$1.55x10^{-1}$	$7.48x10^{-15}$	
Kr-85	$7.38x10^{-6}$	$1.19x10^{-16}$	
Kr-87	$5.45x10^{-1}$	$4.12x10^{-14}$	
Kr-88	$2.44x10^{-1}$	$1.02x10^{-13}$	
Xe-131m	$2.43x10^{-3}$	$3.89x10^{-16}$	
Xe-133m	$1.32x10^{-2}$	$1.37x10^{-15}$	
Xe-133	$5.51x10^{-3}$	$1.56x10^{-15}$	
Xe-135m	2.72	$2.04x10^{-14}$	
Xe-135	$7.63x10^{-2}$	$1.19x10^{-14}$	
Xe-138	2.93	$5.77x10^{-14}$	

Table 15A-4 (Sheet 2 of 4)

NUCLIDE PARAMETERS

C. ALKALI METALS			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Cs-134	$3.84x10^{-5}$	$7.57x10^{-14}$	$1.25x10^{-8}$
Cs-136	$2.2x10^{-3}$	$1.06x10^{-13}$	$1.98x10^{-9}$
Cs-137[1]	$2.64x10^{-6}$	$2.88x10^{-14}$	$8.63x10^{-9}$
Cs-138	1.29	$1.21x10^{-13}$	$2.74x10^{-11}$
Rb-86	$1.55x10^{-3}$	$4.81x10^{-15}$	$1.79x10^{-9}$
D. TELLURIUM GROUP			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Te-127m	$2.65x10^{-4}$	$1.47x10^{-16}$	$5.81x10^{-9}$
Te-127	$7.41x10^{-2}$	$2.42x10^{-16}$	$8.60x10^{-11}$
Te-129m	$8.6x10^{-4}$	$1.55x10^{-15}$	$6.47x10^{-9}$
Te-129	$5.98x10^{-1}$	$2.75x10^{-15}$	$2.42x10^{-11}$
Te-131m	$2.31x10^{-2}$	$7.01x10^{-14}$	$1.73x10^{-9}$
Te-132	$8.86x10^{-3}$	$1.03x10^{-14}$	$2.55x10^{-9}$
Sb-127	$7.5x10^{-3}$	$3.33x10^{-14}$	$1.63x10^{-9}$
Sb-129	$1.6x10^{-1}$	$7.14x10^{-14}$	$1.74x10^{-10}$
E. STRONTIUM AND BARIUM			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Sr-89	$5.72x10^{-4}$	$7.73x10^{-17}$	$1.12x10^{-8}$
Sr-90	$2.72x10^{-6}$	$7.53x10^{-18}$	$3.51x10^{-7}$
Sr-91	$7.3x10^{-2}$	$3.45x10^{-14}$	$4.49x10^{-10}$
Sr-92	$2.56x10^{-1}$	$6.79x10^{-14}$	$2.18x10^{-10}$
Ba-139	$5.02x10^{-1}$	$2.17x10^{-15}$	$4.64x10^{-11}$
Ba-140	$2.27x10^{-3}$	$8.58x10^{-15}$	$1.01x10^{-9}$

Note:

1. The listed average gamma disintegration energy for Cs-137 is due to the production and decay of Ba-137m.

Table 15A-4 (Sheet 3 of 4)

NUCLIDE PARAMETERS

F. NOBLE METALS			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Ru-103	$7.35x10^{-4}$	$2.25x10^{-14}$	$2.42x10^{-9}$
Ru-105	$1.56x10^{-1}$	$3.81x10^{-14}$	$1.23x10^{-10}$
Ru-106	$7.84x10^{-5}$	0.0	$1.29x10^{-7}$
Rh-105	$1.96x10^{-2}$	$3.72x10^{-15}$	$2.58x10^{-10}$
Mo-99	$1.05x10^{-2}$	$7.28x10^{-15}$	$1.07x10^{-9}$
Tc-99m	$1.15x10^{-1}$	$5.89x10^{-15}$	$8.80x10^{-12}$
G. CERIUM GROUP			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Ce-141	$8.89x10^{-4}$	$3.43x10^{-15}$	$2.42x10^{-9}$
Ce-143	$2.1x10^{-2}$	$1.29x10^{-14}$	$9.16x10^{-10}$
Ce-144	$1.02x10^{-4}$	$8.53x10^{-16}$	$1.01x10^{-7}$
Pu-238	$9.02x10^{-7}$	$4.88x10^{-18}$	$1.06x10^{-4}$
Pu-239	$3.29x10^{-9}$	$4.24x10^{-18}$	$1.16x10^{-4}$
Pu-240	$1.21x10^{-8}$	$4.75x10^{-18}$	$1.16x10^{-4}$
Pu-241	$5.5x10^{-6}$	$7.25x10^{-20}$	$2.23x10^{-6}$
Np-239	$1.23x10^{-2}$	$7.69x10^{-15}$	$6.78x10^{-10}$

Table 15A-4 (Sheet 4 of 4)

NUCLIDE PARAMETERS

H. LANTHANIDE GROUP

Nuclide	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
Y-90	$1.08x10^{-2}$	$1.90x10^{-16}$	$2.28x10^{-9}$
Y-91	$4.94x10^{-4}$	$2.60x10^{-16}$	$1.32x10^{-8}$
Y-92	$1.96x10^{-1}$	$1.30x10^{-14}$	$2.11x10^{-10}$
Y-93	$6.86x10^{-2}$	$4.80x10^{-15}$	$5.82x10^{-10}$
Nb-95	$8.22x10^{-4}$	$3.74x10^{-14}$	$1.57x10^{-9}$
Zr-95	$4.51x10^{-4}$	$3.60x10^{-14}$	$6.39x10^{-9}$
Zr-97	$4.1x10^{-2}$	$9.02x10^{-15}$	$1.17x10^{-9}$
La-140	$1.72x10^{-2}$	$1.17x10^{-13}$	$1.31x10^{-9}$
La-141	$1.76x10^{-1}$	$2.39x10^{-15}$	$1.57x10^{-10}$
La-142	$4.5x10^{-1}$	$1.44x10^{-13}$	$6.84x10^{-11}$
Nd-147	$2.63x10^{-3}$	$6.19x10^{-15}$	$1.85x10^{-9}$
Pr-143	$2.13x10^{-3}$	$2.10x10^{-17}$	$2.19x10^{-9}$
Am-241	$1.83x10^{-7}$	$8.18x10^{-16}$	$1.20x10^{-4}$
Cm-242	$1.77x10^{-4}$	$5.69x10^{-18}$	$4.67x10^{-6}$
Cm-244	$4.37x10^{-6}$	$4.91x10^{-18}$	$6.70x10^{-5}$

Table 15A-5

OFFSITE ATMOSPHERIC DISPERSION FACTORS (χ/Q) FOR ACCIDENT DOSE ANALYSIS

Site boundary χ/Q (s/m^3)	
0 – 2 hours[1]	$5.1x10^{-4}$
Low population zone χ/Q (s/m^3)	
0 – 8 hours	$2.2x10^{-4}$
8 – 24 hours	$1.6x10^{-4}$
24 – 96 hours	$1.0x10^{-4}$
96 – 720 hours	$8.0x10^{-5}$

Note:

1. Nominally defined as the 0- to 2-hour interval but is applied to the 2-hour interval having the highest activity releases in order to address 10 CFR Part 50.34 requirements.

Table 15A-6

CONTROL ROOM ATMOSPHERIC DISPERSION FACTORS (χ/Q) FOR ACCIDENT DOSE ANALYSIS

χ/Q (s/m^3) at HVAC Intake for the Identified Release Points[1]						
	Plant Vent or PCS Air Diffuser[3]	Ground Level Containment Release Points[4]	PORV and Safety Valve Releases[5]	Steam Line Break Releases	Fuel Handling Area[6]	Condenser Air Removal Stack[7]
0 – 2 hours	3.0E-3	6.0E-3	2.0E-2	2.4E-2	6.0E-3	6.0E-3
2 – 8 hours	2.5E-3	3.6E-3	1.8E-2	2.0E-2	4.0E-3	4.0E-3
8 – 24 hours	1.0E-3	1.4E-3	7.0E-3	7.5E-3	2.0E-3	2.0E-3
1 – 4 days	8.0E-4	1.8E-3	5.0E-3	5.5E-3	1.5E-3	1.5E-3
4 – 30 days	6.0E-4	1.5E-3	4.5E-3	5.0E-3	1.0E-3	1.0E-3
χ/Q (s/m^3) at Annex Building Door for the Identified Release Points[2]						
	Plant Vent or PCS Air Diffuser[3]	Ground Level Containment Release Points[4]	PORV and Safety Valve Releases[5]	Steam Line Break Releases	Fuel Handling Area[6]	Condenser Air Removal Stack[7]
0 – 2 hours	1.0E-3	1.0E-3	4.0E-3	4.0E-3	6.0E-3	2.0E-2
2 – 8 hours	7.5E-4	7.5E-4	3.2E-3	3.2E-3	4.0E-3	1.8E-2
8 – 24 hours	3.5E-4	3.5E-4	1.2E-3	1.2E-3	2.0E-3	7.0E-3
1 – 4 days	2.8E-4	2.8E-4	1.0E-3	1.0E-3	1.5E-3	5.0E-3
4 – 30 days	2.5E-4	2.5E-4	8.0E-4	8.0E-4	1.0E-3	4.5E-3

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 are used for analysis of the doses due to a postulated small line break outside of containment. The plant vent and PCS air diffuser are potential release paths for other postulated events (loss-of-coolant accident, rod ejection accident, and fuel handling accident inside the containment); however, the values are bounded by the dispersion factors for ground level releases.
4. The listed values represent modeling the containment shell as a diffuse area source, and are 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.

5. The listed values bound the dispersion factors for releases from the steam line safety & power-operated relief valves. These dispersion factors would be used for evaluating the doses in the main control room for a steam generator tube rupture, a main steam line break, a locked reactor coolant pump rotor, and for the secondary side release from a rod ejection accident.
6. The listed values bound the dispersion factors for releases from the fuel storage and handling area. The listed values also bound the dispersion factors for releases from the fuel storage area in the event that spent fuel boiling occurs and the fuel building relief panel opens on high temperature. These dispersion factors are used for the fuel handling accident occurring outside containment and for evaluating the impact of releases associated with spent fuel pool boiling.
7. This release point is included for information only as a potential activity release point. None of the design basis accident radiological consequences analyses model release from this point.

Table 15A-7

CONTROL ROOM SOURCE/RECEPTOR DATA FOR DETERMINATION OF ATMOSPHERIC DISPERSION FACTORS

Source Description	Release Elevation Note 1 (m)	Horizontal Straight-Line Distance to Receptor		
		Control Room HVAC Intake (Elevation 19.9 m) (Δ1)	Annex Building Access (Elevation 1.5 m) (Δ2)	Comment
Plant Vent (⬡ 1)	55.7	147.2 ft (44.9 m)	379.3 ft (115.6 m)	
PCS Air Diffuser (⬡ 2)	69.8	118.1 ft (36.0 m)	343.2 ft (104.6 m)	
Fuel Building Blowout Panel (⬡ 3)	17.4	203.2 ft (61.9 m)	427.4 ft (130.3 m)	Note 3
Radwaste Building Truck Staging Area Door (⬡ 4)	1.5	218.5 ft (66.6 m)	433.5 ft (132.1 m)	Note 3
Steam Vent (⬡ 5)	17.1	61.5 ft (18.8 m)	261.6 ft (79.7 m)	
PORV/Safety Valves (⬡ 6)	19.2	66.9 ft (20.4 m)	255.4 ft (77.8 m)	
Condenser Air Removal Stack (⬡ 7)	38.4	198.3 ft (60.4 m)	58.3 ft (17.8 m)	Note 3
Containment Shell (Diffuse Area Source) (⬡ 8)	Same as receptor elevation (19.9 m or 1.5 m)	42.0 ft (12.8 m)	272.3 ft (83.0 m)	Note 2

Notes:

1. All elevations relative to grade at 0.0 m.
2. For calculating distance, the source is defined as the point on the containment shell closest to receptor.
3. Vertical distance traveled is conservatively neglected.
4. ⬡ – Refer to symbols on Figure 15A-1.
5. Δ – Refer to symbols on Figure 15A-1.

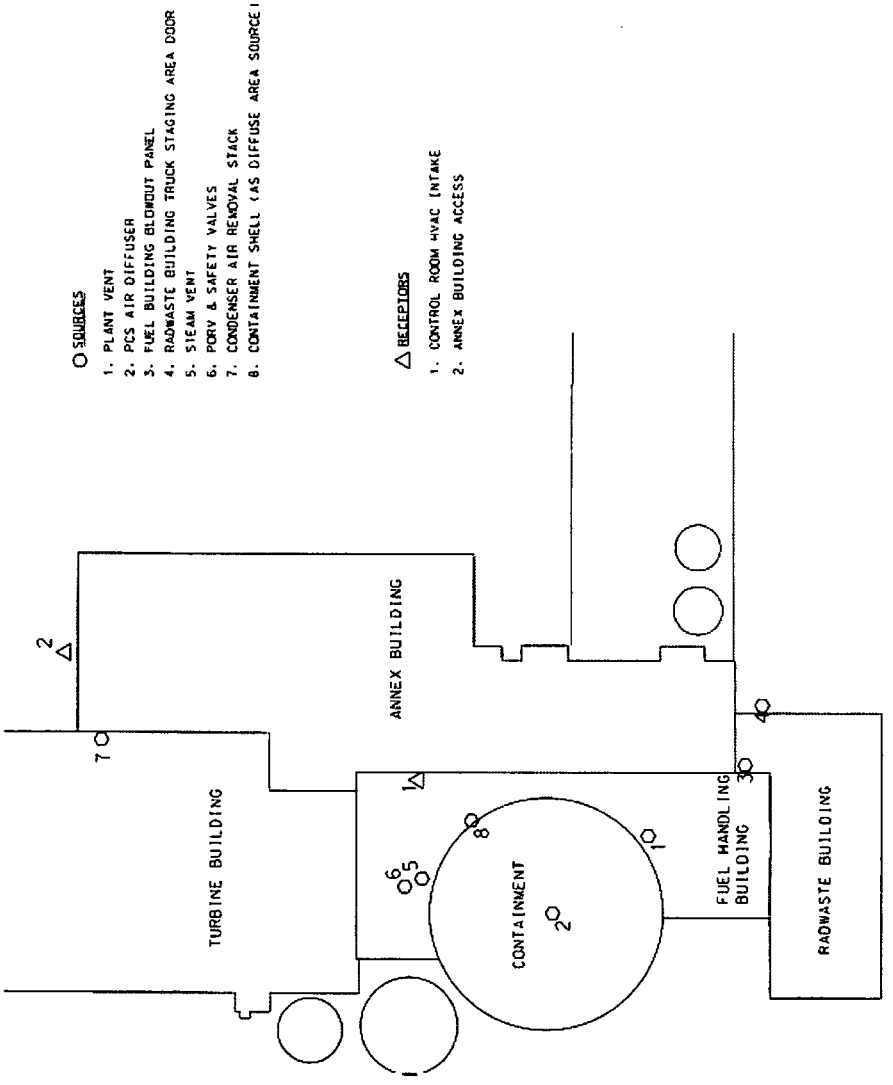

Figure 15A-1

Site Plan with Release and Intake Locations

APPENDIX 15B

REMOVAL OF AIRBORNE ACTIVITY FROM THE CONTAINMENT ATMOSPHERE FOLLOWING A LOCA

The AP1000 design does not depend on active systems to remove airborne particulates or elemental iodine from the containment atmosphere following a postulated loss-of-coolant accident (LOCA) with core melt. Naturally occurring passive removal processes provide significant removal capability such that airborne elemental iodine is reduced to very low levels within a few hours and the airborne particulates are reduced to extremely low levels within 12 hours.

15B.1 Elemental Iodine Removal

Elemental iodine is removed by deposition onto the structural surfaces inside the containment. The removal of elemental iodine is modeled using the equation from the Standard Review Plan (Reference 1):

$$\lambda_d = \frac{K_w A}{V}$$

where:

λ_d = first order removal coefficient by surface deposition

K_w = mass transfer coefficient (specified in Reference 1 as 4.9 m/hr)

A = surface area available for deposition

V = containment building volume

The available deposition surface is 219,000 ft^2, and the containment building net free volume is 2.06 x 10^6 ft^3. From these inputs, the elemental iodine removal coefficient is 1.7 hr^{-1}.

Consistent with the guidance of Reference 1, credit for elemental iodine removal is assumed to continue until a decontamination factor (DF) of 200 is reached in the containment atmosphere. Because the source term for the LOCA (defined in subsection 15.6.5.3) is modeled as a gradual release of activity into the containment, the determination of the time at which the DF of 200 is reached needs to be based on the amount of elemental iodine that enters the containment atmosphere over the duration of core activity release.

15B.2 Aerosol Removal

The deposition removal of aerosols from the containment atmosphere is accomplished by a number of processes including sedimentation, diffusiophoresis, and thermophoresis. All three of the deposition processes are significant contributors to the overall removal process in the AP1000. The large contributions from diffusiophoresis and thermophoresis to the total

removal are a direct consequence of the high heat transfer rates from the containment atmosphere to the containment wall that characterize the passive containment cooling system.

Because of the AP1000 passive containment cooling system design, there are high sensible heat transfer rates (resulting in higher thermophoretic removal of aerosols) when condensational heat transfer is low (and the aerosol removal by diffusiophoresis is also low). The reverse is also true. Thus, there is an appreciable deposition removal throughout the accident from either diffusiophoresis or thermophoresis, in addition to the removal by sedimentation.

15B.2.1 Mathematical Models

The models used for the three aerosol removal processes are discussed as follows.

15B.2.1.1 Sedimentation

Gravitational sedimentation is a major mechanism of aerosol removal in a containment. A standard model (Stokes equation with the Cunningham slip correction factor) for this process is used. The Stokes equation (Reference 2) is:

$$\upsilon_s = \frac{2\rho_p g r^2 Cn}{9\mu}$$

where:

υ_s = settling velocity of an aerosol particle

ρ_p = material density of the particle

g = gravitational acceleration

r = particle radius

μ = gas viscosity

Cn = Cunningham slip correction factor, a function of the Knudsen number (Kn) which is the gas molecular mean free path divided by the particle radius

However, the Stokes equation makes the simplifying assumption that the particles are spherical. The particles are expected to be nonspherical, and it is conventional to address this by introducing a "dynamic shape factor" (Reference 2) in the denominator of the Stokes equation, such that the settling velocity for the nonspherical particle is the same as for a spherical particle of equal volume. The value of the dynamic shape factor (ϕ) thus depends on the shape of the particle and, in general, must be experimentally determined.

The concept of dynamic shape factor can also be applied to a spherical particle consisting of two components, one of which has the density of the particle material, while the other component has a different density (Reference 9). In this manner, the impact of the void fraction in the particle can be modeled. Thus, the revised Stokes equation is:

$$\upsilon_s = \frac{2\rho_p g r^2 Cn}{9\mu\phi}$$

The derivation of ϕ follows.

The two-component particle is considered to have a density ρ_{av} and an effective radius of r_e. Assuming that the second component of the particle is the void volume and letting the void fraction be ε, then the average density of the particle is:

$$\rho_{av} = \text{the average density of the particle} = \rho_p (1-\varepsilon) + \rho_v \varepsilon$$

where:

ρ_v = density of the void material (0.0 for gas filled, 1.0 for water filled)
ε = void fraction
ρ_p = material density (solid particle with no voids)

The definition of ϕ is obtained from the Stokes equation and the equation for mass of a sphere:

$$\frac{2\rho_p g r^2 Cn}{9\mu\phi} = \frac{2\rho_{av} g r_e^2 Cn}{9\mu}$$

which reduces to:

$$\rho_p r^2 = \phi \rho_{av} r_e^2$$

and

$$\frac{4\rho_p \pi r_0^3}{3} = \frac{4\rho_{av} \pi r_e^3}{3}$$

which reduces to:

$$\rho_p r_0^3 = \rho_{av} r_e^3$$

Then:

$$\phi = \frac{\rho_p r^2}{\rho_{av} r_e^2}$$

and

$$r_e = r\left(\frac{\rho_{av}}{\rho_p}\right)^{-1/3}$$

From these two relationships, the dynamic shape factor is given by:

$$\phi = \left(\frac{\rho_{av}}{\rho_p}\right)^{-1/3}$$

15B.2.1.2 Diffusiophoresis

Diffusiophoresis is the process whereby particles are swept to a surface (for example, containment wall) by the flow set up by a condensing vapor (Stefan flow). The deposition rate is independent of the particle size and is proportional to the steam condensation rate on the surface. The standard equation for this phenomenon is due to Waldmann and Schmitt (Reference 3):

$$\upsilon_d = \frac{\sqrt{M_v}}{\sqrt{M_v} + \chi_{a/v}\sqrt{M_a}} \frac{W}{\rho_v}$$

where:

υ_d = diffusiophoretic deposition velocity

$\chi_{a/v}$ = ratio of mole fraction of air to mole fraction of steam in the containment atmosphere

M_v = molecular weight of steam

M_a = molecular weight of air

W = steam condensation rate on the wall

ρ_v = mass density of steam in the containment atmosphere

Because of the design of the passive containment cooling system, steam condensation rates are high at certain times in the design basis LOCA; thus at these times, diffusiophoretic deposition rates are significant.

15B.2.1.3 Thermophoresis

Thermophoresis is the process whereby particles drift toward a surface (for example, the containment wall) under the influence of a temperature gradient in the containment atmosphere at the surface. The effect arises because the gas molecules on the hot side of the particles undergo more collisions with the particle than do those on the cold side. Therefore, there is a net momentum transfer to the particle in the hot-to-cold direction. There are several models in the literature for this effect; the one used is the Brock equation in a form due to Talbot et al. (Reference 4). As indicated below, this model is in agreement with experimental data. The thermophoretic deposition rate is somewhat dependent on particle size and is proportional to the temperature gradient at the wall, or equivalently, the sensible heat transfer rate to the wall. The Talbot equation is:

$$\upsilon_{th} = \frac{2\,C_s\,Cn\,(\mu_g/\rho_g)[\alpha + C_T\,Kn]dT}{[1+2(\alpha + C_T\,Kn)][1+3\,C_M\,Kn]}\left(\frac{1}{T}\right)\frac{dT}{dy}$$

where:

υ_{th} = thermophoretic deposition velocity

α = k_g/k_p which is the ratio of the thermal conductivities of the gas (evaluated at the gas temperature at each time step) and the aerosol particle (k_p is set equal to the thermal conductivity of water – the results are not sensitive to k_p or α.)

Kn = Knudsen number (equal to the gas molecular mean free path divided by the particle radius)

Cn = Cunningham slip correction factor, a function of the Knudsen number

μ_g = gas viscosity

ρ_g = gas density

C_S = slip accommodation coefficient (Reference 4 gives the best value as 1.17.)

C_T = thermal accommodation coefficient (Reference 4 gives the best value as 2.18.)

C_M = momentum accommodation coefficient (Reference 4 gives the best value as 1.14.)

The temperature gradient at the wall, dT/dy, can be evaluated as

$$\frac{dT}{dy} = \frac{\phi_s}{k_g}$$

where ϕ_s is the sensible heat flux to the wall, and k_g is the thermal conductivity of the gas. The sensible heat flux used in the analysis is the convective heat transfer calculated as discussed in subsection 15B.2.4.7.

15B.2.2 Other Removal Mechanisms

In addition to the above mechanisms, there are others that were not considered, including turbulent diffusion and turbulent agglomeration. The neglect of these mechanisms adds further conservatism to the calculation.

15B.2.3 Validation of Removal Mechanisms

The aerosol processes are well established and have been confirmed in many separate effects experiments, which are discussed in standard references (References 2 through 4). The Stokes formula for sedimentation velocity has been well confirmed for particles whose diameters are less than about 50 μm. In the present calculations, these make up basically all of the aerosol.

There are some separate effects validations of the diffusiophoretic effect, but the best confirmation comes from integral experiments such as the LACE tests (Reference 5). Calculations of these and other integral tests accurately predict the integrated mass of plated aerosol material only if diffusiophoresis is taken into account. If it is neglected, the predicted plated mass is about two orders of magnitude too small, compared to the observed plated mass.

The Talbot equation for the thermophoretic effect has been experimentally confirmed to within about 20 to 50 percent over a wide range of particle sizes (Reference 4). The temperature gradient at the wall, which drives this phenomenon, can be approximated by the temperature difference between the bulk gas and the wall divided by an appropriate length scale obtained from heat transfer correlations. Alternatively, because sensible heat transfer rates to the wall are available, it is easier and more accurate to use these rates directly to infer the temperature gradient.

15B.2.4 Parameters and Assumptions for Calculating Aerosol Removal Coefficients

The parameters and assumptions were selected to conservatively model the environment that would be expected to exist as a result of a LOCA with concurrent core melt.

15B.2.4.1 Containment Geometry

The containment is assumed to be a cylinder with a volume of 55,481 m^3 (1.959 x 10^6 ft^3). This volume includes those portions of the containment volume that would be participating in the aerosol transport and mixing; this excludes dead-ended volumes and flooded compartments. The horizontal surface area available for aerosol deposition by sedimentation is 2900 m^2 (31,200 ft^2). This includes projecting areas such as decks in addition to the floor area and excludes areas in dead-ended volumes and areas that would be flooded post-LOCA. The surface area for Brownian diffusive plateout of aerosols is 8008 m^2 (86,166 ft^2).

15B.2.4.2 Source Size Distribution

The aerosol source size distribution is assumed to be lognormal, with a geometric mean radius of 0.22 μm and a geometric standard deviation equal to 1.81. These values are derived from an evaluation of a large number of aerosol distributions measured in a variety of

degraded-fuel tests and experiments. The sensitivity of aerosol removal coefficient calculations to these values is small.

15B.2.4.3 Aerosol Void Fraction

Review of scanning electron microscope photographs of deposited aerosol particles from actual core melt and fission product vaporization and aerosolization experiments (the Argonne STEP-4 test and the INEL Power Burst Facility SFD 1-4 test) indicates that the deposited particles are relatively dense, supporting a void fraction of 0.2.

The above-mentioned test results indicate that a void fraction of 0.2 is appropriate for modeling the aerosols resulting from a core melt. As part of the sensitivity study that was performed for the AP600 project, a case was run with a void fraction of 0.9. That analysis showed that the high void fraction resulted in an integrated release of aerosols over a 24-hour period that was less than 14 percent greater than that calculated when using the void fraction of 0.2. Thus, it is clear that the removal of aerosols from the containment atmosphere is not highly sensitive to the value selected for the void fraction. This is largely due to the fact that, while the selected value for void fraction has a significant impact on the calculated sedimentation removal, the impact on thermophoresis and diffusiophoresis removal is slight or none. The impact for AP1000 of using the higher value for void fraction would be less than was determined for the AP600 since sedimentation removal comprises a smaller fraction of the total removal calculated for the AP1000.

For additional conservatism, the AP1000 aerosol removal analysis uses a void fraction of 0.4 and assumes the voids are filled with air.

15B.2.4.4 Fission Product Release Fractions

Core inventories of fission products are from ORIGEN calculations for the AP1000 at end of the fuel cycle. Fractional releases to the containment of the fission products are those specified in subsection 15.6.5.3.

15B.2.4.5 Inert Aerosol Species

The inert species include SnO_2, UO_2, Cd, Ag, and Zr. These act as surrogates for all inert materials forming aerosols. The ratio of the total mass of inert species to fission product species was assumed to be 1.5:1. This value and the partitioning of the total inert mass among its constituents are consistent with results from degraded fuel experiments (Reference 6).

15B.2.4.6 Aerosol Release Timing and Rates

Aerosol release timing is in accordance with the source term defined in subsection 15.6.5.3. Aerosol release takes place in two main phases: a gap release lasting for 0.5 hour, followed by an early in-vessel release of 1.3 hours duration. During each phase, the aerosols are assumed to be released at a constant rate. These rates were obtained for each species by combining its core inventory, release fraction, and times of release.

Only cesium and iodine are released during the gap release phase. During the in-vessel release phase, the other fission product and inert species are released as well.

15B.2.4.7 Containment Thermal-Hydraulic Data

The thermal-hydraulic parameters used in the aerosol removal calculation are the containment gas temperature, the containment pressure, the steam condensation rate on the wall, the steam mole fraction, and the convective heat transfer rate, all as functions of time. The AP1000-specific parameters were obtained using MAAP4 (Reference 7) for the 3BE-1 severe accident sequence (medium LOCA with failure to inject water from the refueling water storage tank into the reactor vessel). The thermal-hydraulic data are thus consistent with a core melt sequence.

15B.2.5 Aerosol Removal Coefficients

The aerosol removal coefficients are provided in Table 15B-1 starting at the onset of core damage through 24 hours. The removal coefficients for times beyond 24 hours are not of concern because there would be so little aerosol remaining airborne at that time. The values range between 0.29 hr^{-1} and 1.1 hr^{-1} during the time between the onset of core damage (0.167 hour) and 24 hours.

These removal coefficients conservatively neglect steam condensation on the airborne particles, turbulent diffusion, and turbulent agglomeration. Additionally, the assumed source aerosol size is conservatively small being at the low end of the mass mean aerosol size range of 1.5 to 5.5 μm used in NUREG/CR-5966 (Reference 8). Selection of smaller aerosol size would underestimate sedimentation.

Unlike the case for the elemental iodine removal, there is no limit assumed on the removal of aerosols from the containment atmosphere.

15B.3 References

1. NUREG-0800, Section 6.5.2, Revision 2, "Containment Spray as a Fission Product Cleanup System."

2. Fuchs, N. A., The Mechanics of Aerosols, Pergamon Press, Oxford, 1964.

3. Waldmann, L., and Schmitt, K. H., "Thermophoresis and Diffusiophoresis of Aerosols," Aerosol Science, C. N. Davies, ed., Academic Press, 1966.

4. Talbot, L., Chang, R. K., Schefer, R. W., and Willis, D. R., "Thermophoresis of Particles in a Heated Boundary Layer," J. Fluid Mech. 101, 737-758 (1980).

5. Rahn, F. J., "The LWR Aerosol Containment Experiments (LACE) Project," Summary Report, EPRI-NP-6094D, Electric Power Research Institute, Palo Alto, Nov. 1988.

6. Petti, D. A., Hobbins, R. R., and Hagrman, D. L., "The Composition of Aerosols Generated during a Severe Reactor Accident: Experimental Results from the Power Burst Facility Severe Fuel Damage Test 1-4," Nucl. Tech. 105, p.334 (1994).

7. MAAP4 - Modular Accident Analysis Program for LWR Power Plants, Computer Code Manual, May 1994.

8. Powers D. A., and Burson, S. B., "A Simplified Model of Aerosol Removal by Containment Sprays," NUREG/CR-5966, June 1993.

9. Powers, D. A., "Monte Carlo Uncertainty Analysis of Aerosol Behavior in the AP600 Reactor Containment under Conditions of a Specific Design-Basis Accident, Part 1," Technical Evaluation Report, Sandia National Laboratories, June 1995.

Table 15B-1 (Sheet 1 of 3)

AEROSOL REMOVAL COEFFICIENTS IN THE AP1000 CONTAINMENT FOLLOWING A DESIGN BASIS LOCA WITH CORE MELT

Time Interval (hours)	Removal Coefficient (hr^{-1})
0.167 - 0.179	1.141
0.179 - 0.200	1.013
0.200 - 0.251	0.944
0.251 - 0.292	0.882
0.292 - 0.433	0.842
0.433 - 0.631	0.901
0.631 - 0.684	0.821
0.684 - 0.801	0.781
0.801 - 0.893	0.735
0.893 - 1.033	0.699
1.033 - 1.171	0.662
1.171 - 1.233	0.627
1.233 - 1.331	0.594
1.331 - 1.395	0.562
1.395 - 1.429	0.551
1.429 - 1.475	0.576
1.475 - 1.519	0.537
1.519 - 1.579	0.510
1.579 - 1.653	0.483
1.653 - 1.776	0.458
1.776 - 1.903	0.430
1.903 - 1.991	0.462
1.991 - 2.067	0.429
2.067 - 2.176	0.396
2.176 - 2.371	0.380
2.371 - 2.621	0.337

Table 15B-1 (Sheet 2 of 3)

AEROSOL REMOVAL COEFFICIENTS IN THE AP1000 CONTAINMENT FOLLOWING A DESIGN BASIS LOCA WITH CORE MELT

Time Interval (hours)	Removal Coefficient (hr^{-1})
2.621 - 2.822	0.320
2.822 - 2.872	0.357
2.872 - 2.973	0.327
2.973 - 3.176	0.302
3.176 - 3.684	0.287
3.684 - 3.737	0.328
3.737 - 3.839	0.304
3.839 - 3.990	0.298
3.990 - 4.090	0.317
4.090 - 4.438	0.346
4.438 - 4.684	0.369
4.684 - 4.880	0.396
4.880 - 4.928	0.449
4.928 - 5.362	0.435
5.362 - 5.460	0.459
5.460 - 5.511	0.518
5.511 - 5.608	0.487
5.608 - 6.040	0.479
6.040 - 6.090	0.537
6.090 - 6.615	0.506
6.615 - 6.753	0.567
6.753 - 7.194	0.513
7.194 - 7.285	0.594
7.285 - 7.814	0.518
7.814 - 7.904	0.581

Table 15B-1 (Sheet 3 of 3)

AEROSOL REMOVAL COEFFICIENTS IN THE AP1000 CONTAINMENT FOLLOWING A DESIGN BASIS LOCA WITH CORE MELT

Time Interval (hours)	Removal Coefficient (hr^{-1})
7.904 - 8.431	0.528
8.431 - 8.521	0.589
8.521 - 9.387	0.529
9.387 - 9.553	0.568
9.553 - 11.189	0.530
11.189 - 14.937	0.516
14.937 - 17.610	0.506
17.610 - 24	0.492