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August 1, 2003

SUBJECT: Transmittal of Westinghouse Responses to Open Items Identified in the AP1000 Draft Safety Evaluation Report

This letter transmits revised Westinghouse responses to open items identified in the AP1000 Draft Safety Evaluation Report (DSER) that was issued on June 16, 2003. A list of the DSER Open Item responses that are transmitted with this letter is provided in Attachment 1. Attachment 2 provides the DSER Open Item responses.

Please contact me if you have questions regarding this transmittal.

Very truly yours,

M. M. Corletti

M. M. Corletti
Passive Plant Projects & Development
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/Attachments

1. Table 1, "List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1607"
2. Westinghouse Non-Proprietary Responses to US Nuclear Regulatory Commission DSER Open Items dated August 1, 2003

D063

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DCP/NRC1607
Docket 52-006

August 1, 2003

Attachment 1

"List of Westinghouse's Responses to DSER Open Items
as Transmitted in DCP/NRC1607"

Attachment 1

Table 1 "List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1607"	
2.3.4-1 6.4-1 15.3-2	14.2.10-4 Rev. 1 14.3.2-12 Rev. 1 19.1.10.3-1 Rev. 1 19A.2-8 Rev. 1

August 1, 2003

Attachment 1

Westinghouse Non-Proprietary Responses to
AP1000 Draft Safety Evaluation Report (DSER) Open Items

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 2.3.4-1

Original RAI Number(s): 451.006, 451.006 Rev. 1

Summary of Issue:

The hypothetical reference control room χ/Q values calculated by the applicant are listed in Table 15.3-9a of this report. A site selected for an AP1000 facility should have control room χ/Q values equal to or less than the hypothetical Reference χ/Q values shown Table 15.3-9a. In the event a site selected for the AP1000 design exceeds the hypothetical reference χ/Q values, the COL applicant should demonstrate that the radiological consequences associated with the design-basis accidents, using its site-specific χ/Q values, continues to meet the dose reference values given in GDC 19 of 10 CFR Part 50. The staff initially asked the applicant if the methodology and all inputs and assumptions would be evaluated as part of the COL review. The applicant provided a detailed response stating that the methodology, inputs and assumptions would be provided by the COL applicant and also provided additional information about the analysis. The staff issued a second RAI to inquire if the applicant was seeking certification of any of the AP1000 design values used as inputs to the control room χ/Q calculations. The applicant subsequently provided certain design-specific information that was used as input to the assessment and for which the applicant was seeking certification. The staff review of this topic is ongoing, and may reveal other concerns with respect to χ/Q. The staff has identified unresolved issues related to adequate justification for assuming a diffuse release, estimation of initial sigma values, other release assumptions, building cross-sectional areas, and distances between release/receptor pairs. This is Open Item 2.3.4-1. This is also COL Action Item 2.3.4-1 since the resultant χ/Q values are also a function of the site-specific meteorology which cannot be reviewed until site selection.

Westinghouse Response:

The AP1000 control room χ/Q values used in the AP1000 dose analyses were based on the calculation performed for the AP600 Design Certification. This calculation examined a wide range of site meteorological data and plant orientations to develop a conservative set of χ/Q values for use in the AP600 dose analyses. It was determined that the calculated χ/Q values for AP600 would be conservative for AP1000, given the same set of meteorological data and plant orientation, and therefore these χ/Q values were used in the AP1000 dose analyses.

To resolve this DSER Open Item (and related DSER Open Items 15.3-2 and 6.4-1), the NRC requested Westinghouse to perform a compliance assessment of the calculation of control room χ/Q values against the recently issued Regulatory Guide 1.194. Regulatory Guide 1.194 provides specific NRC staff guidance on the use of the ARCON96 code for calculating control room χ/Q. The calculation performed for AP600 and applied to AP1000 used the ARCON96 computer code following the guidance that was supplied in the Code User's Manual which was the only guidance available when the calculation was performed in 1997. Table 2.3.4-1-1 describes the conformance of this calculation to Regulatory Guide 1.194. As shown in Table

2.3.4-1-1, the use of the ARCON96 code generally complies with the new Regulatory Guide. However, some differences exist with respect to the modeling assumptions related to a diffuse source, as mentioned in this DSER open item.

An underlying issue regarding this exercise, and part of the original NRC RAI related to this issue, is whether the control room χ/Q values are being approved for AP1000 as part of Design Certification. Unlike the offsite χ/Q values that were identified as site interface parameters that the COL applicant would later verify for their site, the control room χ/Q values were not identified as a site interface for either the AP600 or AP1000. Westinghouse agrees with the NRC that control room χ/Q values should be identified as a site interface parameter. Thus a COL applicant would verify as part of the COL process that the calculated control room χ/Q values for their site are bounded by those assumed in the DCD dose analysis. However, unlike the site boundary χ/Q values that are based solely on the site meteorological data, the control room χ/Q values are determined based both on site meteorological data and assumptions related to plant design features and layout. Westinghouse believes that the assumptions related to the plant design features are important in ultimately determining acceptable control room doses for design basis accidents, and therefore should be approved as part of Design Certification.

Therefore the following approach is being taken to resolve these issues:

1. Bounding control room χ/Q values will be established for the AP1000. These values will be determined for the various source – receptor locations that are applicable for the various design basis accidents as appropriate. The maximum χ/Q values that will still yield doses within the dose acceptance limits will be calculated consistent with the dose analysis methodology and assumptions described in the DCD Chapters 6 and 15. Consistent with the approach of treating the control room χ/Q values as interface parameters, Westinghouse will revise some assumptions described in the current DCD dose analysis to remove excess conservatism to provide the COL applicant greater flexibility in demonstrating acceptability. A summary of these changes is provided in Table 2.3.4.1-2. DCD section 2.3.6.4 already requires the COL applicant to demonstrate the acceptability of the actual site meteorology.

2. The bounding χ/Q values that are included in Appendix 15A will be referenced as an interface requirement in AP1000 DCD Tier 2, Chapter 2, Table 2-1, Site Parameters.

3. The key control room χ/Q modeling assumptions related to the plant design will be added to DCD Appendix 15A. This information will be similar in content to the information provided in Table 451.006R1-1 that was provided in the Westinghouse revision 1 response to RAI 451.006. This will include the methodology for determining the source sigma values when a diffuse source is assumed.

4. A set of χ/Q values for typical site meteorology and plant orientation will be calculated in accordance with the guidance set forth in Regulatory Guide 1.194. The purpose of the calculation is to define the modeling assumptions for calculating the control room χ/Q values for AP1000, and will serve as an example of an approved method for the COL applicant to

follow to determine the acceptability of their site to meet the control room χ/Q values. Relevant portions of this calculation will be included in DCD Appendix 15A.

Westinghouse will incorporate these changes in the next revision of the DCD. A draft markup of the DCD Chapters 2, 6 and 15 are attached to this DSER open item response. This markup does not include the final values for the control room χ/Q values and the description of the methodologies used to calculate these values. This information will appear in the next revision of the DCD.

Design Control Document (DCD) Revision:

See attached draft markup of the DCD.

PRA Revision:

None

Table 2.3.4-1-1
Conformance of AP1000 Control Room χ/Q Analysis to Reg. Guide 1.194

Parameter	Discussion	Acceptable Input	Conform?	Comments
Lower Measurement Height, meters	The value of this parameter is used by ARCON96 to adjust wind speeds for differences between the heights of the instrumentation and the release.	Use the actual instrumentation height when known. Otherwise, assume 10 meters.	Yes	Used height of 10 m above grade.
Upper Measurement Height, meters	The value of this parameter is used by ARCON96 to adjust wind speeds for differences between the heights of the instrumentation and the release.	Use the actual instrumentation height when known. Otherwise, use the height of the containment or the stack height, as appropriate. If wind speed measurements are available at more than two elevations, the instrumentation at the height closest to the release height should be used.	N/A	The use of wind direction and speed data from the upper meteorological tower instruments is optional in ARCON96. The calculation employed only the lower level wind data. To ensure that ARCON96 computes correctly, the meteorological data input files arbitrarily define the Upper Measurement Height at 100 meters (based on discussion with the program developer.).
Wind Speed Units	ARCON96 requires that wind speed be entered as miles per hour, ms^{-1}, or knots.	Use the wind speed units that correspond to the units of the wind speeds in the meteorological data file.	Yes	
Release Height, meters	The value of the release height is used for three purposes in ARCON96: (1) to adjust wind speeds for differences between the heights of the instrumentation and the release, (2) to determine slant path for ground level releases, (3) to correct off-centerline data for elevated releases.	Use the actual release heights whenever available. Plume rise from buoyancy and mechanical jet effects may be considered in establishing the release height if the analyst can demonstrate with reasonable assurance that the vertical velocity of the release will be maintained during the course of the accident. If actual release height is not available, set release height equal to intake height.	Yes	Actual release heights, with respect to grade datum, are used.

Table 2.3.4-1-1
Conformance of AP1000 Control Room χ/Q Analysis to Reg. Guide 1.194

Parameter	Discussion	Acceptable Input	Conform?	Comments
Building Area, meters^2	ARCON96 uses the value of the building area in the high speed wind speed adjustment for ground-level and vent release models.	Use the actual building vertical cross-sectional area perpendicular to the wind direction. Use default of 2000 m^2 if the area is not readily available. Do not enter zero. Use 0.01 m^2 if a zero entry is desired. *Note: This building area is for the building(s) that has the largest impact on the building wake within the wind direction window. This is usually, but need not always be, the reactor containment. With regard to the diffuse area source option, the building area entered here may be different from that used to establish the diffuse source.*	Yes	The actual vertical cross-sectional areas of major structures perpendicular to the wind direction are used, as appropriate. Where a portion of a structure is shadowed from the wind, only the un-shadowed area is used.

Table 2.3.4-1-1
Conformance of AP1000 Control Room χ/Q Analysis to Reg. Guide 1.194

Parameter	Discussion	Acceptable Input	Conform?	Comments
Vertical Velocity, meters/seconds	In ARCON96, the value of the vertical velocity is used only in vent and stack release models. It is used for the downwash calculation. In the vent release model the velocity is used in the mixed-mode calculation. If the vertical velocity is set to zero, the maximum downwash will be calculated and the release height will be reduced by an amount equal to six times the stack radius.	Note: the vent release model should not be used for DBA accident calculations. For stack release calculations only, use the actual vertical velocity if the licensee can demonstrate with reasonable assurance that the value will be maintained during the course of the accident (e.g., addressed by technical specifications), otherwise, enter zero. If the vertical velocity is set to zero, ARCON96 will reduce the stack height by 6 times the stack radius for all wind speeds. If this reduction is not desired, the stack radius should also be set to zero.	No	AP600 used the vent release mode for the Plant Vent release. Thus, this case produced less conservative results than what would be obtained using the Regulatory Guide. The Spent Fuel Pool Boiling release is modeled as a "capped" vent, with the exhaust rate through the blowout panel equal to the steaming rate of the pool; however, the vertical flow velocity is set to zero. Thus, this case produced slightly less conservative results than what would be obtained using the Regulatory Guide. Also, the Condenser Air Removal Exhaust Stack was modeled as a capped vent with no flow. The results are presumably comparable to a ground level release, but this would have to be confirmed. Releases via either the main steam safety valves (MSSVs) and pilot-operated relief valves (PORVs) are energetic; however, the AP600 model conservatively treats these as ground-level releases with exhaust velocity set to zero. Note that, if it can be demonstrated that the exhaust velocities exceed 5 times the 95%ile wind speed, Regulatory Guide 1.194 allows reducing the computed ground level χ/Qs by a factor of 5.

Table 2.3.4-1-1
Conformance of AP1000 Control Room χ/Q Analysis to Reg. Guide 1.194

Parameter	Discussion	Acceptable Input	Conform?	Comments
Stack Flow, meters^3/s	ARCON96 uses the value of the stack flow in χ/Q calculations for all 3 release types to ensure that the near field concentrations are no greater than the concentration at the release point. The impact diminishes with increasing distance.	Use actual flow if it can be demonstrated with reasonable assurance that the value will be maintained during the course of the accident (e.g., addressed by technical specifications). Otherwise, enter zero. The flow is used in both elevated and ground-level release modes to establish a maximum χ/Q value. This value is significant only if the flow is large and the distance from the release point to the receptor is small.	Yes	The AP600 model conservatively treats releases from the MSSVs and PORVs as ground-level releases with stack flow set to zero.
Stack Radius, meters	ARCON96 uses the value of the stack radius in downwash calculations in the vent and stack release modes.	Use the actual stack internal radius when both the stack radius and vertical velocity are available. If the stack flow is zero, the radius should be set to zero.	Yes	Since stack flow for the MSSV and PORV releases is set to zero, the stack radius is set to zero for these cases.
Distance to Receptor, meters	The value of horizontal distance to the receptor from the release point is used in ARCON96 for calculating the slant range for ground level releases and the off-centerline correction factors for stack release models.	Use the actual straight line horizontal distance between the release point and the control room intake. For ground-level releases, it may be appropriate to consider flow around an intervening building if the building is sufficiently tall that it is unrealistic to expect flow from the release point to go over the building. *Note: If the distance to receptor is less than about 10 meters, ARCON96 should not be used to assess relative concentrations.*	Yes	The actual straight-line horizontal distance between the release point and the control room intake is used in all cases. Flow around intervening buildings is not considered. *No source-receptor distance is less than 10 meters.*

Table 2.3.4-1-1
Conformance of AP1000 Control Room χ/Q Analysis to Reg. Guide 1.194

Parameter	Discussion	Acceptable Input	Conform?	Comments
Intake Height, meters	The value of the intake height is used in ARCON96 for calculating the slant range for ground level releases and the off-centerline correction factors for stack release models.	Use the actual intake height. If the intake height is not available for ground level releases, assume the intake height is equal to the release height. For elevated releases, assume the height of the tallest site building.	Yes	The actual heights at the centerline of the control room intakes are used.
Elevation Difference, meters	The value of this parameter is used by ARCON96 to normalize the release heights and the intake heights when the two heights are specified as "above grade" with different grades for the release point and intake height, or when one measurement is referenced to "above grade" and the other "above sea level."	Use zero unless it is known that the release heights are reported relative to different grades or reference datum.	Yes	All release heights are reported with respect to the same grade datum.

Table 2.3.4-1-1
Conformance of AP1000 Control Room χ/Q Analysis to Reg. Guide 1.194

Parameter	Discussion	Acceptable Input	Conform?	Comments
Direction to Source, degrees	ARCON96 uses the value of this parameter and the Wind Direction Window to establish which range of wind directions should be included in the assessment of the χ/Q.	Use the direction FROM the intake back TO the release point. (Wind directions are reported as the direction from which the wind is blowing. Thus, if the direction from the intake to the release point is north, a north wind will carry the plume from the release point to the intake.) Note: some facilities have a "plant north" shown on site arrangement drawings that is different from "true north." The direction entered must have the same point of reference as the wind directions reported in the meteorological data. For ground level releases, if the plume is assumed to flow around a building rather than over it, the direction may need to be modified to account for the redirected flow. In this case, the χ/Q should be calculated assuming flow around and flow over (through) the building and the higher of the two χ/Q s should be used.	Yes	For the generic (enveloping) control room χ/Qs, the actual plant orientation is unknown. The AP600 model uses actual meteorological data from three existing nuclear power plant sites, representative of an inland rolling hills site, a Midwestern river valley site, and a coastal site. For all three sites, the AP600 plant arrangement was "rotated" through the 16 compass points (i.e., every 22.5°); a separate ARCON96 run was performed for each combination of release location, site, and plant orientation. The maximum 95th percentile χ/Qs for each site over the 16 plant orientations for all source/receptor combinations are then determined. Finally, the maximum of the three sites' maximum 95th percentile χ/Q values are taken as the generic AP600 control room χ/Qs. Because of this conservative, enveloping approach, the orientation of an actual plant with respect to the wind is not relevant.

Table 2.3.4-1-1
Conformance of AP1000 Control Room χ/Q Analysis to Reg. Guide 1.194

Parameter	Discussion	Acceptable Input	Conform?	Comments
Surface Roughness Length, meters	ARCON96 uses the value of this parameter in adjusting wind speeds to account for differences in meteorological instrumentation height and release height.	Use a value of 0.2 in lieu of the default value of 0.1 for most sites. (Reasonable values range from 0.1 for sites with low surface vegetation to 0.5 for forest covered sites.)	No	Used the default value of 0.1 for all three representative sites. The difference in control room χ/Qs computed using 0.1 vs. 0.2 for surface roughness length is virtually negligible (typically <0.5%), with the χ/Qs for some standard averaging intervals being slightly higher and χ/Qs for other intervals being slightly lower.
Wind Direction Window, degrees Code Default	ARCON96 uses the value of this parameter and the Direction to Source to establish which range of wind directions should be included in the assessment of the χ/Q.	Use the default window of 90 degrees (45 degrees on either side of line of sight from the source to the receptor).	No	Used a window equal to the angle that would be affected by the building wake cloud, based on actual building layouts.
Minimum Wind Speed, meters/second Code Default	ARCON96 uses the value of this parameter to identify calm conditions.	Use the default wind speed of 0.5 m/s (regardless of the wind speed units entered earlier), unless there is some indication that the anemometer threshold is greater than 0.6 m/s.	Yes	Used the default wind speed of 0.5 m/s.
Averaging Sector Width Constant Code Default	ARCON96 uses the value of this parameter to prevent inconsistency between the centerline and sector average χ/Qs for wide plumes. Has largest effect on ground level plumes.	Although the default value is 4, a value of 4.3 is preferred. (A future revision to ARCON96 will change the default to 4.3)	No	Used the default value of 4.0. The difference in control room χ/Qs computed using 4.0 vs. 4.3 for the averaging sector width constant is virtually negligible. Typically, the χ/Qs for the 0-2 and 2-8 hour averaging intervals are identical, and the χ/Qs for the other intervals being as much as 10% higher, i.e., more conservative.

Table 2.3.4-1-1
Conformance of AP1000 Control Room χ/Q Analysis to Reg. Guide 1.194

Parameter	Discussion	Acceptable Input	Conform?	Comments
Initial Diffusion Coefficients, meters	ARCON96 uses these parameters in modeling a diffuse source.	These values will normally be set to zero. If the diffuse source option is being used, see Regulatory Position 2.2.4.	No	A diffuse (area) source was modeled for the releases Main Equipment Hatch and the Equipment Hatch at elevation 112'-0" when offsite power is not available. The initial horizontal and vertical diffusion coefficients for the Main Equipment Hatch release were set to the actual dimensions of the east Staging and Storage Area on elevation 135'-3" of the annex building, based on a conservative model of the potential release pathway. The initial horizontal and vertical diffusion coefficients for the release from the Equipment Hatch at elevation 112'-0" were set to the actual dimensions of a section of the auxiliary building wall in the vicinity of the sliding door into the annex building, based on a conservative model of the potential release pathway. These conservatively derived coefficients were not reduced by a factor of 6, as recommended by the Regulatory Guide, for either hatch. However, the AP600 / AP1000 approach also did not take credit for the entire area available for a "diffuse source" as discussed in the Regulatory Guide, which allows the entire periphery of the containment to be assumed as the area for a diffuse source.
Hours in Averages Code Default	The values of this parameter were selected to provide results for desired periods and to provide a smooth χ/Q curve.	Use the default values.	Yes	

Table 2.3.4-1-1
Conformance of AP1000 Control Room χ/Q Analysis to Reg. Guide 1.194

Parameter	Discussion	Acceptable Input	Conform?	Comments
Minimum Number of Hours Code Default	The default values of this parameter will allow processing with up to 10% missing data.	Use the default values.	Yes	

Table 2.3.4-1-2

Examples of Changes in Accident Dose Modeling

Loss-of-Coolant Accident
The aerosol removal coefficients that had been calculated for AP600 and conservatively applied to the AP1000 will be replaced by the values that have recently been generated specifically for the AP1000 design.
Credit will be taken for a 50% reduction in containment leak rate at 24 hours into the accident consistent with the guidance in Appendix A of Regulatory Guide 1.183 (currently the analysis assumes that design basis leakage continues for the duration of the accident).
Operation of the Control Room Habitability System will be assumed to be reestablished after seven days when support from offsite can be credited.
Steam Generator Tube Rupture and Main Steam Line Break
The assumed primary-to-secondary leak rate of 500 gpd per steam generator will be changed to 150 gpd per steam generator consistent with the Technical Specifications.
Rod Ejection
Credit will be taken for a 50% reduction in containment leak rate at 24 hours into the accident consistent with the guidance in Appendix A of Regulatory Guide 1.183 (currently the analysis assumes that design basis leakage continues for the duration of the accident).
The assumed primary-to-secondary leak rate of 500 gpd per steam generator will be changed to 150 gpd per steam generator consistent with the Technical Specifications.
Fuel Handling Accident
Dose analysis will be revised to use a shorter (24 hour versus pervious 100 hours) decay period to accommodate all credible refueling operations.
Locked Pump Rotor
The assumed primary-to-secondary leak rate of 500 gpd per steam generator will be changed to 150 gpd per steam generator consistent with the Technical Specifications.
The assumed level of fuel rod failures is reduced from 16% to 10% (analysis shows no fuel failures are expected).

2.3.2 Local Meteorology

The local meteorology is site specific and will be defined by the Combined License applicant.

2.3.3 Onsite Meteorological Measurement Programs

The onsite meteorological measurement program is site specific and will be defined by the Combined License applicant. The number and location of meteorological instrument towers are determined by actual site parameters.

2.3.4 Short-Term Diffusion Estimates

In the absence of a specific site for use in determining values for short-term diffusion, a study was performed to determine the atmospheric dispersion factors (X/Q values) that would envelope most current plant sites and that could be used to calculate the radiological consequences of design basis accidents. The X/Q values thus derived <u>for offsite</u> are provided in Table 2-1.

This set of <u>offsite</u> X/Q values is representative of potential sites for construction of the AP1000. The values are appropriate for analyses to determine the radiological consequences of accidents. These values were <u>selected to bound 80-90% of U.S. sites.</u> ~~determined using meteorological data representative of a 60-70th percentile U.S. site. The values were calculated following guidance in Regulatory Guide 1.145 considering ground release, building wake, and lateral plume meander under stable atmospheric conditions. Site selection is not restricted to those sites bounded by these X/Q values. If a selected site has X/Q values that exceed the reference site values, the accident doses reported in Chapter 15 would be adjusted to reflect the change in X/Q values.~~

<u>The X/Q values for the control room air intake or the door leading to the control room are dependent not only on the site meteorology but also on the plant design and layout. These X/Q values are addressed in Appendix 15A. Separate sets of X/Q values are identified for each combination of activity release location and receptor location.</u>

2.3.5 Long-Term Diffusion Estimates

The long-term diffusion estimates are site specific and will be provided by the Combined License applicant. The site boundary annual average X/Q shown in Table 2-1 is used to calculate release concentrations at the site boundary for comparison with the activity release limits defined in 10 CFR 20. The value specified is expected to bound atmospheric conditions at most U.S. sites. If a selected site has a X/Q value that exceeds this reference site value, the release concentrations reported in Section 11.3 would be adjusted proportionate to the change in X/Q.

2.3.6 Combined License Information

2.3.6.1 Regional Climatology

Combined License applicants referencing the AP1000 certified design will address site-specific information related to regional climatology.

Table 2-1 (Sheet 1 of 3~~2~~)

SITE PARAMETERS

Air Temperature	
Maximum Safety[a]	115°F dry bulb/80°F coincident wet bulb 81°F wet bulb (noncoincident)
Minimum Safety[a]	-40°F
Maximum Normal[b]	100°F dry bulb/77°F coincident wet bulb 80°F wet bulb (noncoincident)[d]
Minimum Normal[b]	-10°F
Wind Speed	
Operating Basis	145 mph (3 second gust); importance factor 1.15 (safety), 1.0 (nonsafety); exposure C; topographic factor 1.0
Tornado	300 mph
Seismic	
SSE	0.30g peak ground acceleration[c]
Fault Displacement Potential	None
Soil	
Average Allowable Static Bearing Capability	Greater than or equal to 8,600 lb/ft^2 over the footprint of the nuclear island at its excavation depth
Maximum Allowable Dynamic Bearing Capacity for Normal Plus SSE	Greater than or equal to 85,000 lb/ft^2 at the edge of the nuclear island at its excavation depth
Shear Wave Velocity	Greater than or equal to 8,000 ft/sec based on low-strain best-estimate soil properties over the footprint of the nuclear island at its excavation depth
Liquefaction Potential	None

Table 2-1 (Sheet 2 of ~~2~~3)

SITE PARAMETERS

Missiles	
Tornado	4000 - lb automobile at 105 mph horizontal, 74 mph vertical 275 - lb, 8 in. shell at 105 mph horizontal, 74 mph vertical 1 inch diameter steel ball at 105 mph horizontal and vertical
Flood Level	Less than plant elevation 100′
Ground Water Level	Less than plant elevation 98′
Plant Grade Elevation	Less than plant elevation 100′ except for portion at a higher elevation adjacent to the annex building
Precipitation	
Rain	19.4 in./hr (6.3 in./5 min)
Snow/Ice	75 pounds per square foot on ground with exposure factor of 1.0 and importance factors of 1.2 (safety) and 1.0 (non-safety)
Atmospheric Dispersion Values - X/Q[e]	
Site boundary (0-2 hr)	≤ TBD ~~0.60×10^{-3}~~ sec/m^3
Site boundary (annual average)	≤ 2.0×10^{-5} sec/m^3
Low population zone boundary	
0 - 8 hr	≤ TBD ~~1.35×10^{-4}~~-sec/m^3
8 - 24 hr	≤ TBD ~~1.0×10^{-4}~~-sec/m^3
24 - 96 hr	≤ TBD ~~5.4×10^{-5}~~-sec/m^3
96 - 720 hr	≤ TBD ~~2.2×10^{-5}~~-sec/m^3
Population Distribution	
Exclusion area (site)	0.5 mi

Notes:
(a) Maximum and minimum safety values are based on historical data and exclude peaks of less than 2 hours duration.
(b) Maximum and minimum normal values are the 1 percent exceedance magnitudes.
(c) With ground response spectra (at foundation level of nuclear island) as given in Figures 3.7.1-1 and 3.7.1-2.
(d) The noncoincident wet bulb temperature is applicable to the cooling tower only.
(e) For AP1000, the terms "site boundary" and "exclusion area boundary" are used interchangeably. Thus, the X/Q specified for the site boundary applies whenever a discussion refers to the exclusion area boundary.

Table 2-1 (Sheet 3 of 3)

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

(ALL CONTENT OF TABLE TO BE REVISED)

exhausted, the conditions are 87.2°F/41 percent. At 24 hours, when the 24 hour battery heat loads are terminated, the conditions are 84.4°F/45 percent. At 72 hours, the conditions are 85.8°F/39 percent.

Sufficient thermal mass is provided in the walls and ceiling of the main control room to absorb the heat generated by the equipment, lights, and occupants. The temperature in the instrumentation and control rooms and dc equipment rooms following a loss of the nuclear island nonradioactive ventilation system remains below acceptable limits as discussed in subsection 6.4.4. As in the main control room, sufficient thermal mass is provided surrounding these rooms to absorb the heat generated by the equipment. After 72 hours, the instrumentation and control rooms will be cooled by drawing in outside air and circulating it through the room, as discussed in subsection 6.4.2.2.

In the event of a loss of ac power, the nuclear island nonradioactive ventilation system isolation valves automatically close and the main control room emergency habitability system isolation valves automatically open. These actions protect the main control room occupants from a potential radiation release. In instances in which there is no radiological source term present, the compressed air storage tanks are refilled via a connection to the breathable quality air compressor in the compressed and instrument air system (CAS). The compressed air storage tanks can also be refilled from portable supplies by an installed connection in the CAS.

6.4.4 System Safety Evaluation

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

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

15.1.5.2.4 Margin to Critical Heat Flux

The case presented in subsection 15.1.5.2.2 conservatively models the expected behavior of the plant during a steam system piping failure. This includes the tripping of the reactor coolant pumps coincident with core makeup tank actuation. A DNB analysis is performed using limiting assumptions that bound those of subsection 15.1.5.2.2.

Under the low flow (natural circulation) conditions present in the AP1000 transient, the return to power is severely limited by the large negative feedback due to flow and power. The minimum DNBR is conservatively calculated and is above the 95/95 limit.

15.1.5.3 Conclusions

The analysis shows that the DNB design basis is met for the steam system piping failure event. DNB and possible cladding perforation following a steam pipe rupture are not precluded by the criteria. The preceding analysis shows that no DNB occurs for the main steam line rupture assuming the most reactive RCCA stuck in its fully withdrawn position.

15.1.5.4 Radiological Consequences

The evaluation of the radiological consequences of a postulated main steam line break outside containment assumes that the reactor has been operating with the design basis fuel defect level (0.25 percent of power produced by fuel rods containing cladding defects) and that leaking steam generator tubes have resulted in a buildup of activity in the secondary coolant.

Following the rupture, startup feedwater to the faulted loop is isolated and the steam generator is allowed to steam dry. Any radioiodines carried from the primary coolant into the faulted steam generator via leaking tubes are assumed to be released directly to the environment. It is conservatively assumed that the reactor is cooled by steaming from the intact loop.

15.1.5.4.1 Source Term

The only significant radionuclide releases due to the main steam line break are the iodines and alkali metals that become airborne and are released to the environment as a result of the accident. Noble gases are also released to the environment. Their impact is secondary because any noble gases entering the secondary side during normal operation are rapidly released to the environment. ~~The increase in the releases of noble gases beyond the levels associated with normal operation is due to the assumed increase in primary to secondary leakage from the Tech Spec limit of 150 gpd per steam generator to 500 gpd per steam generator.~~

The analysis considers two different reactor coolant iodine source terms, both of which consider the iodine spiking phenomenon. In one case, the initial iodine concentrations are assumed to be those associated with equilibrium operating limits for primary coolant iodine activity. The iodine spike is assumed to be initiated by the accident with the spike causing an increasing level of iodine in the reactor coolant.

The second case assumes that the iodine spike occurs prior to the accident and that the maximum resulting reactor coolant iodine concentration exists at the time the accident occurs.

The reactor coolant noble gas and alkali metal concentrations are assumed to be those associated with the design basis fuel defect level.

The secondary coolant is assumed to have an iodine source term of 0.1 μCi/g dose equivalent I-131. This is 10 percent of the maximum primary coolant activity at equilibrium operating conditions. The secondary coolant alkali metal concentration is also assumed to be 10% of the primary concentration.

15.1.5.4.2 Release Pathways

There are three components to the accident releases:

- The secondary coolant in the steam generator of the faulted loop is assumed to be released out the break as steam. Any iodine and alkali metal activity contained in the coolant is assumed to be released.

- The reactor coolant leaking into the steam generator of the faulted loop is assumed to be released to the environment without any credit for partitioning or plateout onto the interior of the steam generator.

- The reactor coolant leaking into the steam generator of the intact loop would mix with the secondary coolant and thus raise the activity concentrations in the secondary water. While the steam release from the intact loop would have partitioning of non-gaseous activity, this analysis conservatively assumes that any activity entering the secondary side is released.

Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.1.5.4.3 Dose Calculation Models

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

15.1.5.4.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.1.5-1.

15.1.5.4.5 Identification of Conservatisms

The assumptions and parameters used in the analysis contain a number of significant conservatisms:

- The reactor coolant activities are based on a fuel defect level of 0.25 percent. The expected fuel defect level is far less than this (see Section 11.1).

- The assumed leakage of 150 ~~500~~ gallons of reactor coolant per day into each steam generator is conservative. The leakage is expected to be a small fraction of this during normal operation.

- The conservatively selected meteorological conditions are present only rarely.

15.1.5.4.6 Doses

Using the assumptions from Table 15.1.5-1, the calculated total effective dose equivalent (TEDE) doses for the case with accident-initiated iodine spike are determined to be less than TBD ~~0.8~~ rem at the site boundary and TBD ~~1.7~~ rem at the low population zone outer boundary. These doses are small fractions of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is defined, consistent with the Standard Review Plan, as being 10 percent or less. The TEDE doses for the case with pre-existing iodine spike are determined to be less than TBD ~~0.7~~ rem at the site boundary and TBD ~~0.5~~ rem at the low population zone outer boundary. These doses are within the dose guidelines of 10 CFR Part 50.34.

At the time the main steam line break occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE. When this is added to the dose calculated for the main steam line break, the resulting total dose remains less than the values reported above.

15.1.6 Inadvertent Operation of the PRHR Heat Exchanger

15.1.6.1 Identification of Causes and Accident Description

The inadvertent actuation of the PRHR heat exchanger causes an injection of relatively cold water into the reactor coolant system. This produces a reactivity insertion in the presence of a negative moderator temperature coefficient. The overpower/overtemperature protection functions (neutron overpower, overtemperature, and overpower ΔT trips) are intended to prevent a power increase which could lead to a DNBR less than the safety analysis limit. In addition, because the cold leg temperature is reduced which depressurizes the reactor coolant system during this event, the low cold leg temperature or low pressurizer pressure protection functions could generate a reactor trip. These protection functions do not terminate operation of the PRHR heat exchanger.

The inadvertent actuation of the PRHR heat exchanger could be caused by operator error or a false actuation signal. Actuation of the PRHR heat exchanger involves opening one of the isolation valves, which establishes a flow path from one reactor coolant system hot leg, through the PRHR heat exchanger, and back into its associated steam generator cold leg plenum.

The PRHR heat exchanger is located above the core to promote natural circulation flow when the reactor coolant pumps are not operating. With the reactor coolant pumps in operation, flow through the PRHR heat exchanger is enhanced. The heat sink for the PRHR heat exchanger is provided by the IRWST, in which the PRHR heat exchanger is submerged. Because the fluid in the heat

Table 15.1.5-1

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK

Reactor coolant iodine activity	
– Accident-initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Appendix 15A)
– Preaccident spike	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	10% of reactor coolant concentrations at maximum equilibrium conditions
Duration of accident (hr)	72
Atmospheric dispersion (χ/Q) factors	See Table 15A-5 in Appendix 15A
Steam generator in faulted loop	
– Initial water mass (lb)	3.03 E+05
– Primary to secondary leak rate (lb/hr)	52.14 ~~175~~[a]
– Iodine partition coefficient	1.0
– Steam released (lb) 0 - 2 hr 2 - 72 hr	 3.03 E+05 1.225 E+04
Steam generator in intact loop	
– Primary to secondary leak rate (lb/hr)	52.14 ~~175~~[a]
– Iodine partition coefficient	1.0
– Steam released (lb) 0 - 2 hr 2 - 72 hr	 3.0335 E+05 1.225 E+04
Nuclide data	See Table 15A-4

Note:
a. Equivalent to 150 ~~500~~ gpd cooled liquid at 62.4 lb/ft^3.

Code, Section III. Also, the peak cladding surface temperature is considerably less than 2700°F. The cladding temperature is conservatively calculated, assuming that DNB occurs at the initiation of the transient. These results represent the most limiting conditions with respect to the locked rotor event or the pump shaft break.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1. With the reactor tripped, a stable plant condition is eventually attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated locked reactor coolant pump rotor accident assumes that the reactor has been operating with the design basis fuel defect level (0.25 percent of power produced by fuel rods containing cladding defects) and that leaking steam generator tubes have resulted in a buildup of activity in the secondary coolant.

As a result of the accident, it is determined that no fuel rods are damaged such that the activity contained in the fuel-cladding gap is released to the reactor coolant. However, a conservative analysis has been performed assuming 10% ~~16%~~ of the rods are damaged. Activity carried over to the secondary side because of primary-to-secondary leakage is available for release to the environment via the steam line safety valves or the power-operated relief valves.

15.3.3.3.1 Source Term

The significant radionuclide releases due to the locked rotor accident are the iodines, alkali metals (cesiums, rubidiums) and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The initial reactor coolant noble gas and alkali metal concentrations are assumed to be those associated with the design basis fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of the gap inventory of fission products from the portion of the core assumed to fail because of the accident.

Based on NUREG-1465 (Reference 6), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fraction is increased to 8 percent of the inventory for I-131, 10 percent for Kr-85, 5 percent for other iodines and noble gases and 12 percent for alkali metals. Also, to address the fact that the failed fuel rods may have been operating at power levels above the core average, the source term is increased by the lead rod radial peaking factor.

The initial secondary coolant activity is assumed to be 10 percent of the maximum equilibrium primary coolant activity for iodines and alkali metals.

15.3.3.3.2 Release Pathways

There are two components to the accident releases:

- The activity initially in the secondary coolant is available for release as long as steam releases continue.

- The reactor coolant leaking into the steam generators is assumed to mix with the secondary coolant. The activity from the primary coolant mixes with the secondary coolant. As steam is released, a portion of the iodine and alkali metal activity in the coolant is released. The fraction of activity released is defined by the assumed flashing fraction and the partition coefficient assumed for the steam generator. The noble gas activity entering the secondary side is released to the environment. These releases are terminated when the steam releases stop.

Credit is taken for the decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.3.3.3.3 Dose Calculation Models

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

15.3.3.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.3-3.

15.3.3.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of significant conservatisms:

- Although fuel damage is assumed to occur as a result of the accident, no fuel damage is anticipated.

- The reactor coolant activities are based on a fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less than this (see Section 11.1).

- The leakage of reactor coolant into the secondary system, at 300 ~~1000~~ gallons per day, is conservative. The leakage is normally a small fraction of this.

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

15.3.3.3.6 Doses

Using the assumptions from Table 15.3-3, the calculated total effective dose equivalent (TEDE) doses are determined to be less than TBD ~~2.5~~ rem at the exclusion area boundary and less than TBD ~~0.6~~ rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is identified as 10 percent or less consistent with the Standard Review Plan (Reference 4).

At the time the locked reactor coolant pump rotor event occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site

Table 15.3-3

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/gm of dose equivalent I-131 (see Appendix 15A)[a]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/gm dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	10% of design basis reactor coolant concentrations at maximum equilibrium conditions
Fraction of fuel rods assumed to fail	0.10 ~~0.16~~
Core activity	See Table 15A-3
Radial Peaking Factor (for determination of activity in failed fuel rods)	1.65
Fission product gap fractions	
I-131	0.08
Kr-85	0.10
Other iodines and noble gases	0.05
Alkali metals	0.12
Reactor coolant mass (lb)	3.7 E+05
Secondary coolant mass (lb)	6.06 E+05
Condenser	Not available
Duration of accident (hr)	1.5 hr
Atmospheric dispersion factors	See Table 15A-5
Primary to secondary leak rate (lb/hr)	104.3 ~~350~~[b]
Steam released (lb)	
0-1.5 hours[c]	6.48 E+05
Partition coefficient in steam generators for iodine and alkali metals	0.01
Leak flashing fraction[d]	
0-60 minutes	0.04
> 60 minutes	0

Note:

a. The assumption of a pre-existing iodine spike is a conservative assumption for the initial reactor coolant activity. However, compared to the activity released to the coolant from the assumed fuel failures, it is not significant.

b. Equivalent to 300 ~~1000~~ gpd cooled liquid at 62.4 lb/ft^3.

c. Heat removal is achieved by steaming and by passive core cooling system operation in the limiting case where the startup feedwater system is not available. When heat removal by the passive core cooling system exceeds the decay heat load, steam releases are terminated.

d. No credit for iodine partitioning is taken for flashed leakage. Flashing is terminated by the passive core cooling system operation reducing the RCS below the saturation temperature of the secondary.

As a result of the accident, 10 percent of the fuel rods are assumed to be damaged (see subsection 15.4.8.2.1.8) such that the activity contained in the fuel-cladding gap is released to the reactor coolant. In addition, a small fraction of fuel is assumed to melt and release core inventory to the reactor coolant.

Activity released to the containment via the spill from the reactor vessel head is assumed to be available for release to the environment because of containment leakage. Activity carried over to the secondary side due to primary-to-secondary leakage is available for release to the environment through the steam line safety or power-operated relief valves.

15.4.8.3.1 Source Term

The significant radionuclide releases due to the rod ejection accident are the iodines, alkali metals, and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The initial reactor coolant noble gas and alkali metal concentrations are assumed to be those associated with the design fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of fission products from the portion of the core assumed to fail.

Based on NUREG-1465 (Reference 12), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fraction is increased to 10 percent of the inventory for iodine and noble gases and 12 percent for alkali metals. Also, to address the fact that the failed fuel rods may have been operating at power levels above the core average, the source term is increased by the lead rod radial peaking factor.

Even though no fuel centerline melting is expected, a conservative upper limit for fuel melting was determined to be 0.25 percent of the core based on the following assumptions:

1. No more than 50 percent of the rods experiencing clad damage will experience centerline melting. (Based on 10 percent of rods failing, this is 5 percent of the core.)

2. Due to the power distribution within the core, no more than 50 percent of the axial length of the affected fuel rods will experience melting. (This reduces the equivalent number of rods experiencing melting to 2.5 percent of the core.)

3. Of rods experiencing centerline melting, only a conservative maximum of the innermost 10 percent of the fuel volume will actually melt. (Based on 2.5 percent of the rods experiencing melting, the resulting fraction of the core experiencing melting is 0.25 percent.)

All of the noble gases and half of the iodines and alkali metals are assumed to be released from the melted fuel.

The initial secondary coolant activity is assumed to be 10 percent of the maximum equilibrium primary coolant activity for iodines and alkali metals.

15.4.8.3.2 Release Pathways

There are three components to the accident releases:

- The activity initially in the secondary coolant is available for release as long as steam releases continue.

- The reactor coolant leaking into the steam generators is assumed to mix with the secondary coolant. The activity from the primary coolant mixes with the secondary coolant and, as steam is released, a portion of the iodine and alkali metal in the coolant is released. The fraction of activity released is defined by the assumed flashing fraction and the partition coefficient assumed for the steam generator. The noble gas activity entering the secondary side is released to the environment. These releases are terminated when the steam releases stop.

- The activity from the reactor coolant system and the core is released to the containment atmosphere and is available for leakage to the environment through the assumed design basis containment leakage.

Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.4.8.3.3 Dose Calculation Models

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

15.4.8.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.4-4.

15.4.8.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of conservatisms:

- Although fuel damage is assumed to occur as a result of the accident, no fuel damage is anticipated.

- The reactor coolant activities are based on an assumed fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less than this (see Section 11.1).

- The leakage of reactor coolant into the secondary system, at 300 ~~1000~~ gallons per day, is conservative. The leakage is normally a small fraction of this.

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

- The leakage from containment is assumed to continue for a full 30 days. It is expected that containment pressure is reduced to the point that leakage is negligible before this time.

15.4.8.3.6 Doses

Using the assumptions from Table 15.4-4, the calculated total effective dose equivalent (TEDE) doses are determined to be less than TBD ~~3~~ rem at the site boundary and less than TBD ~~2~~ rem at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem total effective dose equivalent identified in 10 CFR Part 50.34. The phrase "well within" is taken as being 25 percent or less.

At the time the rod ejection accident occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE, and when this is added to the dose calculated for the rod ejection accident, the resulting total dose remains less than 2 rem TEDE.

15.4.9 Combined License Information

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

15.4.10 References

1. Barry, R. F., and Risher, D. H., Jr., "TWINKLE--A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Nonproprietary), January 1975.

2. Hargrove, H. G., "FACTRAN--A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908 (Proprietary) and WCAP-7337 (Nonproprietary), June 1972.

3. Burnett. T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

4. Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1A, January 1975.

5. Taxelius, T. G., ed, "Annual Report-SPERT Project, October 1968, September 1969," Idaho Nuclear Corporation, IN-1370, June 1970.

6. Liimataninen, R. C., and Testa, F. J., "Studies in TREAT of Zircaloy-2-Clad, UO_2-Core Simulated Fuel Elements," ANL-7225, January-June 1966, p 177, November 1966.

7. Davidson, S. L., (Ed.), et al., "ANC: Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, (Proprietary) September 1986.

Table 15.4-4 (Sheet 2 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A ROD EJECTION ACCIDENT

Condenser	Not available
Duration of accident (days)	30
Atmospheric dispersion (χ/Q) factors	See Table 15A-5 in Appendix 15A
Secondary system release path	
– Primary to secondary leak rate (lb/hr)	104.3 ~~350~~[a]
– Leak flashing fraction (%)	4.0[b]
– Secondary coolant mass (lb)	6.06 E+05
– Duration of steam release from secondary system (sec)	1800
– Steam released from secondary system (lb)	1.08 E+05
– Partition coefficient in steam generators	
Iodine	0.01
Alkali metals	0.001
Containment leakage release path	
– Containment leak rate (% per day)	~~0.10~~
• 0-24 hr	0.10
• > 24 hr	0.05
– Airborne activity removal coefficients (hr^{-1})	
Elemental iodine	1.7[c]
Organic iodine	0
Particulate iodine or alkali metals	0.1
– Decontamination factor limit for elemental iodine removal	200
– Time to reach the decontamination factor limit for elemental iodine (hr)	3.1

Notes:

a. Equivalent to 300 ~~1000~~ gpd cooled liquid at 62.4 lb/ft^3.

b. No credit for iodine partitioning is taken for flashed leakage.

c. From Appendix 15B.

15.6.2.1 Source Term

The only significant radionuclide releases are the iodines and the noble gases. The analysis assumes that the reactor coolant iodine is at the maximum Technical Specification level for continuous operation. In addition, it is assumed that an iodine spike occurs at the time of the accident. The reactor coolant noble gas activities are assumed to be those associated with the design basis fuel defect level.

15.6.2.2 Release Pathway

The reactor coolant that is spilled from the break is assumed to be at high temperature and pressure. A large portion of the flow flashes to steam, and the iodine in the flashed liquid is assumed to become airborne.

The iodine and noble gases are assumed to be released directly to the environment with no credit for depletion, although a large fraction of the airborne iodine is expected to deposit on building surfaces. No credit is assumed for radioactive decay after release.

15.6.2.3 Dose Calculation Models

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

15.6.2.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.6.2-1.

15.6.2.5 Identification of Conservatisms

The assumptions used contain the following significant conservatisms:

- The reactor coolant activities are based on a fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less than this (see Section 11.1).
- It is unlikely that the conservatively selected meteorological conditions would be present at the time of the accident.

15.6.2.6 Doses

Using the assumptions from Table 15.6.2-1, the calculated total effective dose equivalent (TEDE) doses are determined to be < TBD ~~1.3~~ rem at the exclusion area boundary and < TBD ~~0.3~~ rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase "a small fraction" is taken as being ten percent or less.

At the time the accident occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose

15.6.3.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of significant conservatisms, such as:

- The reactor coolant activities are based on a fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less (see Section 11.1).
- It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

15.6.3.3.6 Doses

Using the assumptions from Table 15.6.3-3, the calculated TEDE doses for the case in which the iodine spike is assumed to be initiated by the accident are determined to be less than TBD ~~1.5~~ rem at the exclusion area boundary and less than TBD ~~0.3~~ rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is defined, consistent with the Standard Review Plan, as being ten percent or less.

For the case in which the SGTR is assumed to occur coincident with a pre-existing iodine spike, the TEDE doses are determined to be less than TBD ~~3.0~~ rem at the exclusion area boundary and less than TBD ~~0.5~~ rem at the low population zone outer boundary. These doses are within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34.

At the time the accident occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour exclusion area boundary dose because pool boiling would not occur until after 2.0 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the doses calculated for the steam generator tube rupture, the resulting total doses remain as reported above.

15.6.3.4 Conclusions

The results of the SGTR analysis show that the overfill protection logic and the passive system design features provide protection to prevent steam generator overfill. Following an SGTR accident, the operators can identify and isolate the faulted steam generator and complete the required actions to terminate the primary-to-secondary break flow before steam generator overfill or ADS actuation occurs.

Even when no operator actions are assumed, the AP1000 protection system and passive design features initiate automatic actions that can terminate a steam generator tube leak and stabilize the reactor coolant system in a safe condition while preventing steam generator overfill and ADS actuation.

The dose calculations take into account the release of activity by way of the containment purge line prior to its isolation near the beginning of the accident and the release of activity resulting from containment leakage. Purge of the containment for hydrogen control is not an intended mode of operation and is not considered in the dose analysis. While the normal residual heat removal system is capable of post-LOCA cooling, it is not a safety-related system and may not be available following the accident. If it is operable, it would be used only if the source term is not far above the normal shutdown primary coolant source term. It is assumed that core cooling is accomplished by the passive core cooling system, which does not pass coolant outside of containment. Thus, there is no recirculation leakage release path to be modeled.

15.6.5.3.1 Source Term

The release of activity to the containment consists of two parts. The initial release is the activity contained in the reactor coolant system. This is followed by the release of core activity.

15.6.5.3.1.1 Primary Coolant Release

The reactor coolant is assumed to have activity levels consistent with operation at the ~~design basis fuel defect level of 0.25 percent. The noble gas source term is listed in Table 11.1-2, and the maximum iodine spike source term is detailed in Table 15A-1 in Appendix 15A.~~ Technical Specification limits of 280 μCi/gm dose equivalent Xe-133 and 60 μCi/gm dose equivalent I-131.

Based on NUREG-1465 (Reference 19), for a plant using leak-before-break methodology, the release of coolant into the containment can be assumed to last for 10 minutes. The AP1000 is a leak-before-break plant, and the water in the reactor coolant system is assumed to blow down into the containment over a period of 10 minutes. The flow rate is assumed to be constant over the 10-minute period. As the reactor coolant enters the containment, the noble gases and half of the iodine activity are assumed to be released into the containment atmosphere.

15.6.5.3.1.2 Core Release

The release of activity from the fuel takes place in two stages as summarized in Table 15.6.5-1. First is the gap release which is assumed to occur at the end of the primary coolant release phase (i.e., at ten minutes into the accident) and continue over a period of half an hour. The second stage is that of the in-vessel core melt in which the bulk of the activity releases associated with the accident occur. The source term model is based on NUREG-1465 and Regulatory Guide 1.183 (Reference 20).

The core fission product inventory at the time of the accident is based on operation near the end of a fuel cycle at 102-percent power and is provided in Table 15A-3 of Appendix 15A. Consistent with NUREG-1465, there are three groups of nuclides considered in the gap activity releases: noble gases, iodines, and alkali metals (cesium and rubidium). For the core melt phase, there are five additional nuclide groups for a total of eight. The five additional nuclide groups are the tellurium group, the noble metals group, the cerium group, the lanthanide group, and barium and strontium. The specific nuclides included in the source term are as shown in Table 15A-3.

Elemental iodine is removed by deposition onto surfaces. Particulates are removed by sedimentation, diffusiophoresis (deposition driven by steam condensation), and thermophoresis (deposition driven by heat transfer). No removal of organic iodine is assumed. Appendix 15B provides a discussion of the models and assumptions used in calculating the removal coefficients.

15.6.5.3.3 Release Pathways

The release pathways are the containment purge line and containment leakage. The activity releases are assumed to be ground level releases.

During the initial part of the accident, before the containment is isolated, it is assumed that containment purge is in operation and that activity is released through this pathway until the purge valves are closed. No credit is taken for the filters in the purge exhaust line.

The majority of the releases due to the LOCA are the result of containment leakage. The containment is assumed to leak at its design leak rate.

15.6.5.3.4 Offsite Dose Calculation Models

The offsite dose calculation models are provided in Appendix 15A. The models address the determination of the TEDE doses from the combined acute doses and the committed effective dose equivalent doses.

The exclusion area boundary dose is calculated for the 2-hour period over which the highest doses would be accrued by an individual located at the exclusion area boundary. Because of the delays associated with the core damage for this accident, the first 2 hours of the accident are not the worst 2-hour interval for accumulating a dose.

The low population zone boundary dose is calculated for the nominal 30-day duration of the accident.

For both the exclusion area boundary and low population zone dose determinations, the calculated doses are compared to the dose guideline of 25 rem TEDE from 10 CFR Part 50.34.

15.6.5.3.5 Main Control Room Dose Model

There are two approaches that may be used for modeling the activity entering the main control room. If power is available, the normal heating, ventilation, and air-conditioning (HVAC) system will switch over to a supplemental filtration mode (Section 9.4). The normal HVAC system is not a safety-class system but provides defense in depth.

Alternatively, if the normal HVAC is inoperable or, if operable, the supplemental filtration train does not function properly resulting in increasing levels of airborne iodine in the main control room, the emergency habitability system (Section 6.4) would be actuated when high iodine activity is detected. The emergency habitability system provides passive pressurization of the main control room from a bottled air supply to prevent inleakage of contaminated air to the main control room. There is a 72-hour supply of air in the emergency habitability system. After this time, the main

control room is assumed to be opened and unfiltered air is drawn into the main control room by way of an ancillary fan. After seven days, offsite support is assumed to be available to reestablish operability of the control room habitability system by replenishing the compressed air supply or to bring the normal control room HVAC into operation with the supplemental filtration train.

The second approach, with the emergency habitability system in use, results in the more conservative determination of doses.

The main control room is accessed by a vestibule entrance which restricts the volume of contaminated air that can enter the main control room from ingress and egress. The equivalent inflow of unfiltered air due to expected ingress/egress has been determined to be 5.0 cfm.

Activity entering the main control room is assumed to be uniformly dispersed. No credit is taken for the removal of airborne activity in the main control room although elemental iodine and particulates would be removed by deposition and sedimentation.

The main control room dose calculation models are provided in Appendix 15A for the determination of doses resulting from activity which enters the main control room envelope.

15.6.5.3.6 Analytical Assumptions and Parameters

The analytical assumptions and parameters used in the radiological consequences analysis are listed in Table 15.6.5-2.

15.6.5.3.7 Identification of Conservatisms

The LOCA radiological consequences analysis assumptions include a number of conservatisms. Some of these conservatisms are discussed in the following subsections.

15.6.5.3.7.1 Primary Coolant Source Term

The source term is based on operation with the design fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less.

15.6.5.3.7.2 Core Release Source Term

The assumed core melt is a major conservatism associated with the analysis. In the event of a postulated LOCA, no major core damage is expected. Release of activity from the core is limited to a fraction of the core gap activity.

15.6.5.3.7.3 Atmospheric Dispersion Factors

The atmospheric dispersion factors assumed to be present during the course of the accident are conservatively selected. Actual meteorological conditions are expected to result in significantly higher dispersion of the released activity.

15.6.5.3.8 LOCA Doses

15.6.5.3.8.1 Offsite Doses

The doses calculated for the exclusion area boundary and the low population zone boundary are listed in Table 15.6.5-3. The doses are within the 10 CFR 50.34 dose guideline of 25 rem TEDE.

The reported exclusion area boundary doses are for the time period of TBD ~~1.0 to 3.0~~ hours. This is the 2-hour interval that has the highest calculated doses. The dose that would be incurred over the first 2 hours of the accident is well below the reported dose.

At the time the LOCA occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because pool boiling would not occur until after 2.5 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the dose calculated for the LOCA, the resulting total dose remains less than that reported in Table 15.6.5-3.

15.6.5.3.8.2 Doses to Operators in the Main Control Room

The doses calculated for the main control room personnel due to airborne activity entering the main control room are listed in Table 15.6.5-3. Also listed on Table 15.6.5-3 are the doses due to direct shine from the activity in the adjacent buildings and sky-shine from the radiation that streams out the top of the containment shield building and is reflected back down by air-scattering. The total of the three dose paths is within the dose criteria of 5 rem TEDE as defined in GDC 19.

As discussed above for the offsite doses, there is the potential for a dose to the operators in the main control room due to iodine releases from postulated spent fuel boiling. The calculated dose from this source is less than 0.01 rem TEDE and, when this is added to the dose calculated for the LOCA, the resulting total dose remains less than that reported in Table 15.6.5-3.

15.6.5.4 Core and System Performance

Subsection 15.6.5.4A describes the large-break LOCA analysis methodology and results. Subsections 15.6.5.4B.1.0 through 15.6.5.4B.4.0 describe the small-break LOCA analysis methodology and results.

15.6.5.4A Large-break LOCA Analysis Methodology and Results

Westinghouse applies the WCOBRA/TRAC computer code to perform best-estimate large-break LOCA analyses in compliance with 10 CFR 50 (Reference 5). WCOBRA/TRAC is a thermal-hydraulic computer code that calculates realistic fluid conditions in a PWR during the blowdown and reflood of a postulated large-break LOCA. The methodology used for the AP1000 analysis is documented in WCAP-12945-P-A and WCAP-14171, Revision 2 (References 10 and 11).

The NRC staff has reviewed and approved the best-estimate LOCA methodology documented in Reference 10 for estimating the 95th percentile PCT (Reference 8) for three-loop and four-loop

Table 15.6.3-3

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE

Reactor coolant iodine activity	
– Accident initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 µCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 335 (see Appendix 15A)
– Preaccident spike	An assumed iodine spike that results in an increase in the reactor coolant activity to 60 µCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	280 µCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal	10% of reactor coolant concentrations at maximum equilibrium conditions
Reactor coolant mass (lb)	3.84 E+05
Offsite power	Lost on reactor trip
Condenser	Lost on reactor trip
Time of reactor trip	Beginning of the accident
Duration of steam releases (hr)	13.19
Atmospheric dispersion factors	See Appendix 15A
Nuclide data	See Appendix 15A
Steam generator in ruptured loop	
– Initial secondary coolant mass (lb)	1.66 E+05
– Primary-to-secondary break flow	See Figure 15.6.3-5
– Flashing fraction for break flow	See Figure 15.6.3-10
– Steam released (lb)	See Table 15.6.3-2
– Iodine partition coefficient	1.0 E-02[a]
Steam generator in intact loop	
– Initial secondary coolant mass (lb)	2.00 E+05
– Primary-to-secondary leak rate (lb/hr)	52.14 ~~175~~[b]
– Steam released (lb)	See Table 15.6.3-2
– Iodine partition coefficient	1.0 E-02

Notes:

a. Iodine partition coefficient does not apply to flashed break flow.

b. Equivalent to 150 ~~500~~ gpd at psia cooled liquid at 62.4 lb/ft^3.

Table 15.6.5-2 (Sheet 1 of 3)

ASSUMPTIONS AND PARAMETERS USED IN CALCULATING RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Primary coolant source data	
– Noble gas concentration	280 µCi/g dose equivalent Xe-133
– Iodine concentration	1.0 µCi/g dose equivalent I-131
– Primary coolant mass (lb)	3.72 E+05
Containment purge release data	
– Containment purge flow rate (cfm)	8800
– Time to isolate purge line (seconds)	30
– Time to blow down the primary coolant system (minutes)	10
– Fraction of primary coolant iodine that becomes airborne	0.5
Core source data	
– Core activity at shutdown	See Table 15A-3
– Release of core activity to containment atmosphere (timing and fractions)	See Table 15.6.5-1
– Iodine species distribution (%)	
• Elemental	4.85
• Organic	0.15
• Particulate	95
Containment leakage release data	
– Containment volume (ft^3)	2.06 E+06
– Containment leak rate, 0-24 hr (% per day)	0.10
– Containment leak rate, >24 hr (% per day)	0.05
– Elemental iodine deposition removal coefficient (hr^{-1})	1.7
– Decontamination factor limit for elemental iodine removal	200
– Removal coefficient for particulates (hr^{-1})	See Appendix 15B
Main control room model	
– Main control room volume (ft^3)	35,700
– Volume of HVAC, including main control room and technical center (ft^3)	105,500
– Initial interval (prior to actuation of emergency habitability system)	

• Air intake flow (cfm)	1925
• Filter efficiency	Not applicable

Table 15.6.5-2 (Sheet 2 of 3)

ASSUMPTIONS AND PARAMETERS USED IN CALCULATING RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Main control room model (cont.)

– Main control room activity level at which the emergency habitability system is actuated (Ci/m^3 of dose equivalent I-131)	2.0E-6
– Time at which emergency habitability system is actuated (hr)	TBD ~~0.2622~~
– Interval with operation of the emergency habitability system	
• Flow from compressed air bottles of the emergency habitability system (cfm)	60
• Unfiltered inleakage (cfm)	5.0
• Recirculation flow (cfm)	Not applicable
– Time at which the compressed air supply of the emergency habitability system is depleted (hr)	72
– After depletion of emergency habitability system bottled air supply (>72 hr)	
• Air intake flow (cfm)	1700
• Intake flow filter efficiency (%)	Not applicable
• Recirculation flow (cfm)	Not applicable
– Time at which the emergency habitability system is returned to operation (hr)	168
– Atmospheric dispersion factors (sec/m^3) - from Table 15A-6~~5~~	
• 0 - TBD ~~0.2622~~ hr	TBD ~~1.2E-3~~ (at HVAC intake)
• TBD~~0.2622~~ - 2 hr	TBD ~~6.6E-4~~ (at entrance)
• 2 - 8 hr	TBD ~~3.8E-4~~ (at entrance)
• 8 - 24 hr	TBD ~~1.9E-4~~ (at entrance)
• 24 - 72 hr	TBD ~~1.8E-4~~ (at entrance)
• 72 - 96 hr	TBD ~~3.0E-4~~ (at HVAC intake)
• 96 - 168 ~~720~~ hr	TBD ~~2.6E-4~~ (at HVAC intake)
• 168 - 720 hr	TBD (at entrance)
– Occupancy	
• 0 - 24 hr	1.0
• 24 - 96 hr	0.6
• 96 - 720 hr	0.4
– Breathing rate (m^3/sec)	3.5 E-04

Table 15.6.5-3

RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT WITH CORE MELT

	TEDE Dose (rem)
Exclusion zone boundary dose (TBD ~~1.0 – 3.0~~ hr)[1]	TBD ~~24.8~~
Low population zone boundary dose (0 - 30 days)	TBD ~~10.7~~
Main control room dose (emergency habitability system in operation)	
– Airborne activity entering the main control room	TBD ~~4.4~~ rem
– Direct radiation from adjacent structures	0.15 rem
– Sky-shine	0.01 rem
– Total	TBD ~~4.56~~ rem

Note:
1. This is the 2-hour period having the highest dose.

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 24 hours decay, which bounds any credible refueling operation.

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 TBD ~~2.4~~ rem TEDE at the site boundary and TBD ~~0.6~~ 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.

~~Additionally, an evaluation was performed to determine the impact of a fuel handling accident occurring with less than 100 hours of fission product decay. The evaluation assumed that the fuel handling accident occurred after only 24 hours of decay. The resulting doses are <4.3 rem TEDE at the site boundary and <1.0 rem TEDE at the low population zone outer boundary. These doses remain well within the dose guideline of 25 rem TEDE.~~

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)	3468
– Decay time (hr)	24 ~~100~~
Core source term after 24 ~~100~~ hours decay (Ci)	
I-130	TBD ~~1.34 E+ 04~~
I-131	TBD ~~6.90 E+ 07~~
I-132	TBD ~~5.85 E+ 07~~
I-133	TBD ~~7.26 E+ 06~~
I-135	TBD ~~5.19 E+ 03~~
Kr-85m	TBD ~~5.10 E+ 00~~
Kr-85	TBD ~~1.05 E+ 06~~
Kr-88	TBD
Xe-131m	TBD ~~1.02 E+ 06~~
Xe-133m	TBD ~~2.45 E+ 06~~
Xe-133	TBD ~~1.32 E+ 08~~
Xe-135m	TBD ~~8.31 E+ 02~~
Xe-135	TBD ~~2.52 E+ 05~~
Number of fuel assemblies in core	157
Amount of fuel damage	One assembly
Maximum rod radial peaking factor	1.65
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

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 ~~were calculated~~ 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 χ/Q values is identified for each potential activity release location that has been identified and the two control room receptor locations. The χ/Q values have been selected in concert with the design basis accident radiological consequences analyses to obtain limiting values. In this manner the maximum acceptable χ/Q values consistent with meeting dose acceptance criteria have been obtained. These χ/Q values are listed in Table 15A-6.

Combined License Applicants referencing the AP1000 certified design will confirm that the site-specific χ/Q values are 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, the Combined License Applicant will address 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. The Combined License Applicant should consider 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. 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 that the Combined License Applicant should use when determining the site-specific control room χ/Q values using the ARCON96 code (References 4 and 5). ~~As recommended by the NRC, the calculation of χ/Q values was performed using the computer code ARCON96 (Reference 4), which provides a time-based building wake model for the determination of χ/Q. The χ/Q values were calculated using three separate meteorological data bases to encompass a range of potential plant sites. Three existing power plant sites were used: a seacoast site, a river valley site, and a rolling hills site. Each site data base included five years of meteorological data.~~

~~To address the uncertainty regarding the actual orientation of the AP1000 at a site, the calculation of χ/Q was performed with the plant orientation ranging through the 16 compass points (every 22.5 degrees).~~

~~The ARCON96 code was run for all combinations of source and reception points for each of the 16 plant orientations at each of the three sites. These runs produced the 95th percentile χ/Q values for the five post-accident time periods of interest. Additional conservatism was added to the results for each site by selecting the maximum 95th percentile χ/Q value from the 16 compass points and then selecting the maximum of the three site values for application to the AP1000 reference site. The AP1000 reference site thus, for each time period, selectively combines the most conservative of the calculated values for the three sites and the 16 plant orientations. The χ/Q values are provided in Table 15A-5.~~

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

The locations of the potential release points for a Loss-of-Coolant Accident and their relationship to the main control room air intake and the personnel access door are shown in Figure 15A-1. Figure 15A-2 shows the locations of the potential release points associated with other postulated accidents relative to the possible paths for air entry into the main control room.

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-5 ~~(Sheet 1 of 2)~~

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

Site boundary χ/Q (s/m^3)	
0 – 2 hours[(1)]	TBD ~~6.0x10^{-4}~~
Low population zone χ/Q (s/m^3)	
0 – 8 hours	TBD~~1.35x10^{-4}~~
8 – 24 hours	TBD~~1.0x10^{-4}~~
24 – 96 hours	TBD~~5.4x10^{-5}~~
96 – 720 hours	TBD~~2.2x10^{-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~~5 (Sheet 2 of 2)~~

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

(ALL CONTENT OF TABLE TO BE REVISED)

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~~	~~1.03E-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.~~

~~5. The listed values bound the dispersion factors for releases from the steam vents, the steam line safety & power-operated relief valves, and the condenser air removal stack. 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. Additionally, these dispersion coefficients are conservative for the small line break outside containment.~~

~~6. The listed values bound the dispersion factors for releases from the fuel storage and handling area. These dispersion factors would be used for the fuel handling accident occurring outside containment.~~

~~7. The listed values 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 to be used for evaluating the impact of releases associated with spent fuel pool boiling.~~

Table 15A-7

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

CONTENT TO BE DETERMINED

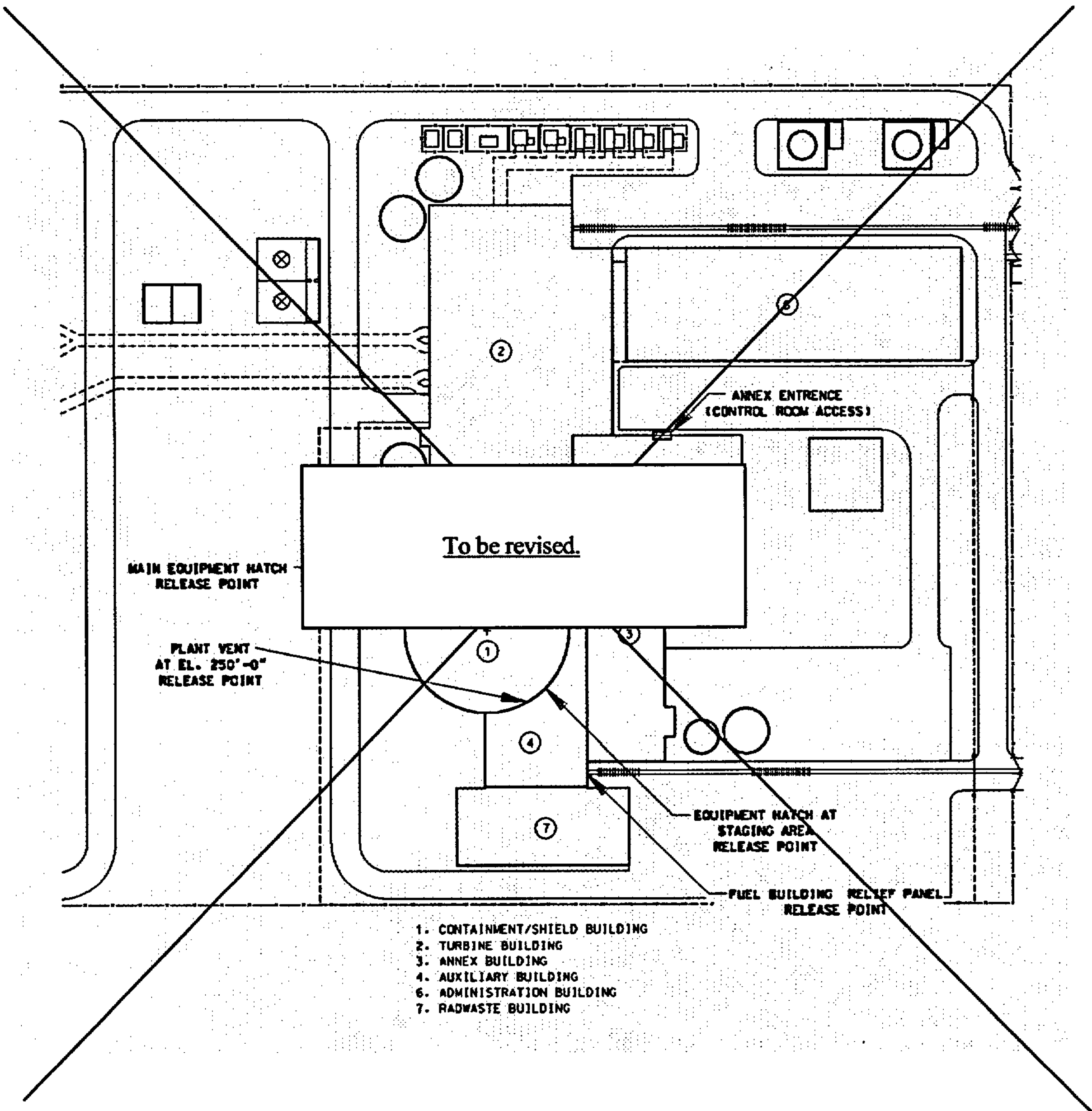


Figure 15A-1

Site Plan With Release and Intake Locations (LOCA Cases)

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.

~~The plant parameters that were used for the calculation of aerosol removal coefficients are those for the AP600 design. It is considered conservative to apply the resulting removal coefficients to the AP1000. This is further addressed in Section 15B.2.6.~~

15B.2.4.1 Containment Geometry

The containment is assumed to be a cylinder with a volume of 55,481 ~~45,900~~ m^3 (1.959 ~~1.62~~ 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 ~~7040~~ m^2 (86,166 ~~75,750~~ ft^2).

~~The above values for areas and containment volume are those for the AP600. The AP1000 would have the same surface area available for sedimentation since there are no changes being implemented for the AP1000 that would significantly impact horizontal surface area inside containment. The area available for diffusive plateout on the AP1000 would be greater than that for the AP600 by 10,400 ft² because of the additional containment height. Consideration of this additional area would increase the rate of heat removal and thus would also increase the removal of aerosols.~~

~~The increase in the AP1000 containment height adversely affects the removal of aerosols by sedimentation. The greater containment height also results in an increase in containment volume of 3.3E5 ft^3. The larger volume results in a reduction in airborne concentration which, when taken alone, results in a reduction in the aerosol removal coefficient.~~

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.

15B.2.4.4 Fission Product Release Fractions

Core inventories of fission products are from ORIGEN calculations for the AP1000 ~~AP600~~ at end of the fuel cycle. Fractional releases to the containment of the fission products are those specified in subsection 15.6.5.3.

~~For the AP1000 there is a greater inventory of fission products in the core for release to the containment atmosphere. The increase is roughly proportional to the increase in core power from that for the AP600. The increase in airborne fission products would increase the aerosol removal coefficients.~~

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 total heat transfer rate, all as functions of time. The AP1000~~AP600~~-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.

~~Using the AP600 thermal-hydraulic data is a conservatism since the AP1000 would have significantly greater decay heat available to power the aerosol removal process. Also, despite the increase in containment volume, the pressure in the AP1000 containment following the postulated LOCA would be greater than for the AP600. With the increased pressure and temperature in the containment there would be an increase in the heat removal rate through the containment shell and an associated increase in the rate of aerosol removal to the containment shell.~~

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 TBD ~~0.43~~ hr^{-1} and TBD ~~0.72~~ 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.2.6 Impact of AP1000 Design Differences on Aerosol Removal Coefficients~~

~~The aerosol removal coefficients reported in Table 15B-1 are from AP600 and are applied to AP1000 analyses.~~

~~As discussed in 15B.2.4, the greater containment height and volume for the AP1000 results in a reduction in the aerosol removal coefficients. However, this increase in volume is offset by a number of other differences identified in 15B.2.4:~~

- ~~Increased heat removal area compared to the AP600 design~~
- ~~Greater fission product activity released to the containment atmosphere due to the higher power level of the AP1000 core (thus increasing airborne concentration)~~
- ~~Increased decay heat to be removed through the containment shell~~

~~Increased post-LOCA temperature and pressure inside containment resulting in higher heat transfer rates~~

~~Thus, the aerosol removal coefficients in Table 15B-1 are conservative for AP1000 analyses.~~

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.

Table 15B-1

AEROSOL REMOVAL COEFFICIENTS IN THE AP1000 ~~AP600~~ CONTAINMENT FOLLOWING A DESIGN BASIS LOCA WITH CORE MELT~~(1)~~

(CONTENT TO BE REVISED)

Time Interval (hours)		Removal Coefficient (hr^{-1})
~~0.167~~	~~0.3~~	~~0.59~~
~~0.3~~	~~0.4~~	~~0.58~~
~~0.4~~	~~0.5~~	~~0.55~~
~~0.5~~	~~0.6~~	~~0.53~~
~~0.6~~	~~0.7~~	~~0.50~~
~~0.7~~	~~0.8~~	~~0.51~~
~~0.8~~	~~0.9~~	~~0.53~~
~~0.9~~	~~1.0~~	~~0.67~~
~~1.0~~	~~1.2~~	~~0.65~~
~~1.2~~	~~1.3~~	~~0.67~~
~~1.3~~	~~1.7~~	~~0.70~~
~~1.7~~	~~1.8~~	~~0.72~~
~~1.8~~	~~1.9~~	~~0.70~~
~~1.9~~	~~2.0~~	~~0.72~~
~~2.0~~	~~2.1~~	~~0.71~~
~~2.1~~	~~2.2~~	~~0.72~~
~~2.2~~	~~2.3~~	~~0.71~~
~~2.3~~	~~2.6~~	~~0.69~~
~~2.6~~	~~2.8~~	~~0.67~~
~~2.8~~	~~3.4~~	~~0.65~~
~~3.4~~	~~3.8~~	~~0.64~~
~~3.8~~	~~4.0~~	~~0.63~~
~~4.0~~	~~4.5~~	~~0.62~~
~~4.5~~	~~5.0~~	~~0.60~~
~~5.0~~	~~5.5~~	~~0.59~~
~~5.5~~	~~6.0~~	~~0.57~~
~~6.0~~	~~6.5~~	~~0.56~~
~~6.5~~	~~7.0~~	~~0.54~~
~~7.0~~	~~8.0~~	~~0.53~~
~~8.0~~	~~9.0~~	~~0.51~~
~~9.0~~	~~10.0~~	~~0.50~~
~~10.0~~	~~12.0~~	~~0.48~~
~~12.0~~	~~14.0~~	~~0.47~~
~~14.0~~	~~20.0~~	~~0.45~~
~~20.0~~	~~24.0~~	~~0.43~~

~~Note:~~

~~1. To provide additional conservatism, the aerosol removal coefficients provided in this table are 0.1 hr^{-1} lower than the values calculated using the models and assumptions of this appendix.~~

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **6.4-1**

Original RAI Number(s): **451.006, 451.006 Rev. 1**

Summary of Issue:

The staff has not completed its review of the applicant's control room atmospheric dispersion factors (see Section 2.3.4 of this report). These factors are an input to the radiological analyses. Pending resolution of the staff's concerns with the hypothetical reference control room χ/Q values, review of the control room habitability radiological consequences analyses for design basis accidents is also incomplete as discussed in DSER Open Item 15.3-2. Therefore, the resolution of issues associated with the analysis of the dose to MCR personnel during design-basis accidents is DSER Open Item 6.4-1.

Westinghouse Response:

This item will be resolved through the resolution of DSER Open Item 2.3.4-1.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 15.3.6-1

Original RAI Number(s): 470.009, 470.011

Summary of Issue:

The staff has not completed its evaluation of the applicant's assumptions on aerosol removal in containment, as discussed in RAIs 470.009 and 470.011. To verify the applicant's assessment, the staff will perform independent radiological consequence calculations for a postulated design-basis LOCA coincident with the loss of spent fuel pool cooling capability once these issues are resolved. This is Open Item 15.3.6-1.

Westinghouse Response:

Sufficient information has already been provided for the NRC to proceed with its evaluation, specifically in our responses to RAI 470.009 (transmitted by Westinghouse letter DCP/NRC1535, November 26, 2002) and RAI 470.011 Rev. 1 (transmitted by Westinghouse letter DCP/NRC1571, April 11, 2003).

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.2.10-4 Revision 1

Original RAI Number(s): 261.018

Summary of Issue:

RG 1.68, Appendix A, Section 5.m.m recommends that the power ascension test program include demonstrations that the dynamic response of the plant is in accordance with design for the case of automatic closure of all main steam line isolation valves (MSIVs). In reviewing the power ascension test program test abstracts, the NRC staff noted that no MSIV closure testing is performed during power ascension testing. In RAI 261.007b, the staff requested that the applicant provide additional information regarding performance of MSIV closure testing. In their November 13, 2002, RAI response, the applicant stated that the dynamic response of the plant to closure of all MSIVs is bounded by a plant trip from 100 percent power, which is performed in Test Abstract 14.2.10.4.24. The NRC staff lacks sufficient information to conclude that the plant trip from 100 percent bounds the MSIV closure transient. In RAI 261.018, the NRC staff requested the applicant to provide additional information regarding the basis for the statement that the MSIV closure transient is bounded by a plant trip from 100 percent power. This is Open Item 14.2.10-4.

Westinghouse Response:

This question was originally identified as RAI 261.018 Rev. 0. Westinghouse provided a response to RAI 261.018 Rev. 0 and transmitted it to the NRC via DCP/NRC1592 dated 05/21/03.

NRC Additional Comments:

As the turbine by-pass valves will open during the test proposed Westinghouse, it is not clear that the proposed test "bounds" the MSIV closure transient. Westinghouse is requested to provide additional information.

Westinghouse Additional Response:

Rather than say the plant trip from 100% power, which is performed in test abstract 14.2.10.4.24, "bounds" the MSIV test identified in RG 1.68 m.m, it would be more correct to say that the proposed test allows sufficient information to be obtained to demonstrate that the dynamic response of the plant is in accordance with the design. The pressure transient in the plant resulting from opening the main generator breaker during the proposed test can be compared to analyses and is sufficient to confirm that the plant responds as predicted.

As previously noted, Westinghouse has traditionally not performed RG 1.68 test m.m on its plants as closure of the MSIVs at full or reduced power would lead to a severe transient, which could lead to the opening of the plant safety valves.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-12 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Section 3.1, "Emergency Response Facilities," the staff finds this ITAAC unacceptable because it does not address the radiological habitability or the ventilation system for the technical support center; both of which should be the same as, or comparable to the main control room ITAAC. This is Open Item 14.3.2-12.

Westinghouse Response:

Westinghouse will revise the DCD as shown below.

NRC Additional Comments:

The proposed additional Tier 1 ITAAC is: "The TSC provides a suitable workspace environment." First, what does "suitable" mean? It is not clear that it means "habitable" and compliance with the GDC 19 criteria for all design basis accidents. Second, it ties the Inspections, Tests, Analyses, and Acceptance Criteria to Table 2.7.1-4, which includes reference to the MCR. Third, the MCR (and now, proposed TSC) criteria in Table 2.7.1-4 are ambiguous, as it applies to meeting GDC 19 criteria for the TSC, and limit testing to controls in the MCR only.

The logic seems to imply that if the control room-operated testing is successful, then the equipment works, then the MCR (and TSC) are "habitable," and then they meet GDC 19, and this applies to all design basis accidents. Such a connection is not clearly laid out.

The specific testing does not indicate whether (or not) there are any non-MCR controlled components that need to be tested, in order to confirm the TSC habitability design commitment. The ITAAC could be something as simple as: "The TSC meets GDC 19 criteria for all design basis accidents;" or possibly something like "The TSC provides a habitable workspace environment," with clear Inspections, Tests, Analyses, and Acceptance Criteria on what "habitable" means – rather than reference to another section (unless, of course, that reference and criteria are clear). The logic appears to be too remote from what should be a clear TSC design commitment statement (including the Inspections, Tests, and Analyses statement), and having an objective Acceptance Criteria. Then, a reference to Inspections, Tests, Analyses, and Acceptance Criteria elsewhere in the Tier 1 DCD is appropriate. This should be clearly indicated in both the Tier 2 DCD and Tier 1 ITAAC, as appropriate.

Westinghouse Additional Response:

The word "suitable" in the original proposed revision to DCD Tier 1 will be changed to "habitable" as shown below.

Also, a clarification of the intent of references to other ITAACs in the ITAAC tables will be add to the DCD as shown below.

In addition to the ITAAC testing, preoperational testing of the VBS is described in DCD Tier 2 subsection 14.2.9.2.10. The changes shown below will be made to DCD Tier 2 subsection 14.2.9.2.10 to clarify the importance of the TSC-related functions.

Design Control Document (DCD) Revision:

DCD Tier 1 section 3.1

- Add new item 6, under Design Description as follows:

  "6. The TSC provides a ~~suitable~~ **habitable** workspace environment."

- Revise Table 3.1-1 to include new item 6 as follows:

6. The TSC provides a ~~suitable~~ **habitable** workspace environment.	See Tier 1 Material, ~~subsection 2.7.1~~ **Table 2.7.1-4, items 1, 8a), 8c), 12 and 13,** Nuclear Island Nonradioactive Ventilation System	See Tier 1 Material, ~~subsection 2.7.1~~ **Table 2.7.1-4, items 1, 8a), 8c), 12 and 13,** Nuclear Island Nonradioactive Ventilation System

DCD Tier 1 section 2.7.1

- Revise item 8.c) under Design Description as follows:

  The VBS maintains MCR **and** TSC habitability when radioactivity is detected.

- Revise item 8.c) in Table 2.7.1-4 under Design Commitment as follows:

8.c) The VBS maintains MCR and TSC habitability when radioactivity is detected.	See item 12 in this table.	See item 12 in this table.

DCD Tier 1 section 1.2

- Revise Tier 1 Section 1.2 as follows:

Implementation of ITAAC

The ITAACs are provided in tables with the following three-column format:

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria

Each design commitment in the left-hand column of the ITAAC tables has an associated ITA requirement specified in the middle column of the tables.

The identification of a separate ITA entry for each design commitment shall not be construed to require that separate inspections, tests, or analyses must be performed for each design commitment. Instead, the activities associated with more than one ITA entry may be combined, and a single inspection, test, or analysis may be sufficient to implement more than one ITA entry.

An ITA may be performed by the licensee of the plant or by its authorized vendors, contractors, or consultants. Furthermore, an ITA may be performed by more than a single individual or group, may be implemented through discrete activities separated by time, and may be performed at any time prior to fuel load (including before issuance of the combined license for those ITAACs that do not necessarily pertain to as-installed equipment). Additionally, an ITA may be performed as part of the activities that are required to be performed under 10 CFR Part 50 (including, for example, the quality assurance (QA) program required under Appendix B to Part 50); therefore, an ITA need not be performed as a separate or discrete activity.

Many of the acceptance criteria include the words "A report exists and concludes that..." When these words are used it indicates that the ITAAC for that design commitment will be met when it is confirmed that appropriate documentation exists. Appropriate documentation can be a single document or a collection of documents that meet the stated acceptance criteria. Examples of appropriate documentation include design reports, test reports, inspection reports, analysis reports, evaluation reports, design and manufacturing procedures, certified data sheets, commercial dedication procedures and records, quality assurance records, calculation notes, and equipment qualification data packages.

Many ITAAC are only a reference to another Tier 1 location, either a section, subsection, or ITAAC table entry (e.g., "See Tier 1 Material..."). This reference is an indication that the acceptance criteria for that design commitment are satisfied when the acceptance criteria for the referenced Tier 1 sections, subsections, or table entries are satisfied. If a complete Tier 1 section is referenced, this indicates that all the acceptance criteria in that section must be met before the referencing design commitment is satisfied.

DCD Tier 2 subsection 14.2.9.2.10

14.2.9.2.10 Nuclear Island Nonradioactive Ventilation System Testing

Purpose

The purpose of the nuclear island nonradioactive ventilation system testing is to verify that the as-installed system properly performs the following defense-in-depth functions, as described in subsection 9.4.1:

- Protect the main control room **and technical support center** from smoke infiltration
- Provide the capability to remove smoke from the main control room, **technical support center**, and Class 1E electrical equipment rooms
- Provide heating, ventilation, and cooling for the main control room, **technical support center**, and Class 1E electrical equipment rooms
- Provide air filtration to limit radioactivity in the main control room **and technical support center**
- Maintain passive heat sinks at acceptably low initial temperatures
- Maintain the main control room **and technical support center** at positive pressure

The safety-related functions associated with this system are tested as part of the main control room emergency habitability testing described in subsection 14.2.9.1.6.

Prerequisites

The construction testing of the nuclear island nonradioactive ventilation system has been completed. The required preoperational testing of central chilled water system, the hot water heating system, the ac electrical power and distribution systems, and other interfacing systems required for operation of the above systems has been completed. Data collection is available as needed to support the specified testing and system configurations.

General Test Acceptance Criteria and Methods

Nuclear island nonradioactive ventilation system performance is observed and recorded during a series of individual component and integrated system testing to verify the system performs its defense-in-depth functions. The following testing demonstrates that the system performs its defense-in-depth functions as described in subsection 9.4.1 and appropriate design specifications:

a) Proper function of the fans, filters, heaters, coolers, and dampers is verified.

b) Proper operation of instrumentation, controls, actuation signals, and alarms and interlocks is verified. This testing includes the following:

- Smoke detectors and alarms
- Air handling unit and fan flows, controls, and alarms
- Differential air pressures and alarms
- Air and air filtration unit charcoal temperatures, controls, and alarms
- Air relative humidity measurements, controls, and alarms
- Isolation/shutoff damper controls
- Fire/smoke damper controls

This testing includes operation from the main control room.

c) The proper air flows from and through each air handling unit, as well as to and from the main control room, **technical support center**, and other equipment rooms is established for each mode of operation.

d) The main control room ~~is~~ **and technical support center are**verified to be maintained at the proper positive pressure.

e) The main control room, **technical support center**, class 1E equipment rooms, and passive heat sink areas are verified to be maintained at their proper temperature during hot functional testing.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19.1.10.3-1 (Response Revision 1)

Original RAI Number(s): None

Summary of Issue:

Representative Sequences for Assigning Source Terms

The accident sequences used to represent the various release categories are identified and briefly described in PRA Chapter 45. Additional sequence information is provided in PRA Chapter 34. The basis for selecting the representative sequence for each release category is not provided. Such information is necessary in order to confirm that the sequence selected to represent each release category is reasonably representative of the collection of sequences assigned that category, in terms of the magnitude, timing, energy, and elevation of release. Based on the limited information that was provided, the staff noted a number of inconsistencies. Specifically, for release category CFE releases from the ADS Stage 4 valves enter directly into containment rather than into the IRWST, and given the location of the valves relative to the containment shell, would not result in containment failure from diffusion flames as assumed in the PRA. For release category CFL containment failure is assumed at 3 hours, which is inconsistent with the time frame for late containment failure. Also, important details impacting the release characteristics need to be documented, such as whether an additional decontamination factor is credited in determining the source term for SGTR events (as it was in AP600), and the containment isolation failure location and size assumed for containment isolation sequences. This is Open Item 19.1.10.3-I.

Westinghouse Response (Revision 1):

~~Please see Attachment 45A of the AP1000 PRA revision 3 for resolution of the inconsistencies and justification of the source terms selected for the CFE, CFI and CFL release categories~~.

~~The accident sequence representing the CFE release category is 3D-4. The sequence description in Chapter 34 incorrectly states that the sequence is initiated by the spurious opening of 2 stage-4 ADS valves. Actually, the sequence is correctly initiated by the spurious opening of 2 stage-2 ADS valves. The sequence description in Chapter 34 has been corrected as outlined below.~~

~~No additional decontamination is credited for the SGTR source terms in the BP release category in the AP1000 PRA, which is conservative with respect to the approach in the AP600 PRA. There is no reference to a decontamination factor applied to the BP release category in the AP1000 PRA documentation or the design control document chapter 19~~.

In revision 2 of the AP1000 PRA report, the source term from MAAP4 accident sequence 3BE-3 represents the CFI, intermediate containment failure, and the source term from accident sequence 3BE-7 represents the late containment failure case. Both of these cases are actually early containment failure cases and are incorrectly chosen to represent the intermediate and late release category source terms. MAAP4 sequences that appropriately represent the CFI and CFL release categories have been added to the PRA report in Chapter 34, and their source terms are outlined in Chapter 45.

The case that represents release category CFE is 3D-4. Section 34.4.8.1 states that the accident sequence is initiated by 2 Stage 4 ADS valves spuriously opening. This statement is not correct. The accident sequence is initiated by 2 Stage 2 ADS valves spuriously opening. Case 3D-4 correctly models the conditions that could potentially produce a diffusion flame at the IRWST vents. Section 34.4.8.1 will be revised to include this correction as shown below.

The fission product releases for each of the containment failure release categories (BP, CI, CFE, CFI, and CFL) were chosen conservatively to result in the earliest possible releases with the greatest magnitudes to maximize the 24 hour and 72 hour site boundary doses. Each of these release categories is discussed below.

Release Category BP

The release category BP, containment bypass, source terms are taken from Case 6E-1, a steam generator tube rupture case with a stuck open secondary system relief valve and no ADS. The release category BP represents all direct releases to the environment, bypassing the containment. The dominant sequences are all steam generator tube rupture sequences, either as the initiating event or induced by high pressure core melt. The contribution of interfacing systems LOCA frequency to the BP release category is negligible.

A comparison of releases from the release category BP accident sequences presented in Chapter 39 is presented in Table 1. The releases from the 6E-1 case are bounding and produce the most limiting doses.

Table 1 – Comparison of BP Releases at 24 Hours

Sequence	Core Uncovery (s)	Core Relocation (s)	Released Fraction at 24 hours			Induced RCS Ruptures
			FREL(1) Noble Gases	FREL(2) CsI (volatiles)	FREL(4) SrO (non-volatiles)	
1A-1	11144	19604	5.3E-1	5.5E-3	2.3E-4	HL creep: none SGT creep: 14000 s
1A-2	4980	11495	6.3E-1	1.2E-2	3.0E-4	HL creep: 22175 s SGT creep: 7000 s
1AP-1	121125	144724	6.0E-2	4.0E-4	2.6E-5	HL creep: none SGT creep 133253 s
1AP-2	143614	157909	6.5E-2	2.9E-3	1.6E-4	HP creep: none SGT creep: 139113 s
6E-1	30764	36844	1.0E-0	3.0E-1	3.5E-3	/
6E-2	42483	48447	5.0E-1	2.3E-1	1.6E-2	/

Release Category CFE

Release category CFE, early containment failure, is represented by the source term from accident sequence 3D-4. This sequence is initiated by a spurious opening of the ADS-2 valves, and fails the containment early due to a diffusion flame at the IRWST vents near the containment shell. Accident sequences 3BE-3 and 3BE-7 also represent early containment failure sequences, 3BE-3 by early hydrogen detonation, and 3BE-7 by ex-vessel steam explosion.

Offsite dose were calculated for all three of these accident sequences, 3D-4, 3BE-3 and 3BE-7. The results are summarized in Table 2. The offsite doses for accident sequence 3D-4 bound the offsite dose for the other cases.

Table 2
24 Hour Site Boundary Whole Body Dose (effective dose equivalent)

Case	Dose (sieverts)
3D-4	42.3
3BE-3	32.5
3BE-7	25.8

Release Category CI

Release category CI, containment isolation failure, is represented by the source term from accident sequence 3C-2, a vessel rupture that fails containment on accident initiation. This sequence was chosen because of the very early timing of the release, and the extensive damage to the core in the vessel rupture sequence. Therefore, 3C-2 is an appropriate representation for the CI release category.

Release Category CFI

Release category CFI, intermediate containment failure is represented by the source term from accident sequence CFI. Containment failure is induced by a hydrogen detonation event, which occurs when the global containment hydrogen concentration exceeds 10 volume percent. Release category CFI has a very small frequency and does not contribute significantly to the plant risk. Variations in the source term will not impact the plant risk. Therefore, sequence CFI is an appropriate representation for the CFI release category.

Release Category CFL

Release category CFL, late containment failure is represented by the source term from accident sequence CFL. Passive containment cooling water is failed, and containment failure is induced by containment overpressurization, which occurs over the long-term of the accident sequence. Release category CFL has a very small frequency and does not contribute significantly to the plant risk. Variations in the source term will not impact the plant risk. Therefore, sequence CFL is an appropriate representation for the CFL release category.

Design Control Document (DCD) Revision:

None

PRA Revision:

Chapter 34 Revision

34.4.8.1 3D-4

The sequence description and assumptions are listed below.

- ~~Two valves of ADS Stage 4 spuriously open~~ **Two valves of ADS Stage 2 spuriously open**
- Failure of PRHR
- 0/2 ADS stage 1
- 0/2 ADS stage 2
- 0/2 ADS stage 3
- 0/4 ADS stage 4
- 0/2 CMTs
- 2/2 accumulators
- 0/2 IRWST gravity injection lines
- 0/2 IRWST recirculation lines
- 2/2 cavity flooding lines
- Hydrogen igniters operating
- Upper compartment failure due to hydrogen release from IRWST

The main events of the case are shown in Table 34-20, while relevant plots are presented in Figures 34-273 through 34-289.

Revisions to AP1000 PRA Report Revision 3 Chapters 34, 45, 49, and 59 are attached. All of these revisions are part of the Westinghouse Revision 1 response to this DSER Open Item.

In addition, as part of the Westinghouse Revision 1 response to this DSER Open Item, Attachment 45A (PRA Report Revision 3) will be deleted in its entirety.

34.4.13 *Intermediate and Late Containment Failure Cases*

34.4.13.1 CFI – Intermediate Containment Failure Case

The sequence description and assumptions are listed below:

- Accident class 3BE DVI line break in the PXS compartment (PXS is flooded through broken DVI line)
- Hydrogen burn and containment failure from deflagration-to-detonation transition (DDT) when containment global hydrogen concentration exceeds 10%
- 2/2 ADS stage 1 – automatic
- 2/2 ADS stage 2 – automatic
- 2/2 ADS stage 3 – automatic
- 4/4 ADS stage 4 – automatic
- 1/2 CMTs
- 1/2 accumulators
- 0/2 IRWST gravity injection lines
- 1/2 cavity flooding through recirculation lines
- no hydrogen igniters operating

The accident sequence timing is presented in Table 34-27. Relevant plots are presented in Figures 34-392 through 34-406.

34.4.13.2 CFL – Late Containment Failure Case

The sequence description and assumptions are listed below:

- Accident class 3BE Medium LOCA in a hot leg to the loop compartment
- failure of passive containment cooling system cooling water
- containment failure from long-term containment overpressure at 91 psig (ASME service level C)
- 2/2 ADS stage 1 – automatic
- 2/2 ADS stage 2 – automatic
- 2/2 ADS stage 3 – automatic
- 4/4 ADS stage 4 – automatic
- 2/2 CMTs
- 2/2 accumulators
- 0/2 IRWST gravity injection lines
- 1/2 cavity flooding through recirculation lines
- no hydrogen igniters credited, containment is steam inerted

The accident sequence timing is presented in Table 34-28. Relevant plots are presented in Figures 34-407 through 34-425.

Table 34-27

CFI Event Summary

Time (Seconds)	Description
0.0	DVI Line Break to PXS Compartment
20.5	Reactor Scram
25.3	Main Coolant Pump Trip
25.3	CMT Actuation
57.2	PCS Actuation
617.4	ADS Stage 1 Actuation – Automatic
737.4	ADS Stage 2 Actuation – Automatic
857.4	ADS Stage 3 Actuation – Automatic
904.8	Accumulator Water Depleted
1298.0	Containment Water Level @ 83'
1587.8	ADS Stage 4 Actuation – Automatic
2480.0	Core Uncovery
3370.9	Cavity Flooding Actuation
3422.0	Onset of Core Melting (TCRHOT > 2500°K)
6250.0	Containment Water Level @ 98'
N/A	Core Relocation to Lower Plenum
N/A	Lower Plenum Dryout
7080.0	Hot Leg Submerged
N/A	PRHR Actuation
N/A	IRWST Injection Initiated
N/A	IRWST Low Level – Switchover to Recirculation
N/A	Vessel Failure
25000.	Global Hydrogen Burn and DDT
25002.	Containment Failure

Table 34-28

CFL Event Summary

Time (Seconds)	Description
0.0	MLOCA Hot Leg Break to Loop Compartment
35.0	Reactor Scram
42.7	Main Coolant Pump Trip
42.7	CMT Actuation
N/A	PCS Actuation
690.0	Containment Water Level @ 83'
757.4	ADS Stage 1 Actuation – Automatic
877.4	ADS Stage 2 Actuation – Automatic
977.4	ADS Stage 3 Actuation – Automatic
1070.8	Accumulator Water Depleted
1729.5	ADS Stage 4 Actuation – Automatic
2461.1	Core Uncovery
3315.5	Cavity Flooding Actuation
3402.0	Onset of Core Melting (TCRHOT > 2500°K)
4990.0	Containment Water Level @ 98'
N/A	Core Relocation to Lower Plenum
N/A	Lower Plenum Dryout
5621.0	Hot Leg Submerged
N/A	PRHR Actuation
N/A	IRWST Injection Initiated
N/A	IRWST Low Level – Switchover to Recirculation
N/A	Vessel Failure
108573.	Containment Failure

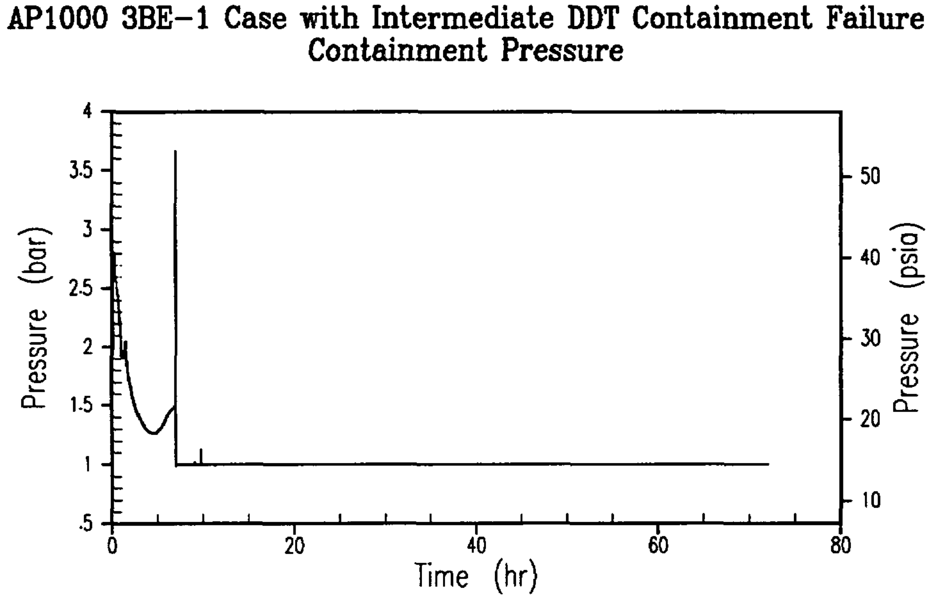


Figure 34-392

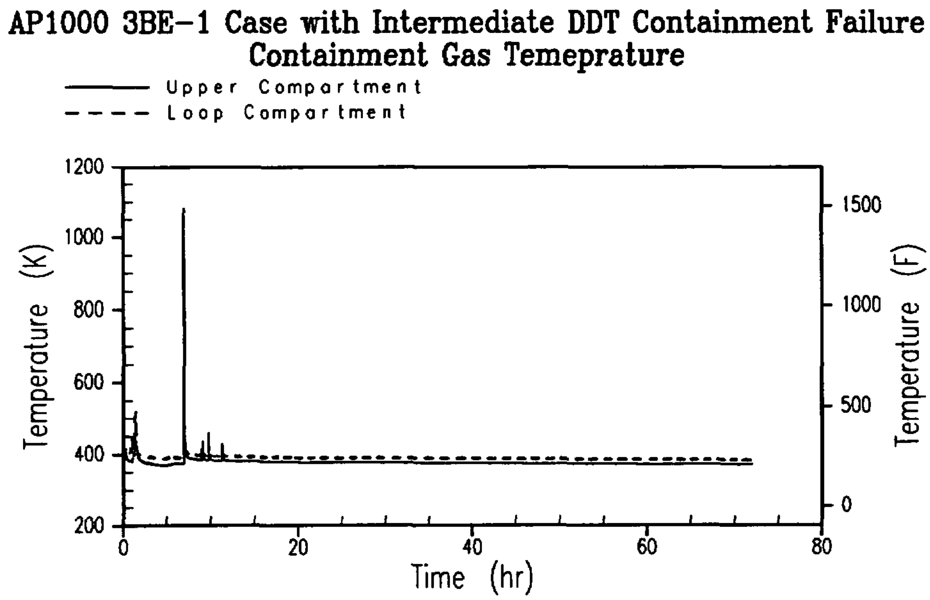


Figure 34-393

AP1000 3BE-1 Case with Intermediate DDT Containment Failure
Containment Hydrogen Concentration

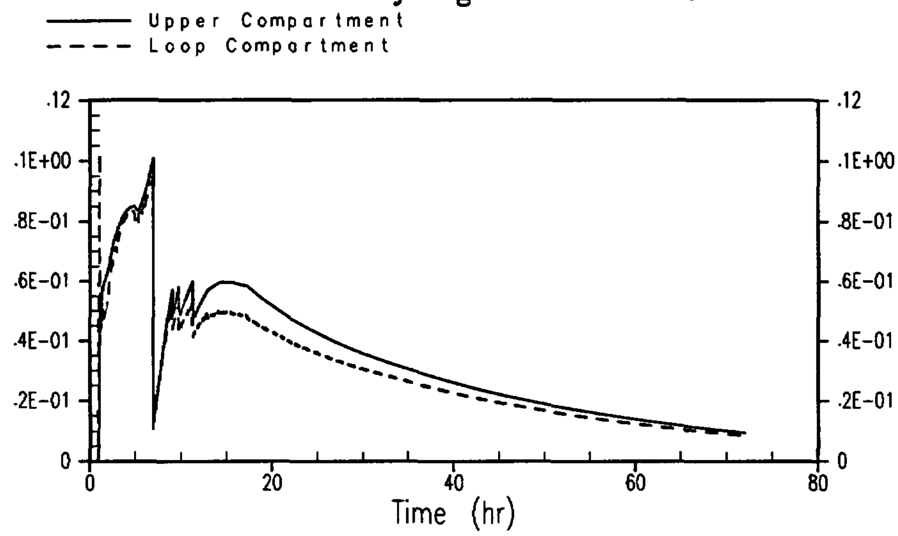


Figure 34-394

AP1000 3BE-1 Case with DDT Intermediate Containment Failure
Noble Gases Release Fraction

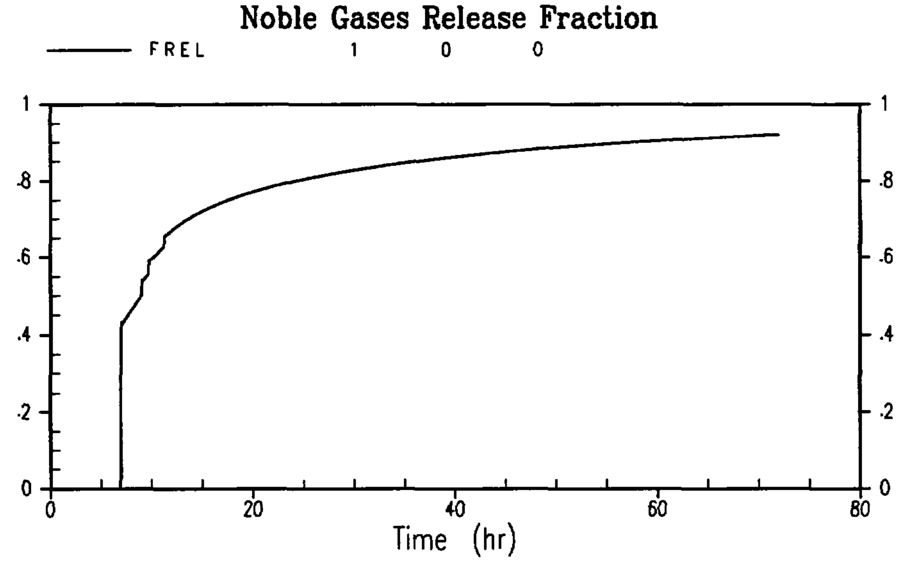


Figure 34-395

AP1000 3BE-1 Case with DDT Intermediate Containment Failure
CsI and RbI Release Fraction

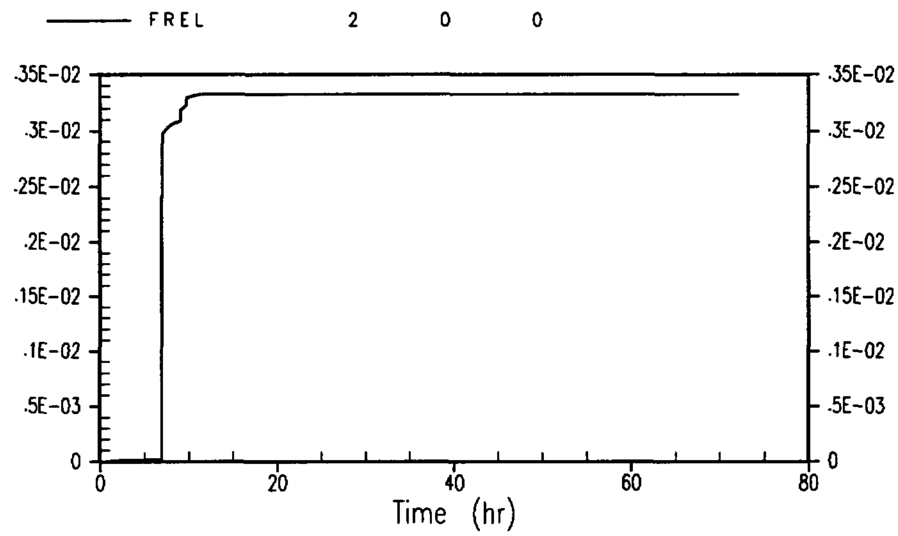


Figure 34-396

AP1000 3BE-1 Case with DDT Intermediate Containment Failure
Te02 Release Fraction

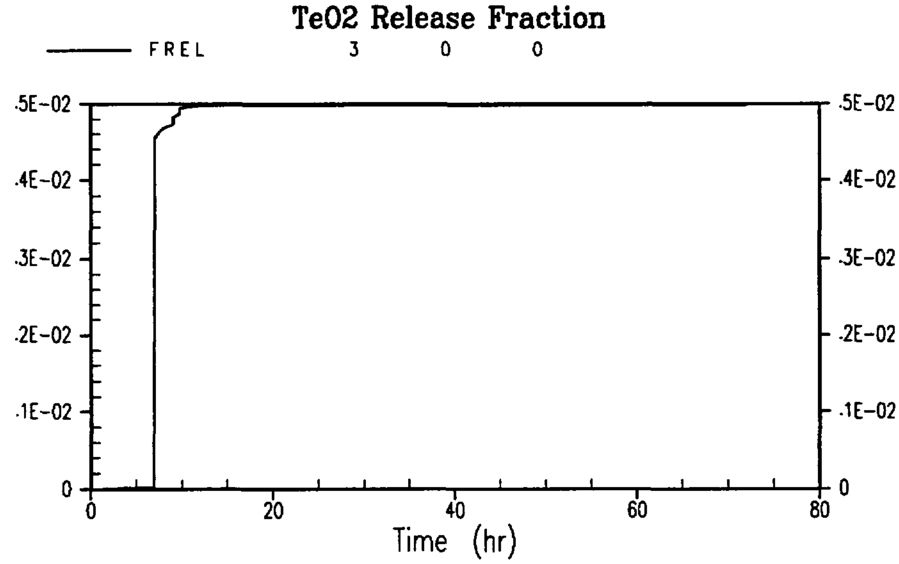


Figure 34-397

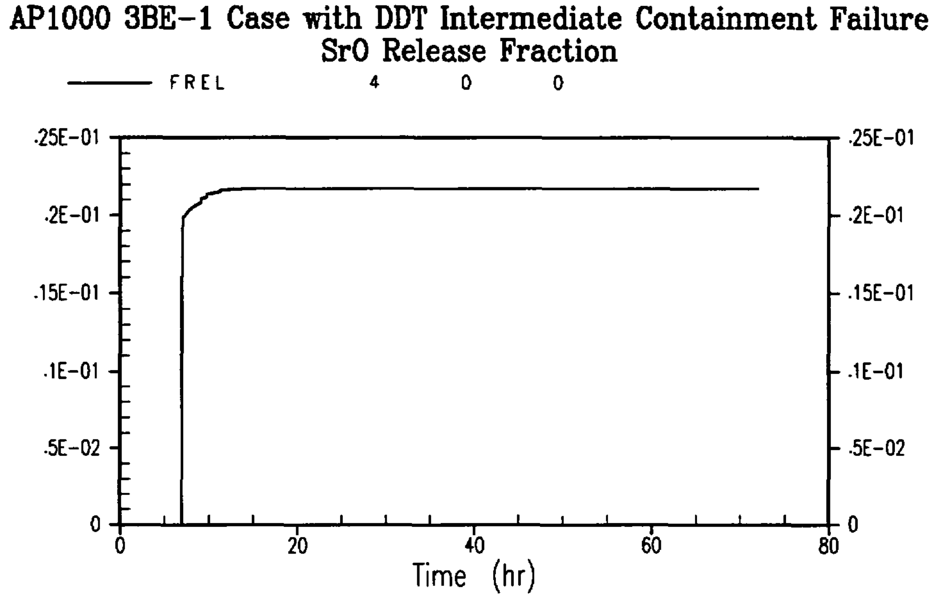


Figure 34-398

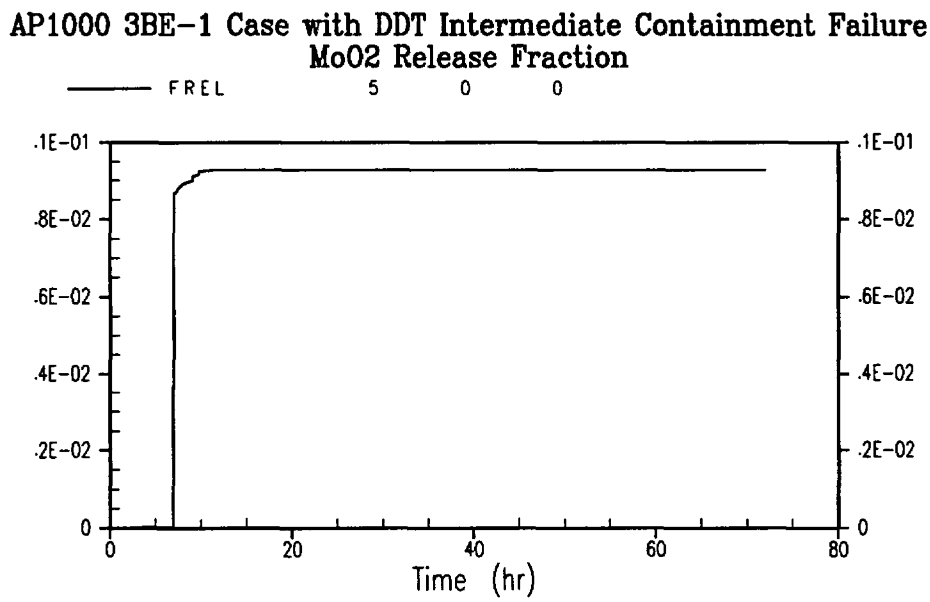


Figure 34-399

AP1000 3BE-1 Case with DDT Intermediate Containment Failure
CsOH and RbOH Release Fraction

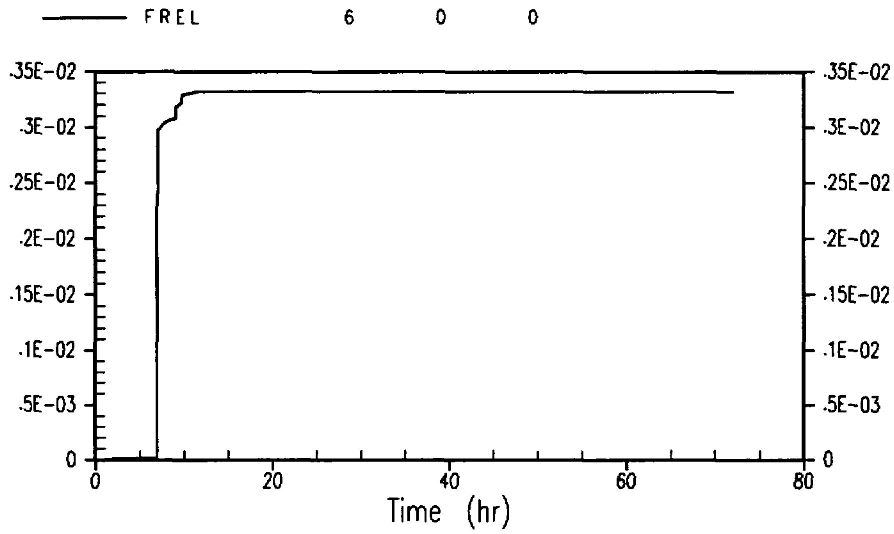


Figure 34-400

AP1000 3BE-1 Case with DDT Intermediate Containment Failure
BaO Release Fraction

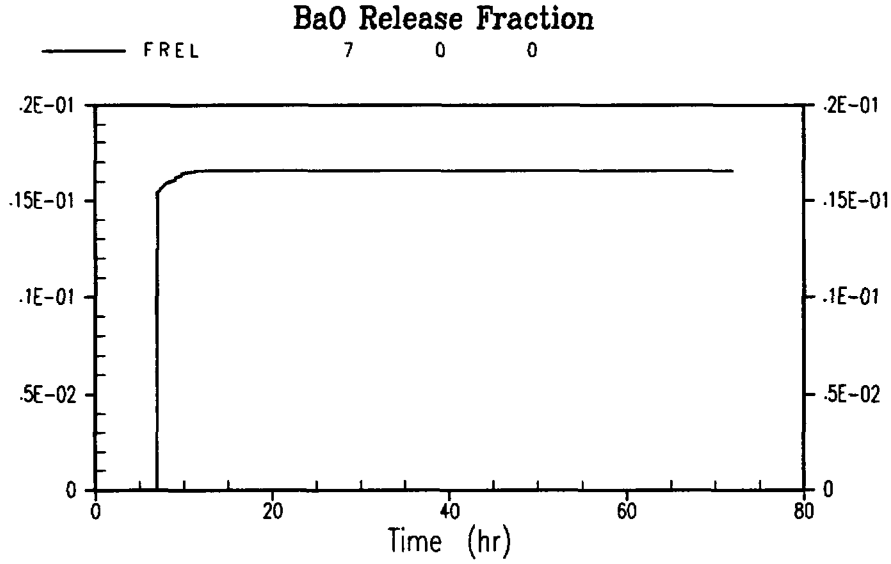


Figure 34-401

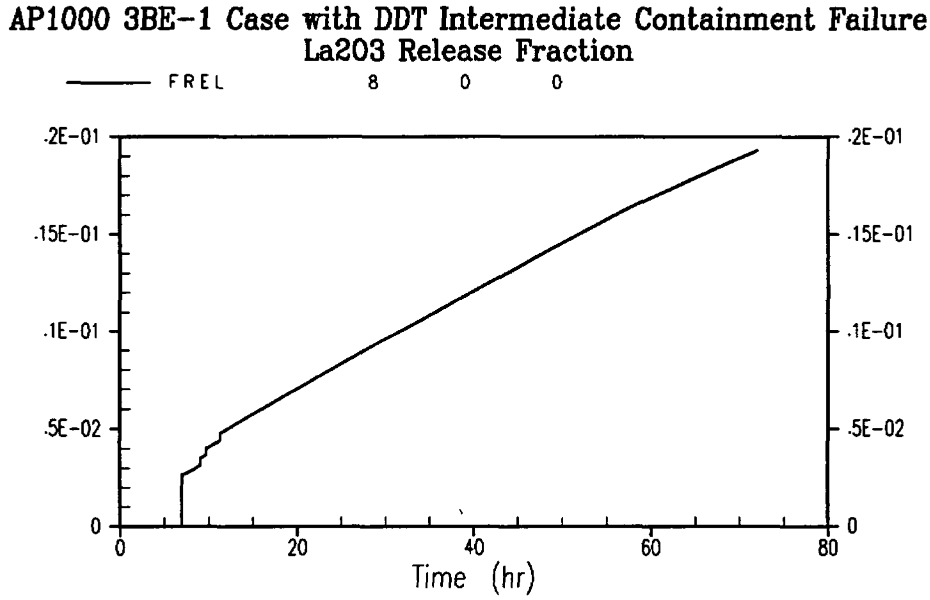


Figure 34-402

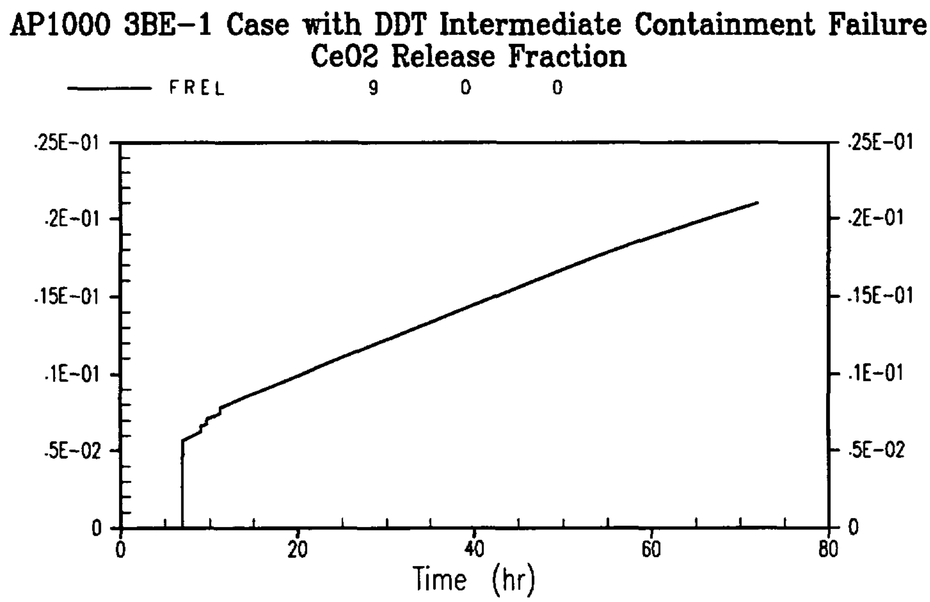


Figure 34-403

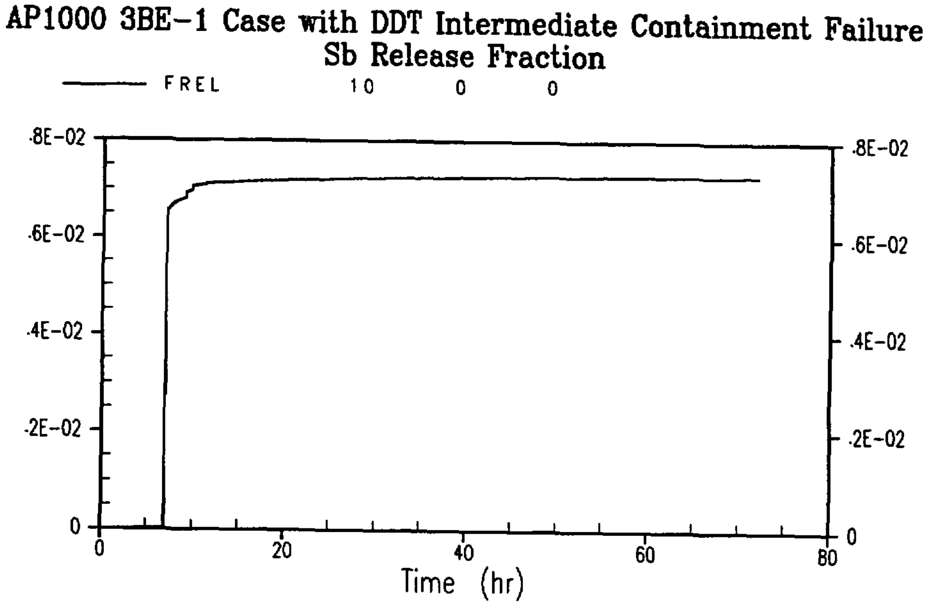


Figure 34-404

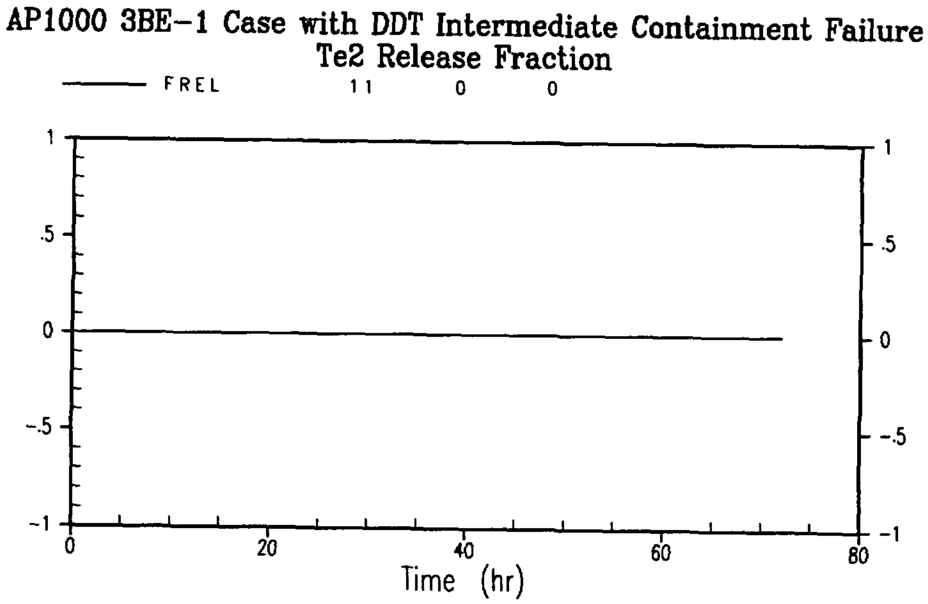


Figure 34-405

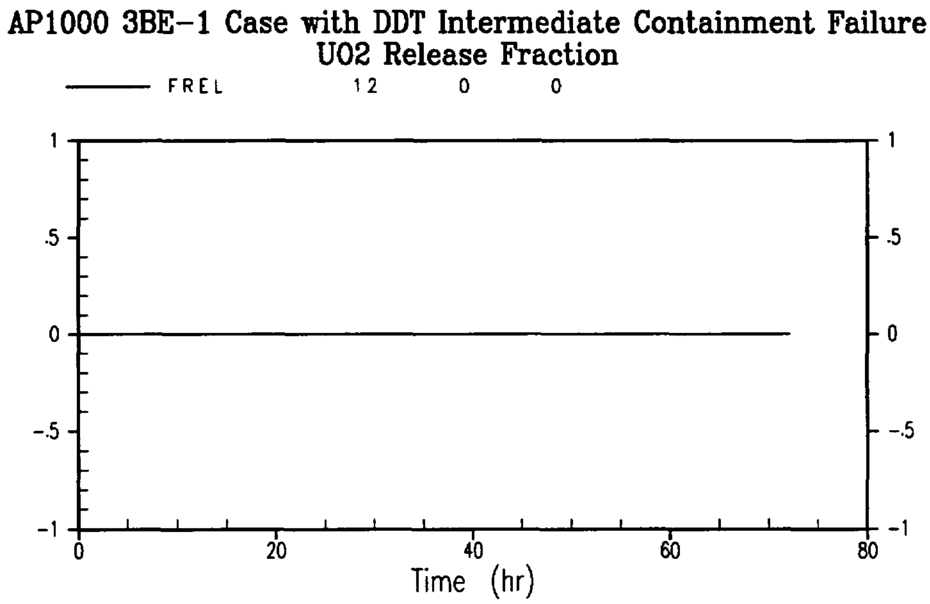


Figure 34-406

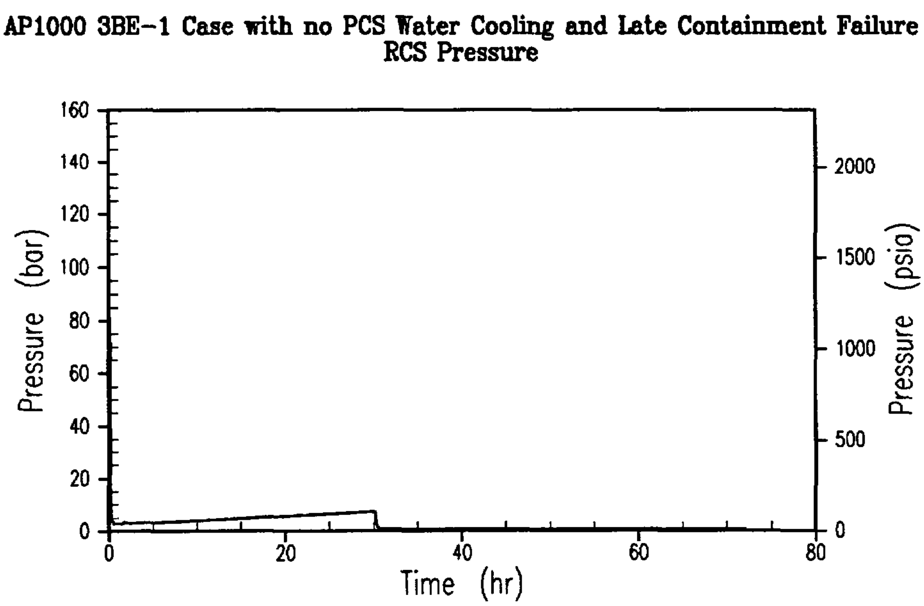


Figure 34-407

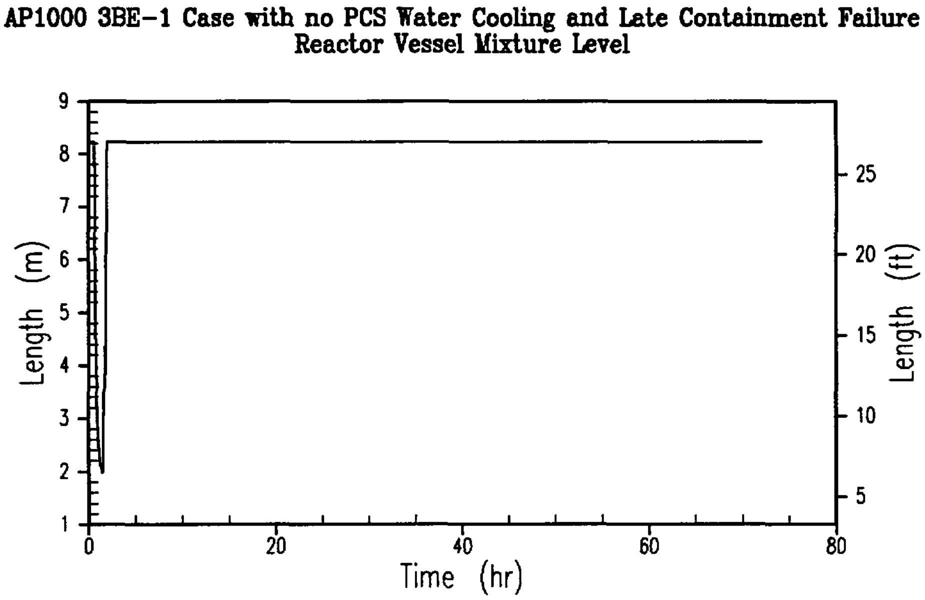


Figure 34-408

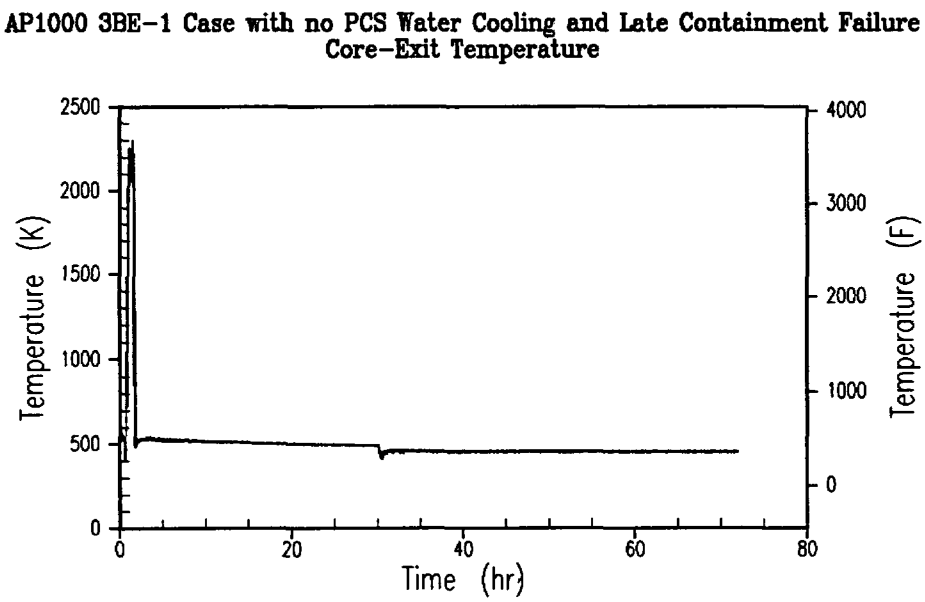


Figure 34-409

AP1000 3BE-1 Case with no PCS Water Cooling and Late Containment Failure
In-Vessel Hydrogen Generation

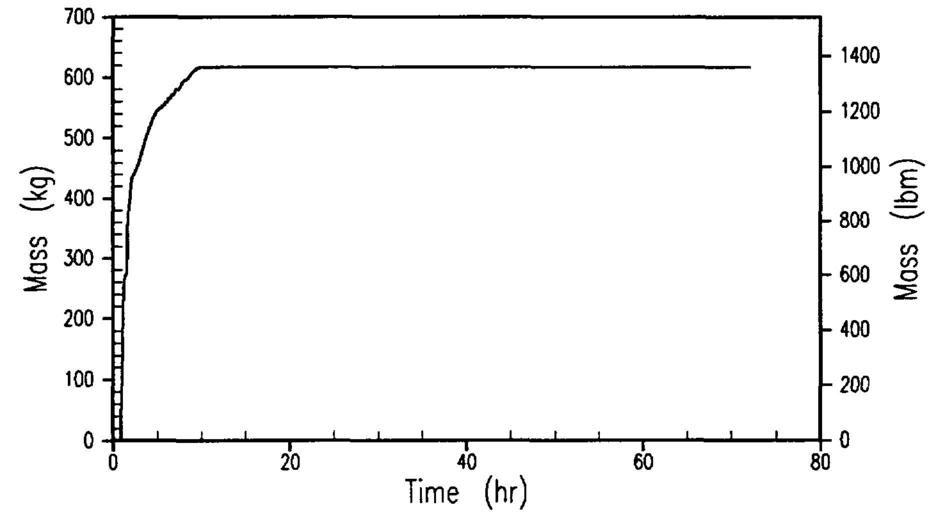


Figure 34-410

AP1000 3BE-1 Case with no PCS Water Cooling and Late Containment Failure
Containment Pressure

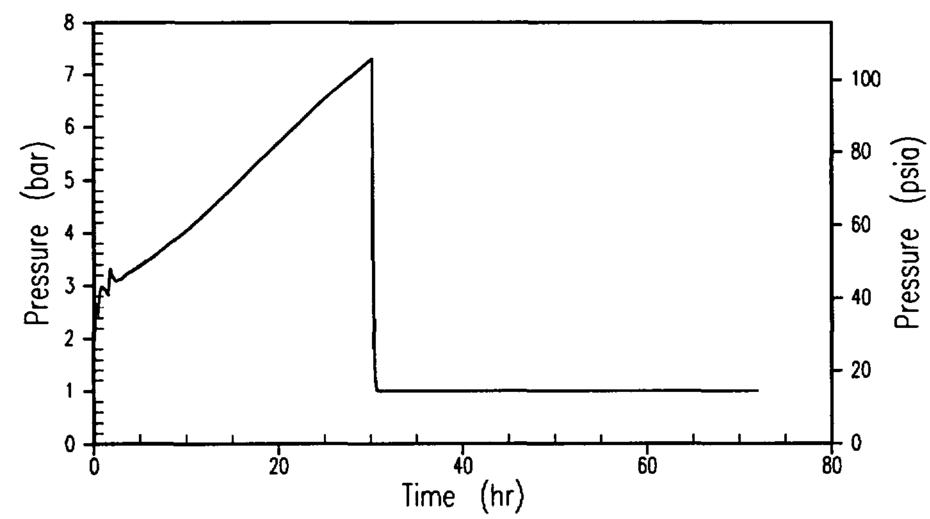


Figure 34-411

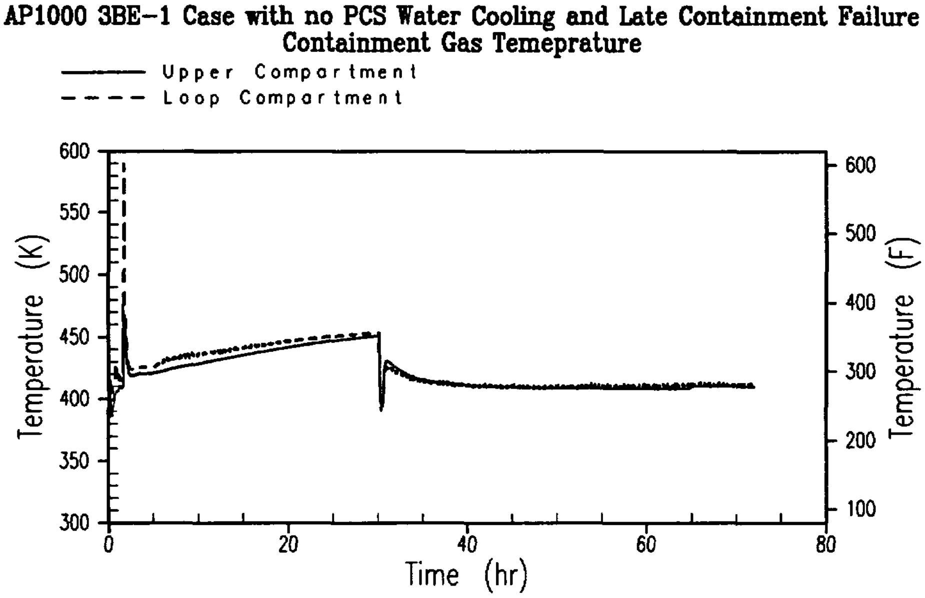


Figure 34-412

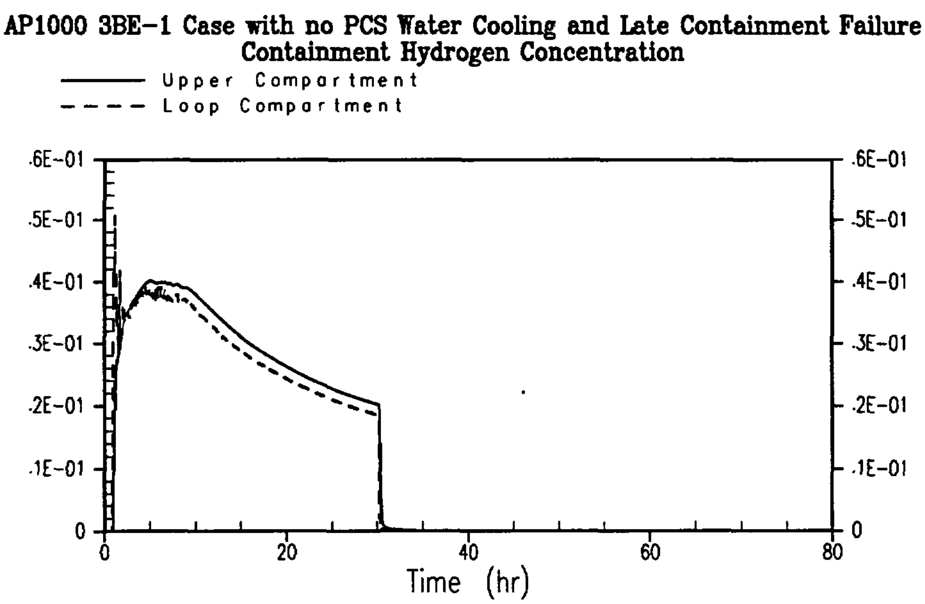


Figure 34-413

AP1000 3BE-1 Case with no PCS Water Cooling and Late Containment Failure
Noble Gases Release Fraction

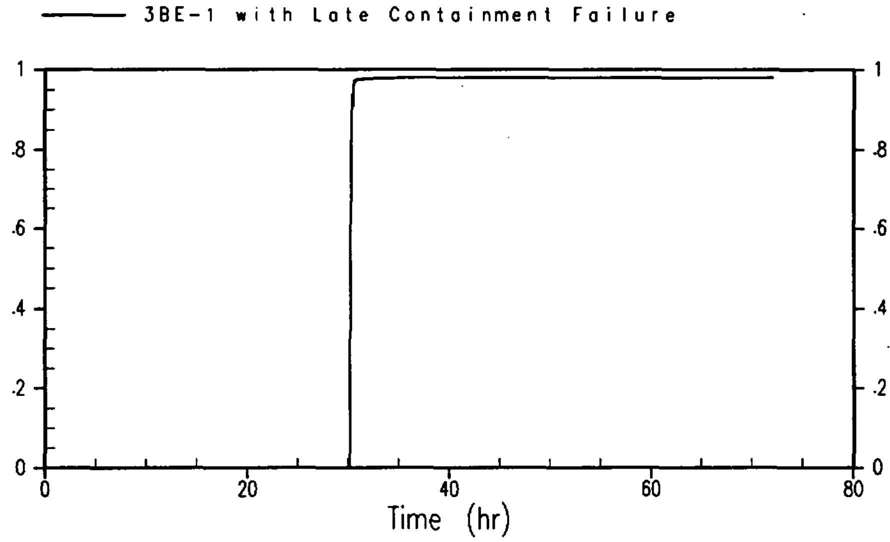


Figure 34-414

AP1000 3BE-1 Case with no PCS Water Cooling and Late Containment Failure
CsI and RbI Release Fraction

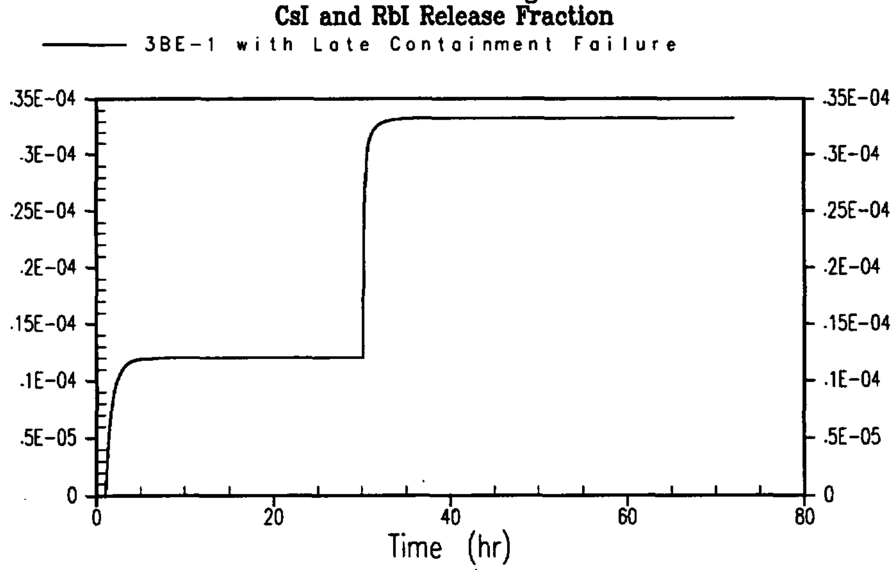


Figure 34-415

AP1000 3BE-1 Case with no PCS Water Cooling and Late Containment Failure
TeO2 Release Fraction

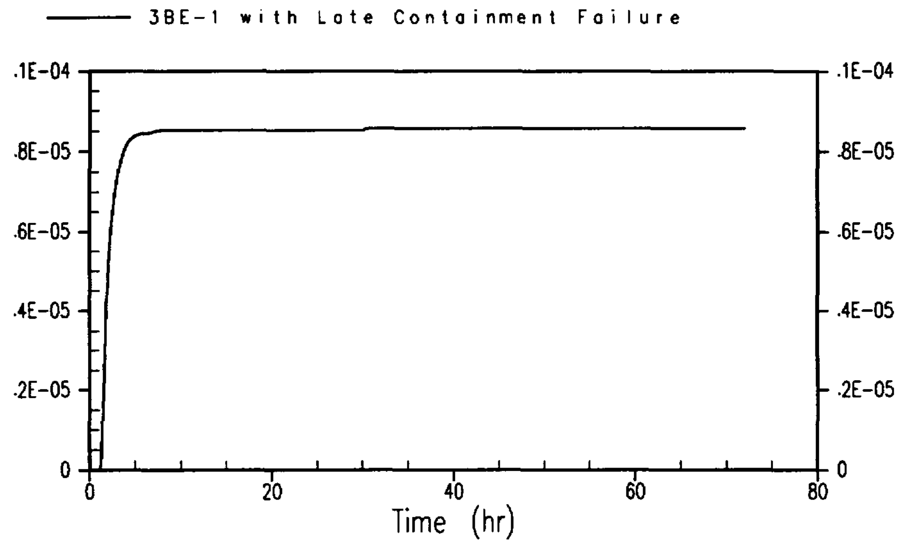


Figure 34-416

AP1000 3BE-1 Case with no PCS Water Cooling and Late Containment Failure
SrO Release Fraction

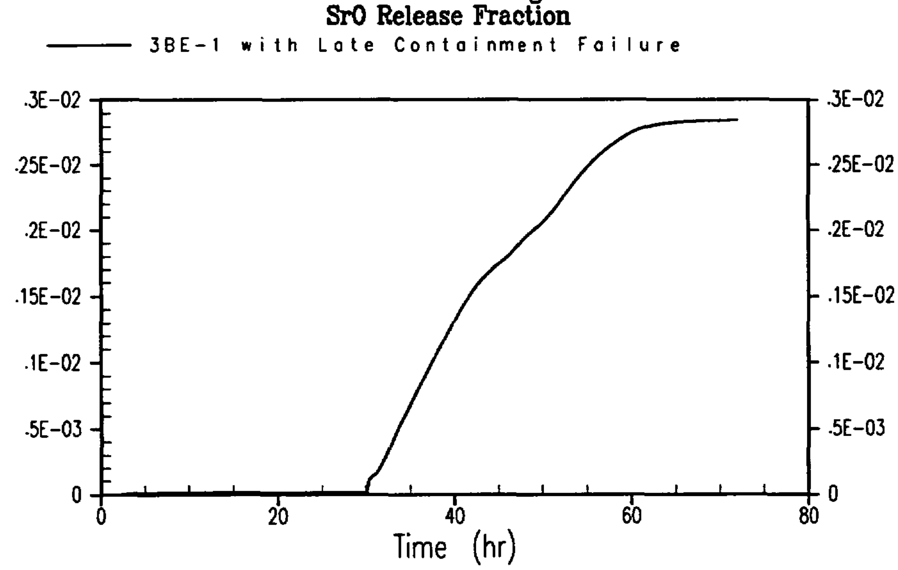


Figure 34-417

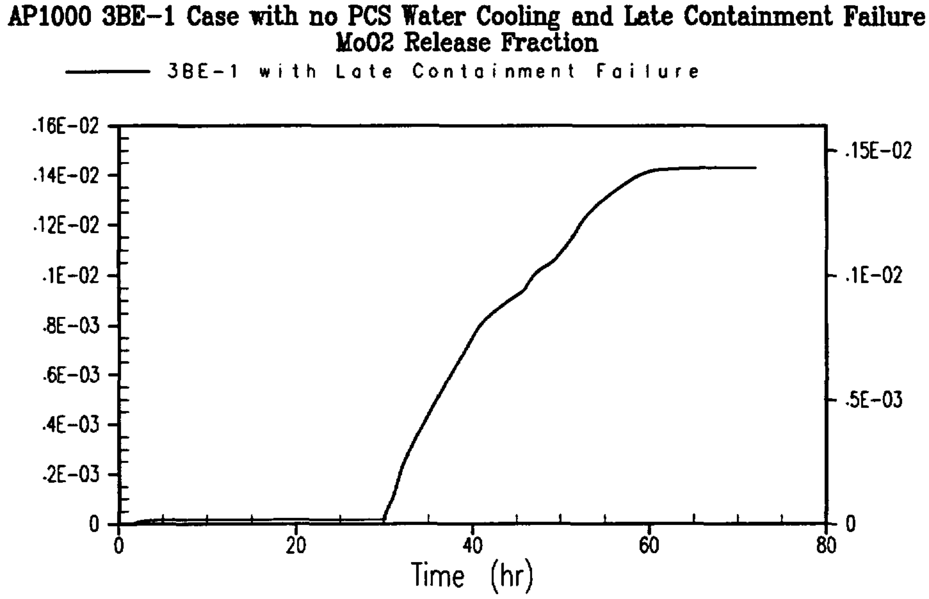


Figure 34-418

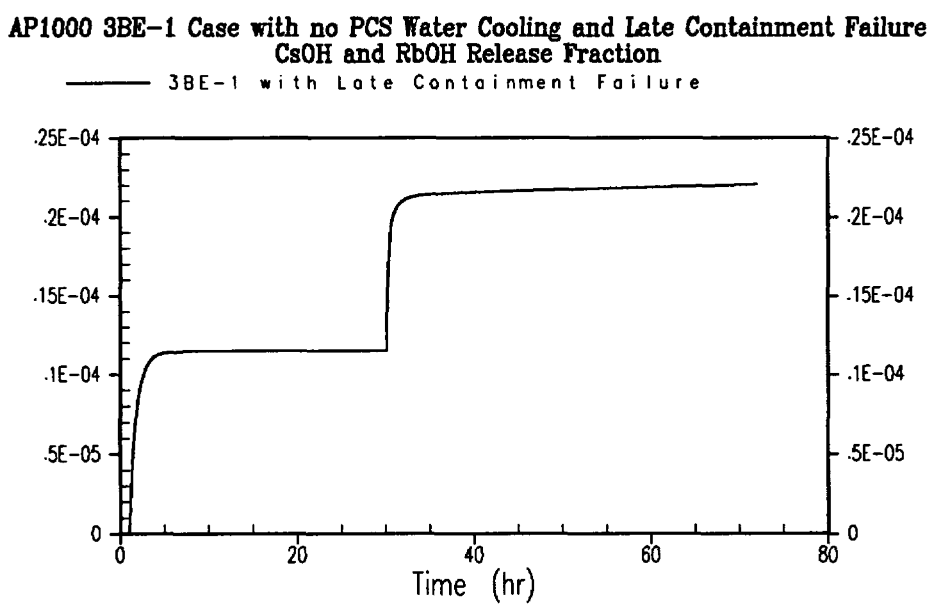


Figure 34-419

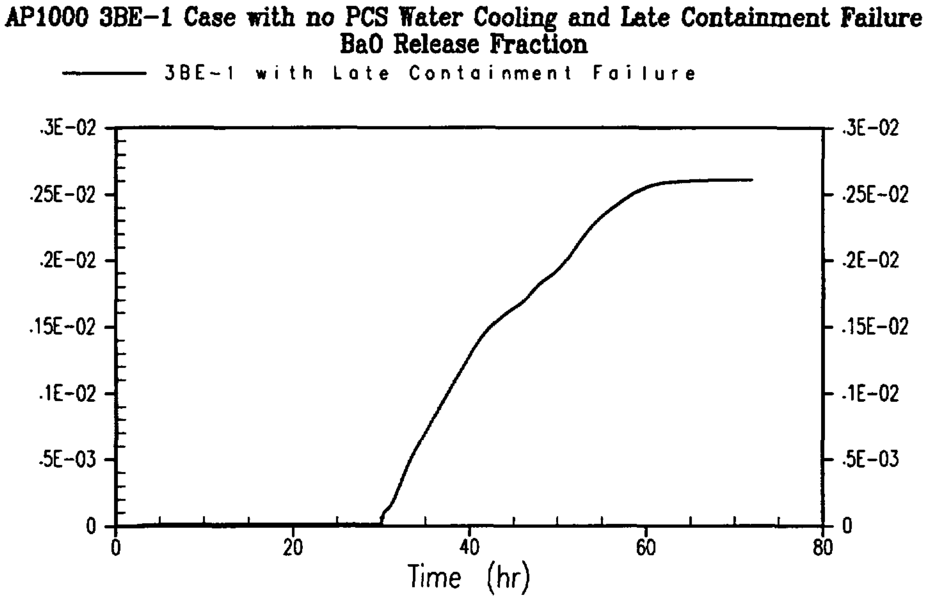


Figure 34-420

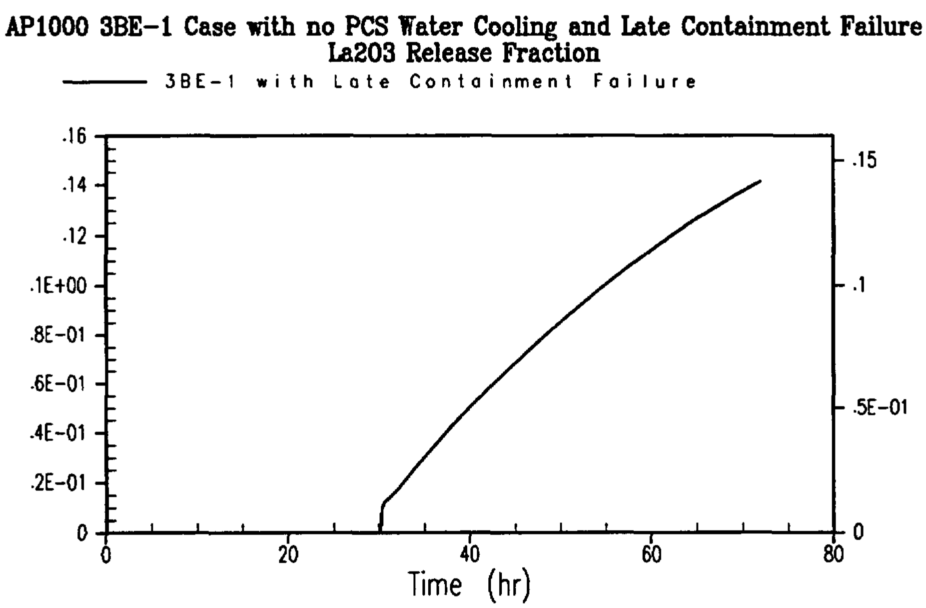


Figure 34-421

AP1000 3BE-1 Case with no PCS Water Cooling and Late Containment Failure
CeO2 Release Fraction

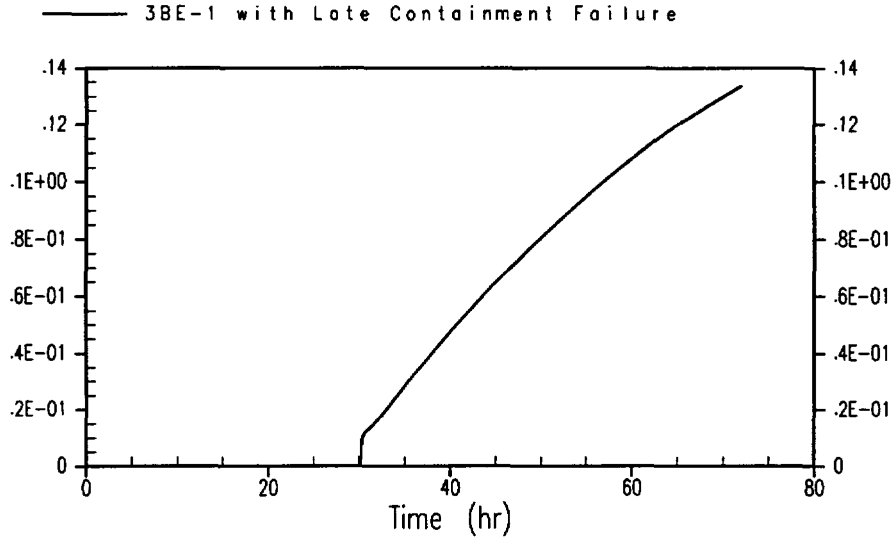


Figure 34-422

AP1000 3BE-1 Case with no PCS Water Cooling and Late Containment Failure
Sb Release Fraction

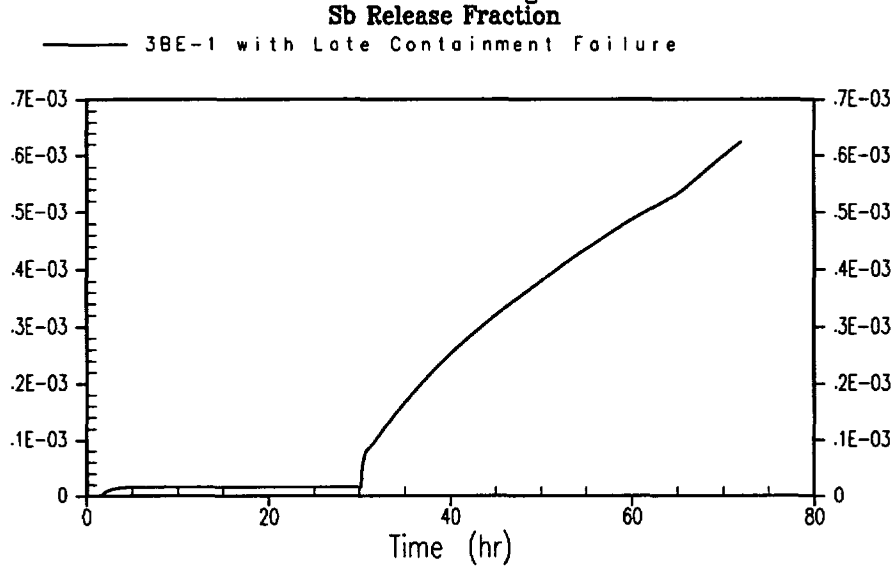


Figure 34-423

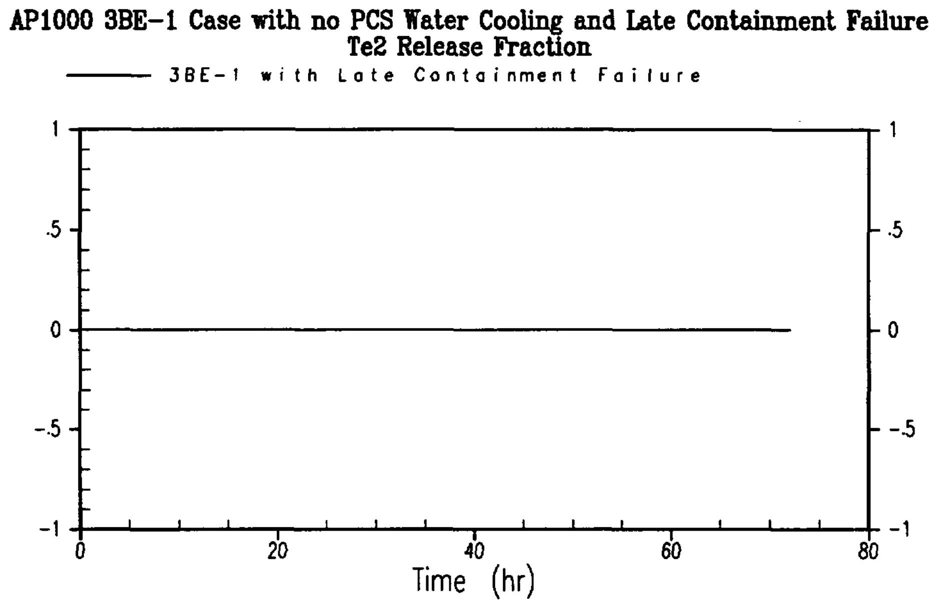


Figure 34-424

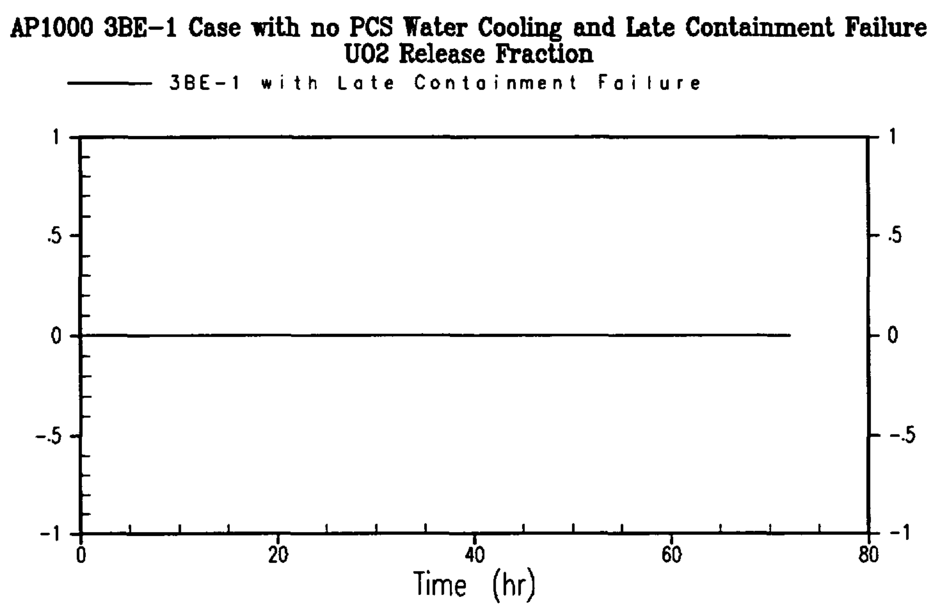


Figure 34-425

Chapter 45 Revisions to PRA Report Revision 3

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~~ **initiated** core damage **sequence** in accident class 1A are used to represent the BP release.

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

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	1.0E-3	1.2E-5	9.5E-6	1.1E-5	1.3E-5	1.1E-5	1.2E-5	1.3E-6	1.5E-6	1.3E-5	0.0E0	0.0E0
BP	1.0E-0~~6.3E-1~~	3.2E-1~~1.2E-2~~	2.5E-1~~1.9E-2~~	3.6E-3~~3.0E-4~~	4.5E-2~~5.0E-3~~	2.1E-1~~1.1E-2~~	8.9E-3~~7.9E-4~~	1.3E-4~~1.2E-5~~	8.0E-4~~3.5E-5~~	2.2E-1~~5.5E-2~~	0.0E0~~0.0E0~~	0.0E0~~0.0E0~~
CI	6.4E-1	4.6E-2	2.1E-2	2.0E-2	4.0E-2	1.8E-2	3.2E-2	2.4E-4	7.4E-4	2.7E-2	0.0E0	0.0E0
CFE	8.1E-1	5.7E-2	3.2E-2	3.5E-3	1.4E-2	5.5E-2	5.3E-3	6.5E-5	2.5E-4	2.3E-2	0.0E0	0.0E0
CFI	8.0E-1~~7.5E-1~~	3.3E-3~~8.6E-3~~	5.0e-3~~1.3E-2~~	2.2e-2~~1.42E-2~~	9.3e-3~~1.7E-2~~	3.3E-3~~8.6E-3~~	1.7E-2~~1.7E-2~~	8.3E-3~~1.2E-3~~	1.1E-2~~1.2E-3~~	7.2E-3~~1.5E-2~~	0.0E0~~0.0E0~~	0.0E0~~0.0E0~~
CFL	1.3E-3~~8.7E-1~~	1.2E-5~~3.7E-2~~	8.5E-6~~4.4E-2~~	1.7E-5~~2.0E-3~~	1.7E-5~~1.0E-2~~	1.1E-5~~2.8E-2~~	1.7E-5~~2.0E-3~~	8.5E-6~~2.3E-5~~	9.0E-6~~1.6E-4~~	1.7E-5~~2.9E-2~~	0.0E0~~0.0E0~~	0.0E0~~0.0E0~~
DIRECT	3.0E-3	3.6E-5	2.9E-5	3.3E-5	3.9E-5	3.3E-5	2.8E-5	3.9E-6	4.5E-6	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	2.6E-3	1.2E-5	9.5E-6	1.1E-5	1.3E-5	1.1E-5	1.2E-5	1.4E-6	1.5E-6	1.3E-5	0.0E0	0.0E0
BP	1.0E-0~~6.3E-1~~	4.5E-1~~1.2E-2~~	2.6E-1~~1.9E-2~~	3.6E-3~~3.0E-4~~	4.5E-2~~5.0E-3~~	2.5E-1~~1.1E-2~~	8.9E-3~~7.9E-4~~	1.3E-4~~1.2E-5~~	8.0E-4~~3.5E-5~~	2.7E-1~~5.5E-2~~	0.0E0~~0.0E0~~	0.0E0~~0.0E0~~
CI	7.8E-1	4.6E-2	2.1E-2	2.0E-2	4.0E-2	1.8E-2	2.2E-2	2.4E-4	7.4E-4	2.9E-2	0.0E0	0.0E0
CFE	9.6E-1	5.7E-2	3.2E-2	3.5E-3	1.4E-2	5.5E-2	5.3E-3	6.5E-5	2.5E-4	2.3E-2	0.0E0	0.0E0
CFI	9.2E-1~~8.5E-1~~	3.3E-3~~8.6E-3~~	5.0E-3~~1.3E-2~~	2.2E-2~~1.5E-2~~	9.3E-3~~1.7E-2~~	3.3E-3~~8.6E-3~~	1.7E-2~~1.8E-2~~	1.9E-2~~2.1E-3~~	2.1E-2~~2.2E-3~~	7.3E-3~~1.5E-2~~	0.0E0~~0.0E0~~	0.0E0~~0.0E0~~
CFL	9.8E-1~~4.7E-1~~	3.3E-5~~3.7E-2~~	8.6E-6~~4.4E-2~~	2.8E-3~~2.0E-3~~	1.4E-3~~1.0E-2~~	2.2E-5~~2.8E-2~~	2.6E-3~~3.0E-3~~	1.4E-1~~3.3E-5~~	1.3E-1~~1.6E-4~~	6.2E-4~~2.9E-2~~	0.0E0~~0.0E0~~	0.0E0~~0.0E0~~
DIRECT	7.8E-3	3.6E-5	2.9E-5	3.3E-5	3.9E-5	3.3E-5	3.6E-5	4.2E-6	4.5E-6	3.9E-5	0.0E0	0.0E0

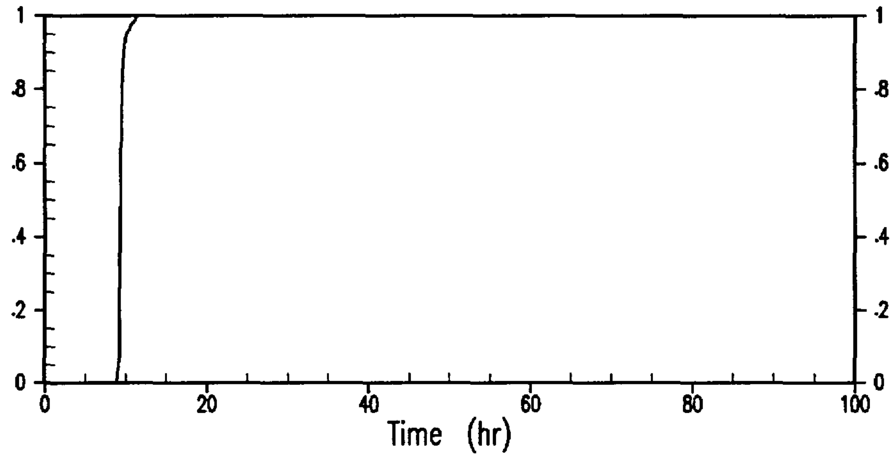


Figure 45-13

Release Category BP, Case 6E-1 – SGTR with Stuck Open SG Safety Valve: Release Fraction of Noble Gases

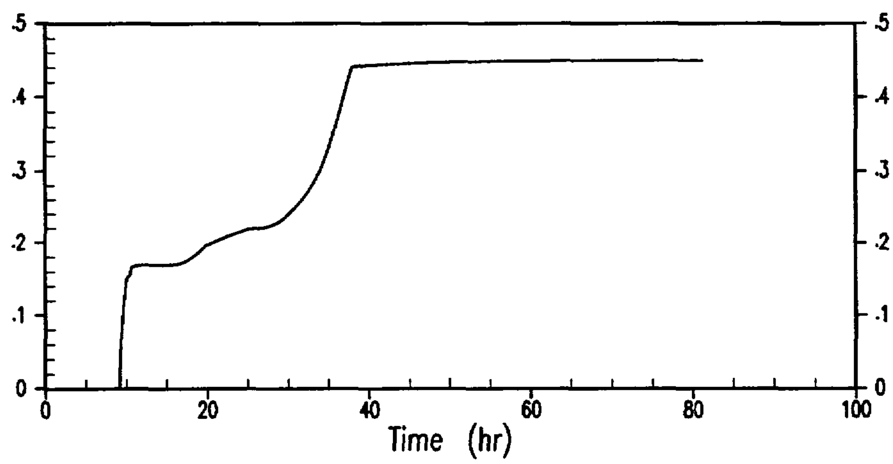


Figure 45-14

Release Category BP, Case 6E-1 – SGTR with Stuck Open SG Safety Valve: Release Fraction of Cesium Iodide

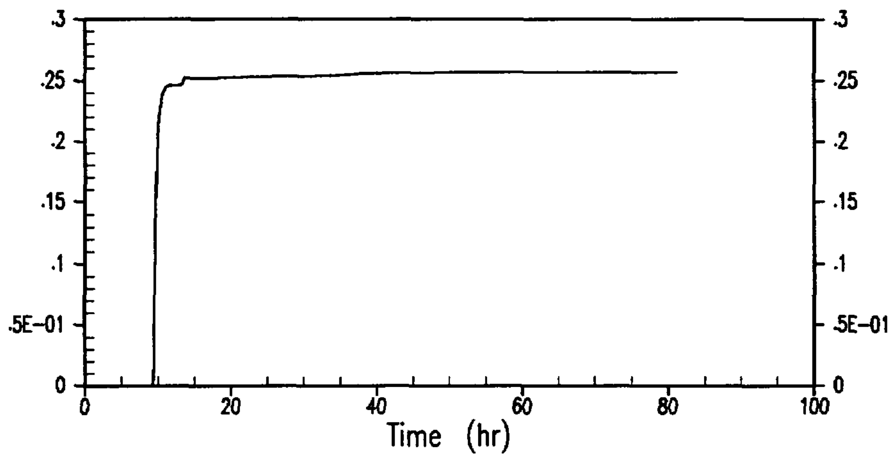


Figure 45-15

Release Category BP, Case 6E-1 – SGTR with Stuck Open SG Safety Valve: Release Fraction of Tellurium Dioxide

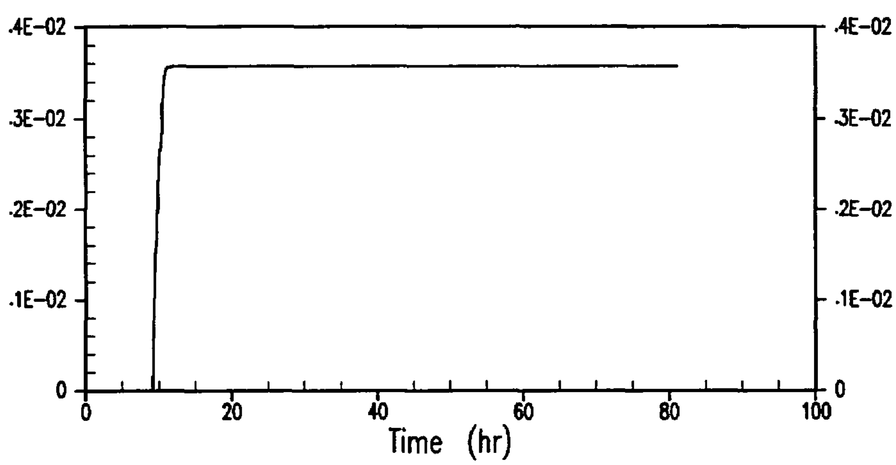


Figure 45-16

Release Category BP, Case 6E-1 – SGTR with Stuck Open SG Safety Valve: Release Fraction of Strontium Oxide

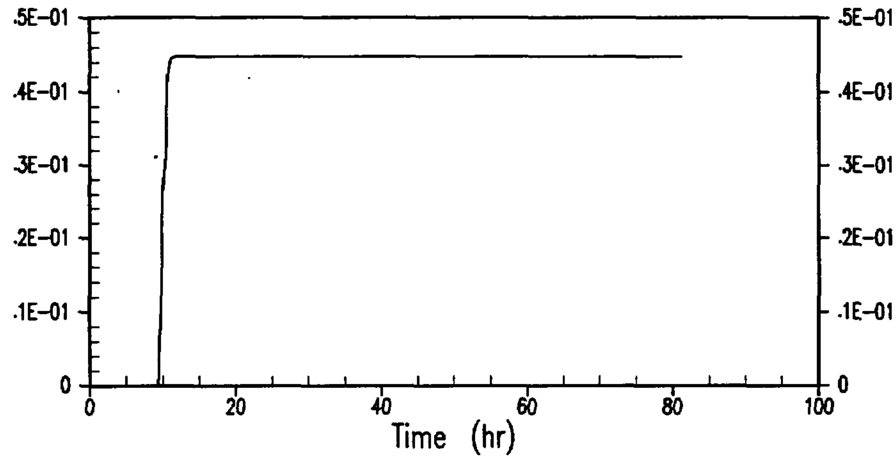


Figure 45-17

Release Category BP, Case 6E-1 – SGTR with Stuck Open SG Safety Valve: Release Fraction of Molybdenum Dioxide

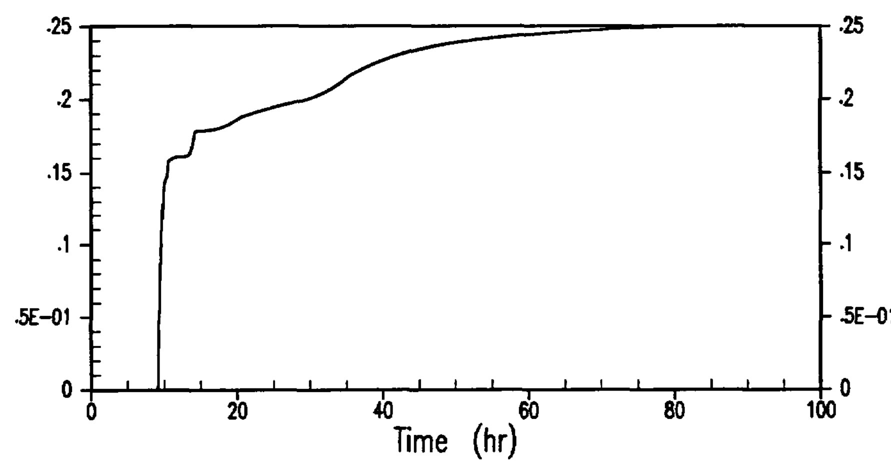


Figure 45-18

Release Category BP, Case 6E-1 – SGTR with Stuck Open SG Safety Valve: Release Fraction of Cesium Hydroxide

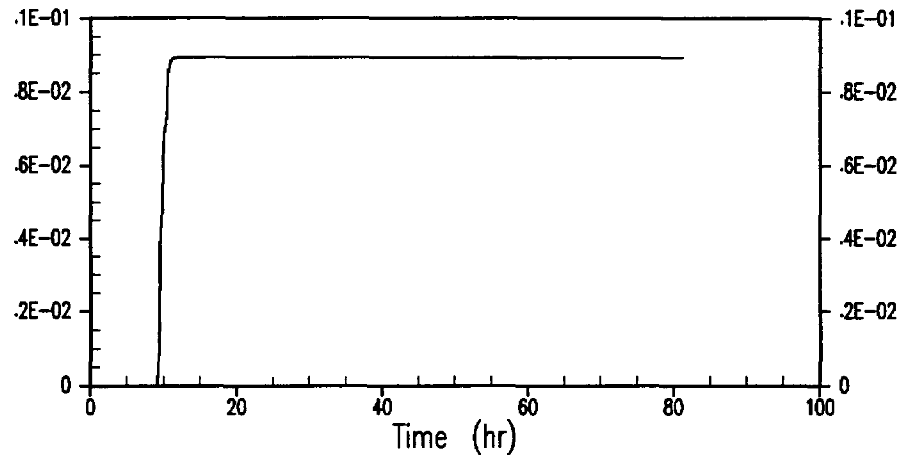


Figure 45-19

Release Category BP, Case 6E-1 – SGTR with Stuck Open SG Safety Valve: Release Fraction of Barium Oxide

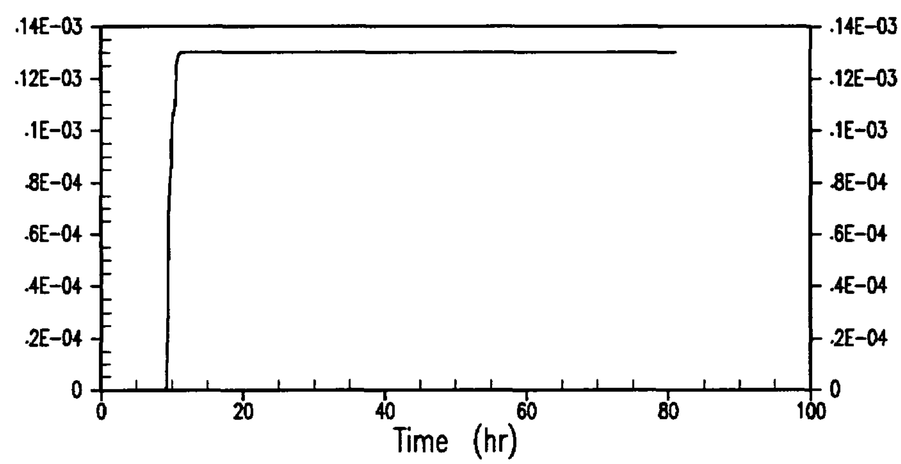


Figure 45-20

Release Category BP, Case 6E-1 – SGTR with Stuck Open SG Safety Valve: Release Fraction of Dilanthanum Trioxide

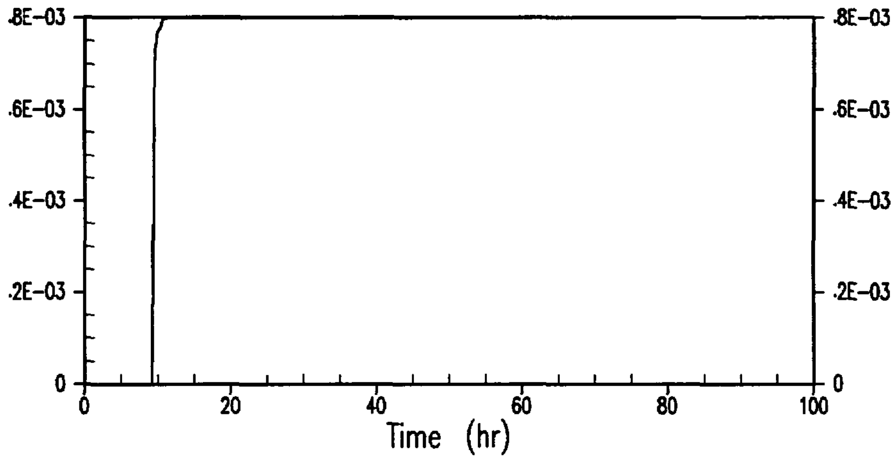


Figure 45-21

Release Category BP, Case 6E-1 – SGTR with Stuck Open SG Safety Valve: Release Fraction of Cerium Dioxide

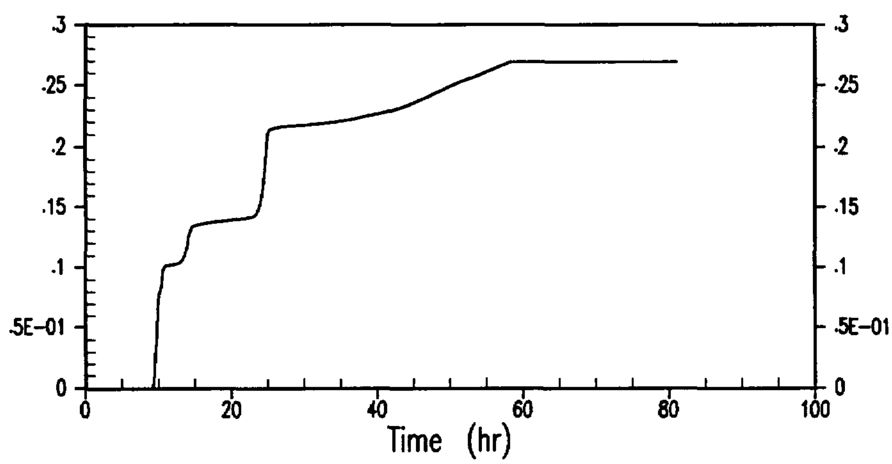


Figure 45-22

Release Category BP, Case 6E-1 – SGTR with Stuck Open SG Safety Valve Creep of SG Tubes: Release Fraction of Tin

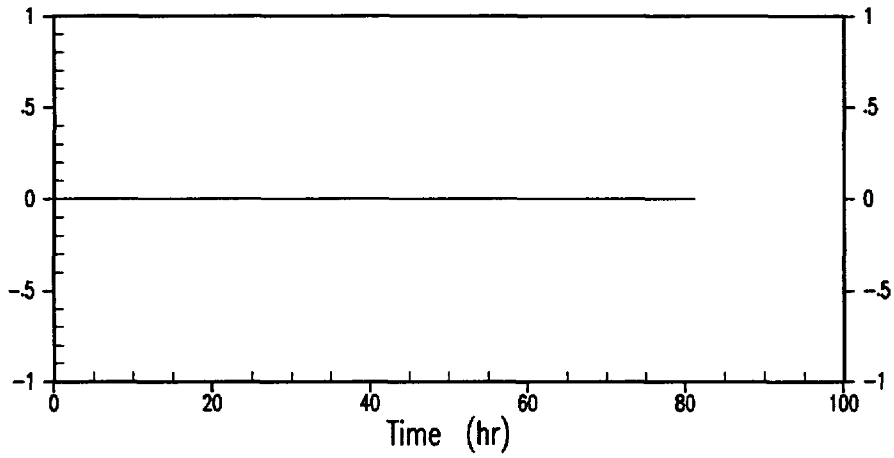


Figure 45-23

Release Category BP, Case 6E-1 – SGTR with Stuck Open SG Safety Valve: Release Fraction of Tellurium

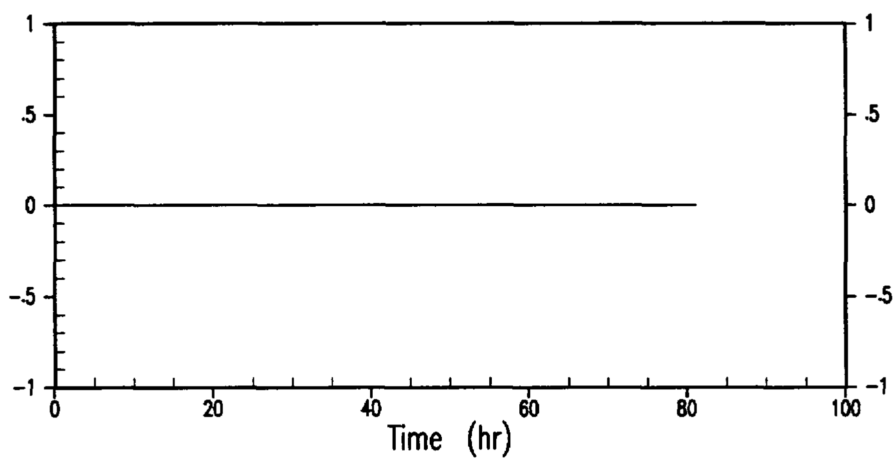


Figure 45-24

Release Category BP, Case 6E-1 – SGTR with Stuck Open SG Safety Valve: Release Fraction of Uranium Dioxide

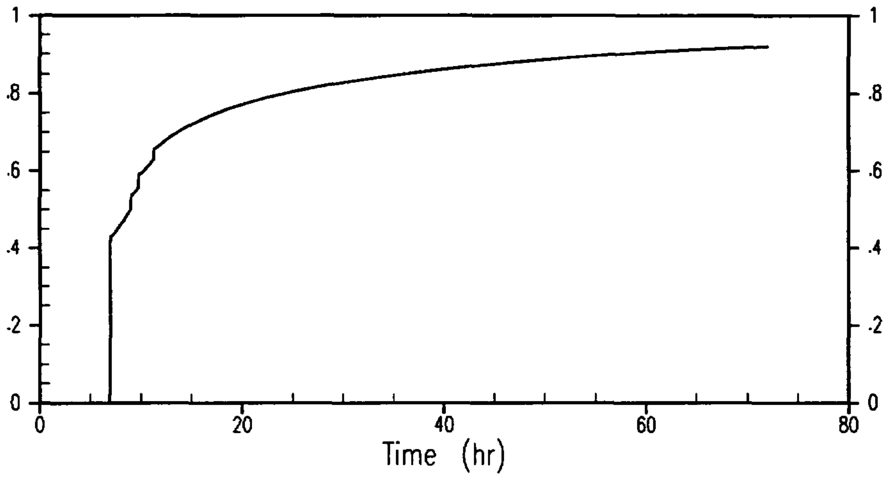


Figure 45-49

Release Category CFI, Case 3BE-3 – DVI Line Break, Failed Gravity Injection, No PXS Flooding: Release Fraction of Noble Gases

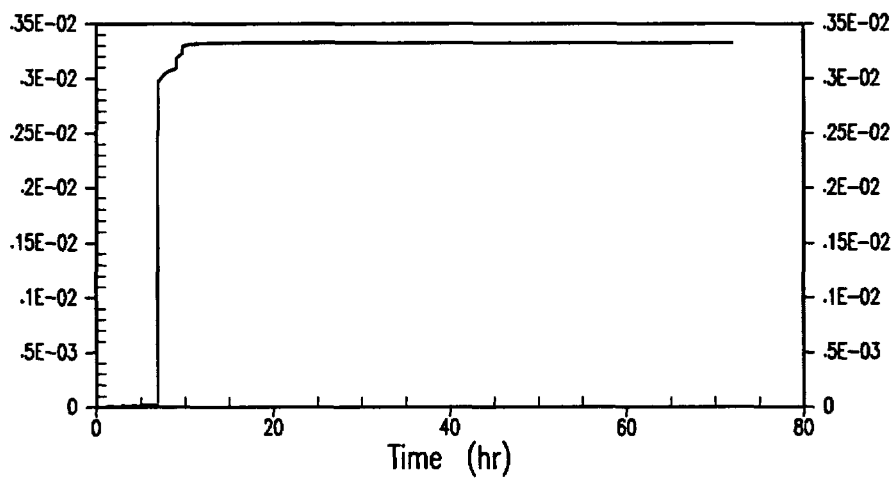


Figure 45-50

Release Category CFI, Case 3BE-3 – DVI Line Break, Failed Gravity Injection, No PXS Flooding: Release Fraction of Cesium Iodide

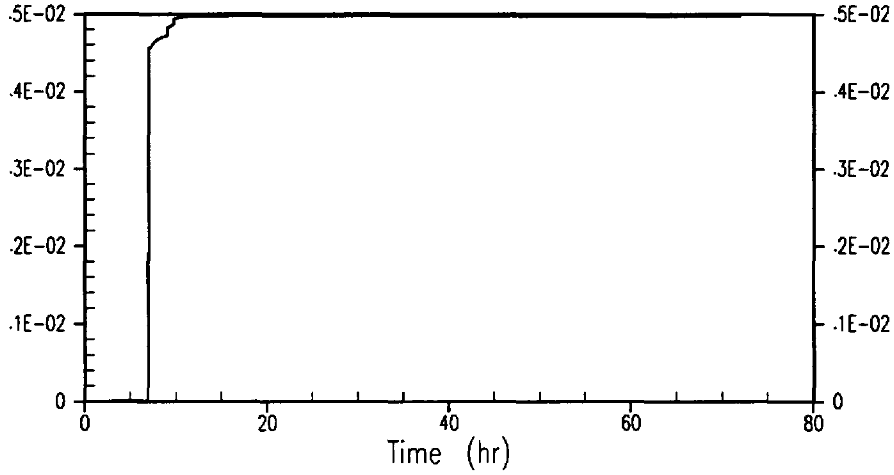


Figure 45-51

Release Category CFI, Case 3BE-3 – DVI Line Break, Failed Gravity Injection, No PXS Flooding: Release Fraction of Tellurium Dioxide

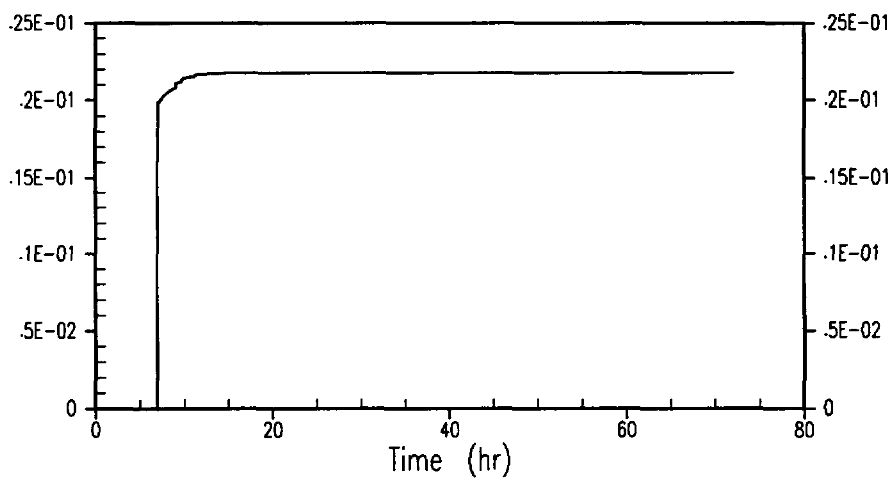


Figure 45-52

Release Category CFI, Case 3BE-3 – DVI Line Break, Failed Gravity Injection, No PXS Flooding: Release Fraction of Strontium Oxide

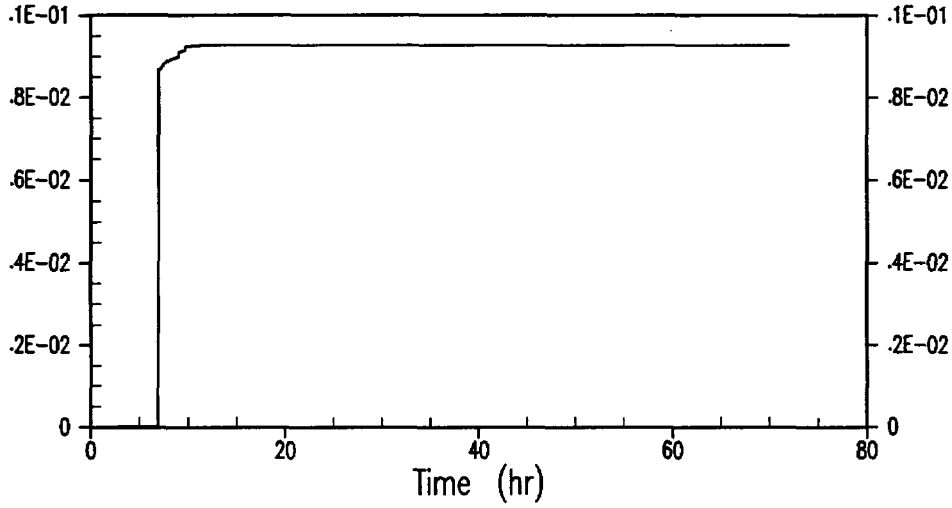


Figure 45-53

Release Category CFI, Case 3BE-3 – DVI Line Break, Failed Gravity Injection, No PXS Flooding: Release Fraction of Molybdenum Dioxide

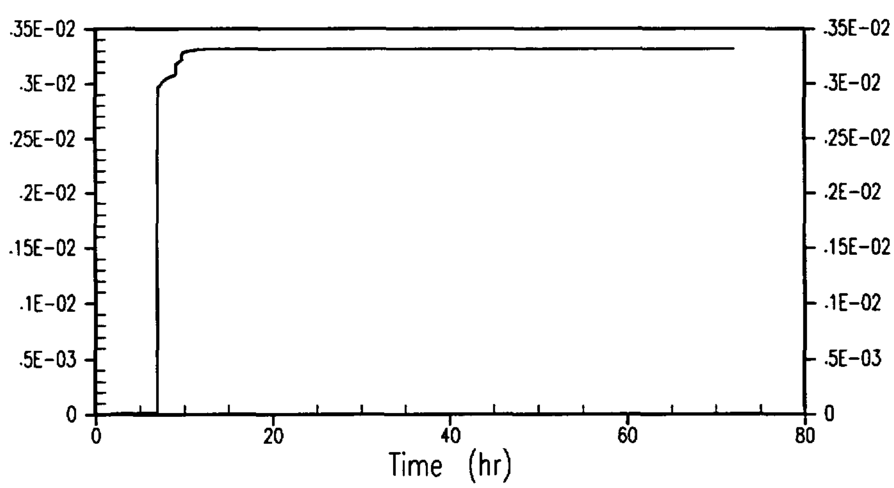


Figure 45-54

Release Category CFI, Case 3BE-3 – DVI Line Break, Failed Gravity Injection, No PXS Flooding: Release Fraction of Cesium Hydroxide

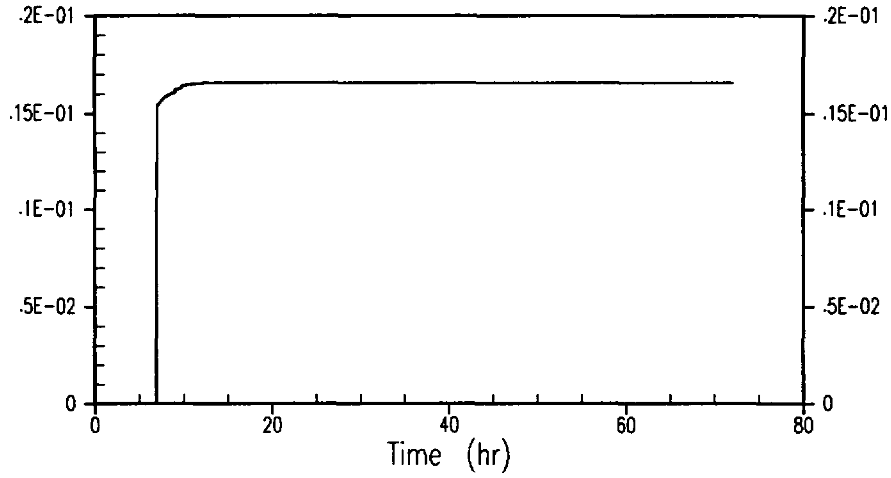


Figure 45-55

Release Category CFI, Case 3BE-3 – DVI Line Break, Failed Gravity Injection, No PXS Flooding: Release Fraction of Barium Oxide

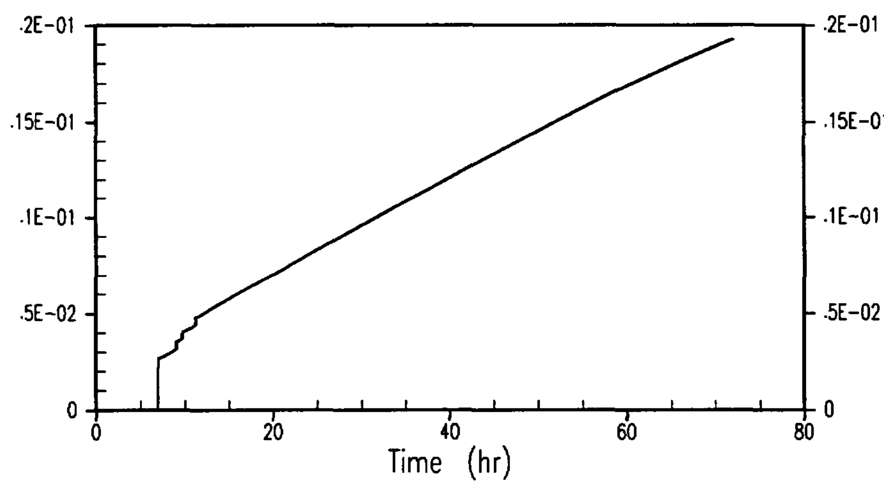


Figure 45-56

Release Category CFI, Case 3BE-3 – DVI Line Break, Failed Gravity Injection, No PXS Flooding: Release Fraction of Dilanthanum Trioxide

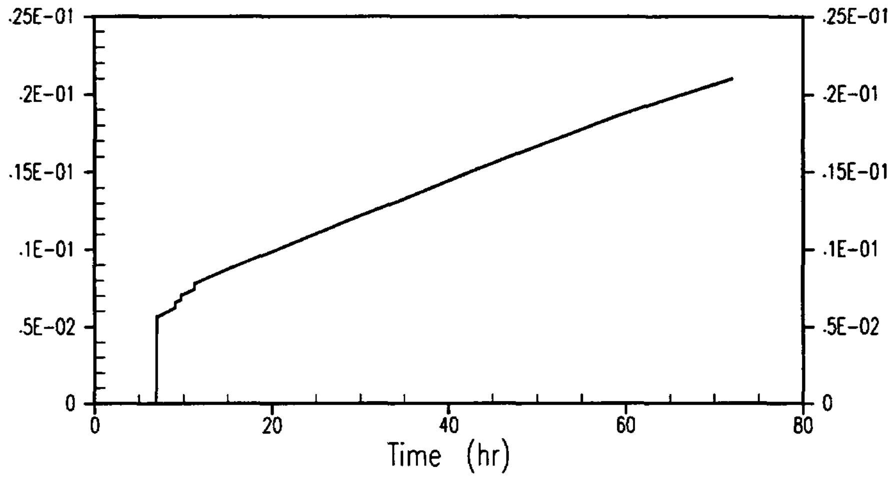


Figure 45-57

Release Category CFI, Case 3BE-3 – DVI Line Break, Failed Gravity Injection, No PXS Flooding: Release Fraction of Cerium Dioxide

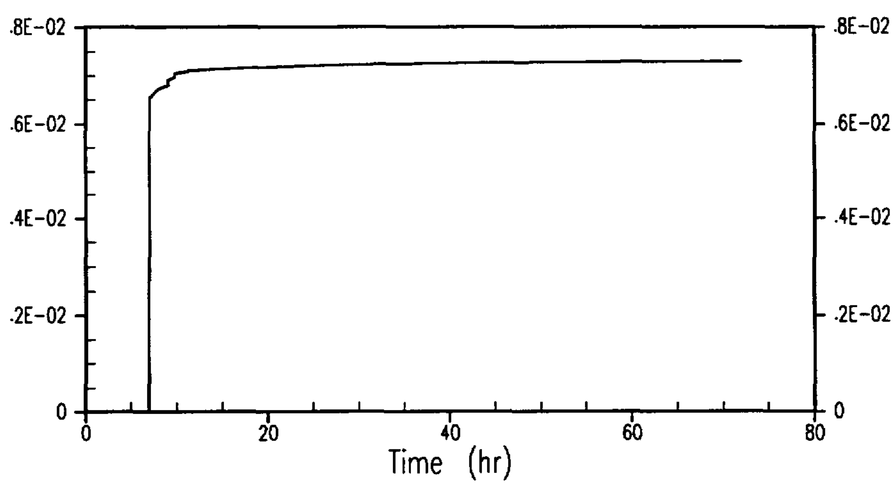


Figure 45-58

Release Category CFI, Case 3BE-3 – DVI Line Break, Failed Gravity Injection, No PXS Flooding: Release Fraction of Tin

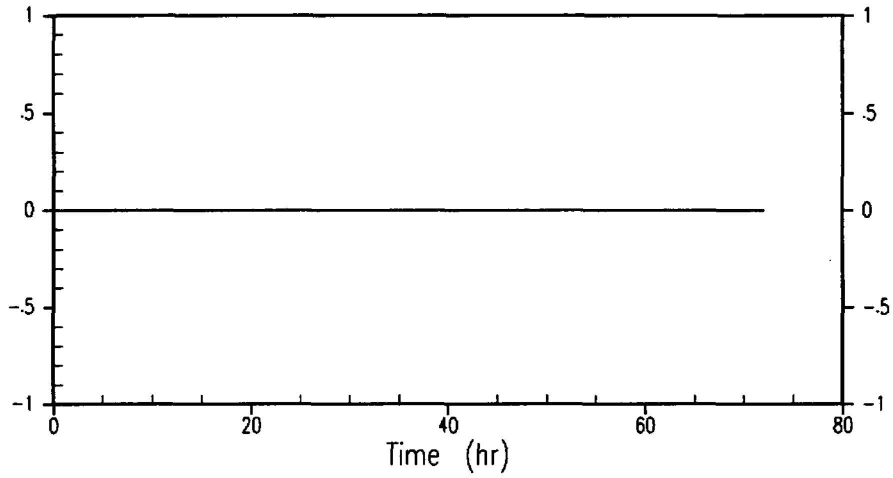


Figure 45-59

Release Category CFI, Case 3BE-3 – DVI Line Break, Failed Gravity Injection, No PXS Flooding: Release Fraction of Tellurium

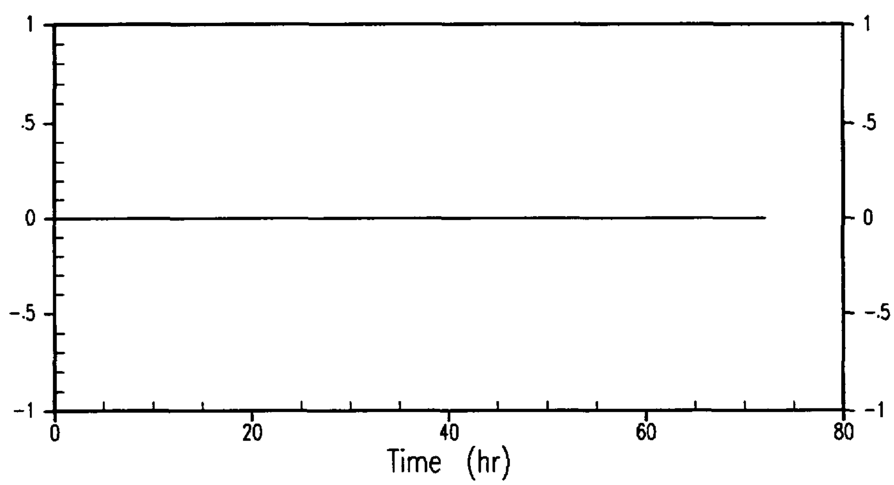


Figure 45-60

Release Category CFI, Case 3BE-3 – DVI Line Break, Failed Gravity Injection, No PXS Flooding: Release Fraction of Uranium Dioxide

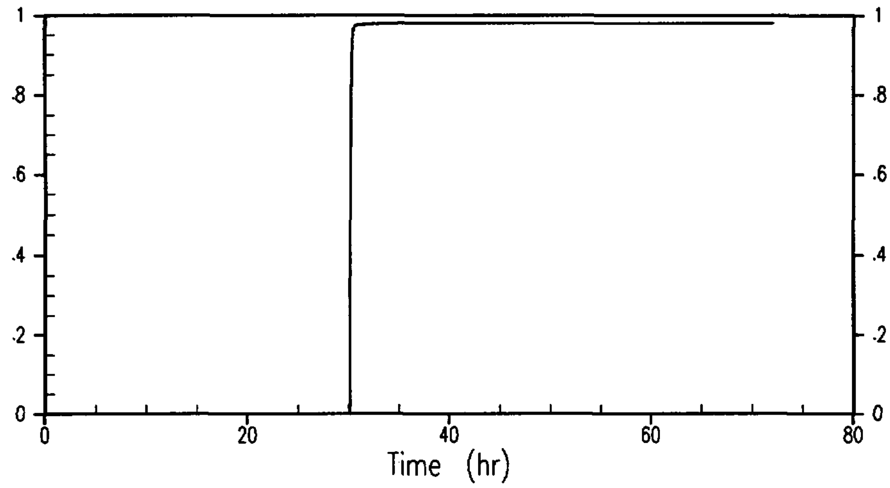


Figure 45-61

Release Category CFL, Case 3BE-7 – SBLOCA with Failed Gravity Injection: Release Fraction of Noble Gases

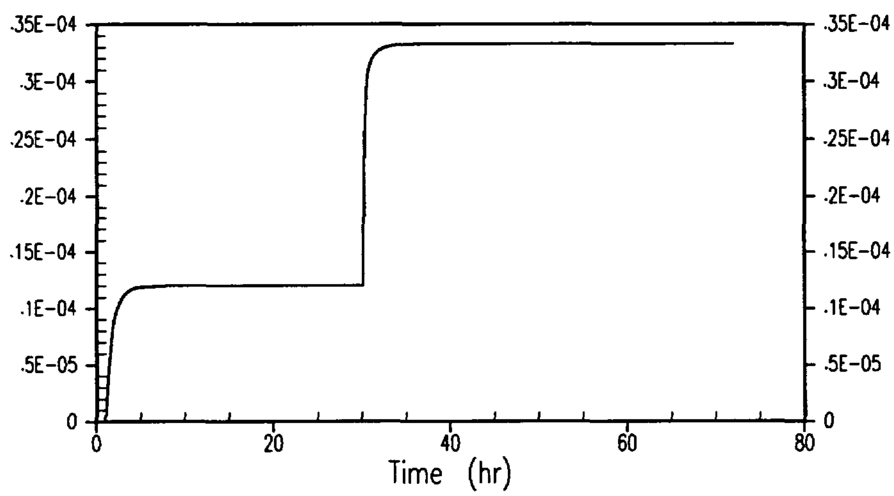


Figure 45-62

Release Category CFL, Case 3BE-7 – SBLOCA with Failed Gravity Injection: Release Fraction of Cesium Iodide

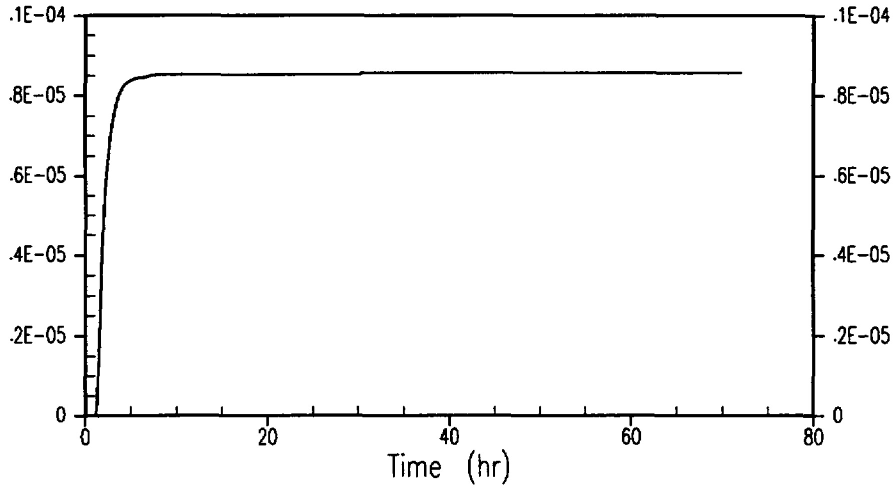


Figure 45-63

Release Category CFL, Case 3BE-7 – SBLOCA with Failed Gravity Injection: Release Fraction of Tellurium Dioxide

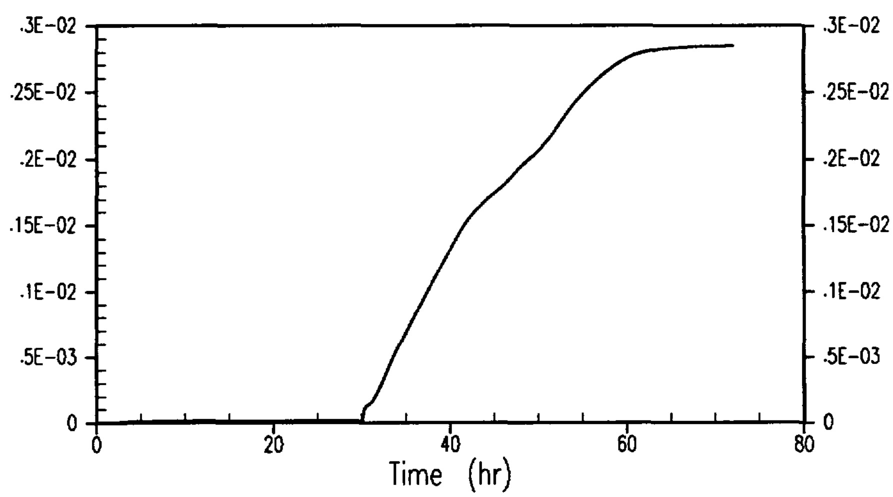


Figure 45-64

Release Category CFL, Case 3BE-7 – SBLOCA with Failed Gravity Injection: Release Fraction of Strontium Oxide

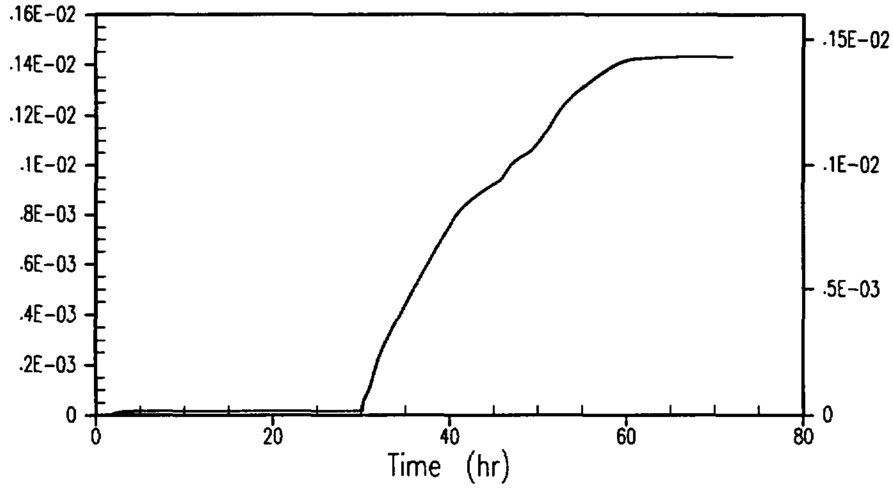


Figure 45-65

Release Category CFL, Case 3BE-7 – SBLOCA with Failed Gravity Injection: Release Fraction of Molybdenum Dioxide

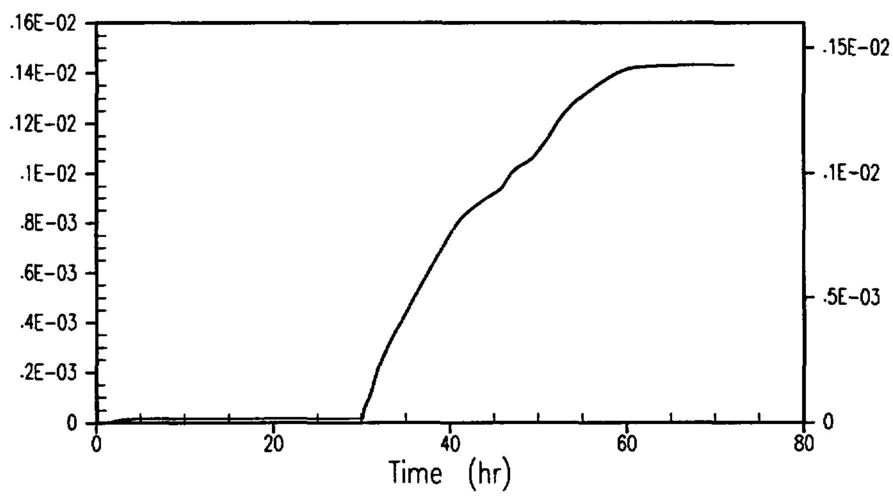


Figure 45-66

Release Category CFL, Case 3BE-7 – SBLOCA with Failed Gravity Injection: Release Fraction of Cesium Hydroxide

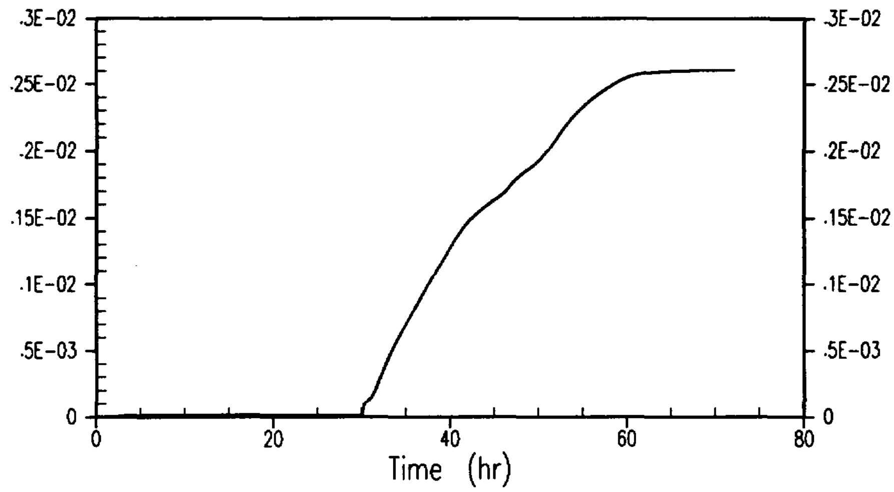


Figure 45-67

Release Category CFL, Case 3BE-7 – SBLOCA with Failed Gravity Injection: Release Fraction of Barium Oxide

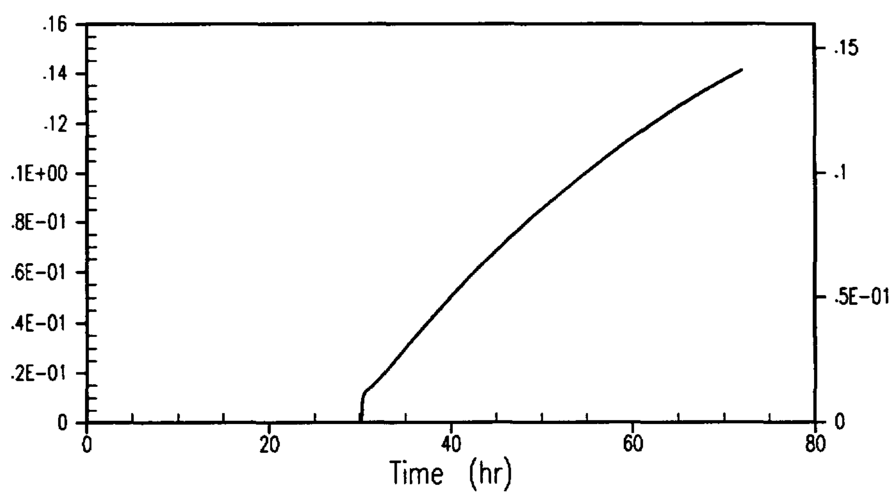


Figure 45-68

Release Category CFL, Case 3BE-7 – SBLOCA with Failed Gravity Injection: Release Fraction of Dilanthanum Trioxide

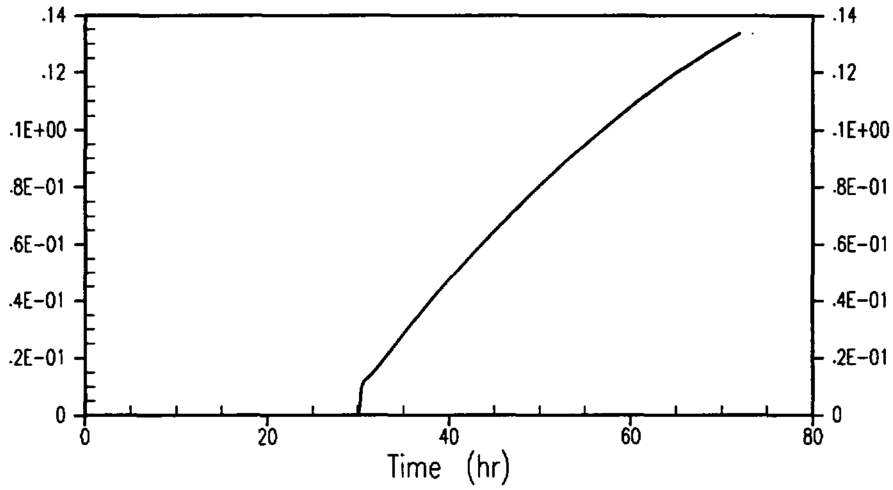


Figure 45-69

Release Category CFL, Case 3BE-7 – SBLOCA with Failed Gravity Injection: Release Fraction of Cerium Dioxide

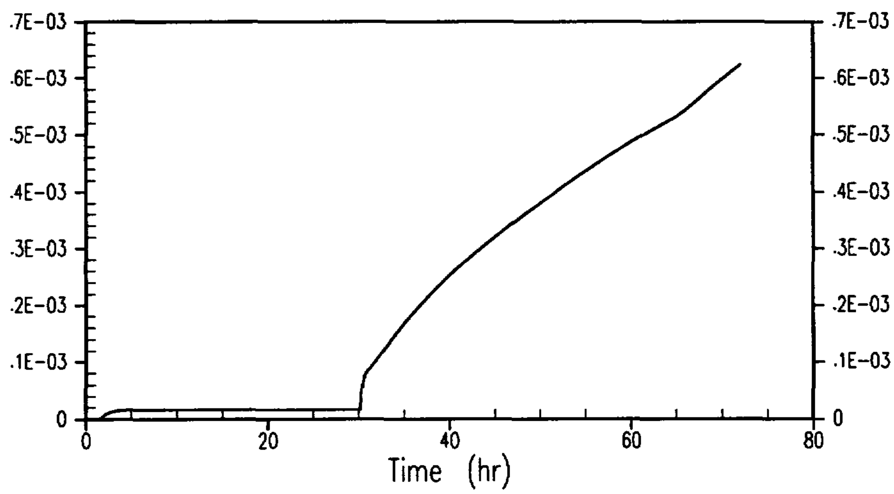


Figure 45-70

Release Category CFL, Case 3BE-7 – SBLOCA with Failed Gravity Injection: Release Fraction of Tin

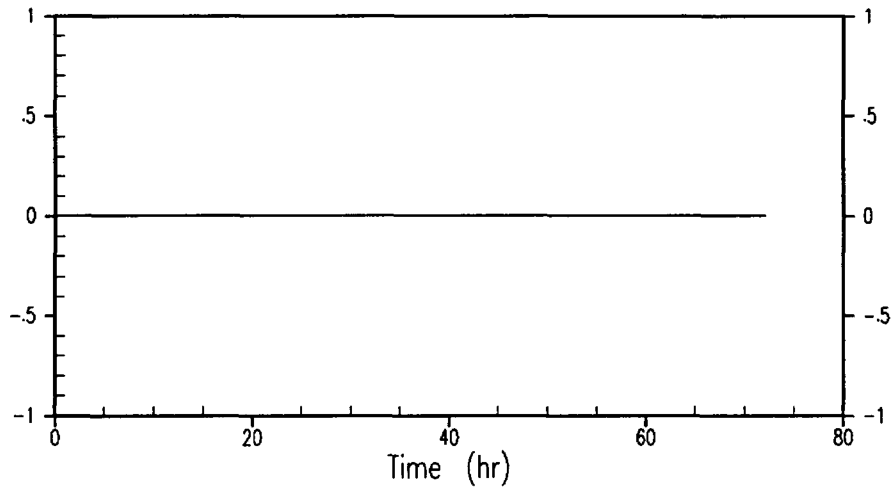


Figure 45-71

Release Category CFL, Case 3BE-7 – SBLOCA with Failed Gravity Injection: Release Fraction of Tellurium

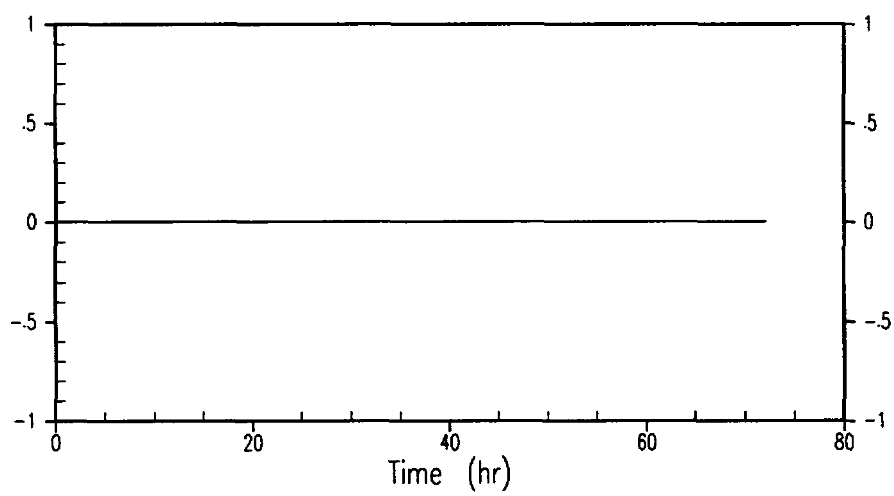


Figure 45-72

Release Category CFL, Case 3BE-7 – SBLOCA with Failed Gravity Injection: Release Fraction of Uranium Dioxide

REVISIONS TO PRA REPORT REVISION 3 CHAPTER 49

OFFSITE DOSE RISK QUANTIFICATION

49.1 Introduction

The potential ground-level exposure, expressed as both effective dose equivalent (EDE) whole-body dose and acute red bone marrow dose, resulting from the possible accidental release of radioactive fission products is discussed in this chapter. Chapter 45 provides the estimated source term information; that is, the accidental release conditions. These conditions are:

- The amount of released material
- Release energy
- Duration
- Location

for the six identified release categories.

This information is used here, along with data provided in the Electric Power Research Institute (EPRI) Advanced Light Water Reactor Utility Requirements Document (Reference 49-1), to perform the atmospheric dispersion analyses. These analyses are conducted to estimate the EDE whole-body dose and acute red bone marrow dose, both at the site boundary (0.5 miles). The population whole-body dose out to 80.5 kilometers and the downwind, centerline, ground-level thyroid dose at the site boundary (0.5 miles) are also calculated for information. The estimated site boundary whole-body dose and the acute red bone marrow dose are compared to the Westinghouse goal of <25 rems (0.25 sieverts), at a frequency not to exceed $1x10^{-6}$ per year. This is consistent with the goal provided in Reference 49-1.

It should be noted that Reference 49-1 recommends evaluation of the whole-body dose. However, it does not clearly identify whether this dose analysis should be based on an acute or committed dose (EDE) basis. Consequence codes such as the MELCOR Accident Consequence Code System (MACCS and MACCS2) and their predecessor CRAC2 (codes are recommended by Reference 49-1) can only calculate the EDE whole-body dose, therefore the committed dose has been used in previous ALWR analyses. It is felt however, that the whole-body acute centerline dose is more appropriate for this 25 rem dose calculation since the purpose of this calculation is to establish a margin to the Nuclear Regulatory Commission (NRC) safety goals and the NRC staff safety goal implementation requirement. In this context, it is acute health effects versus long term effects from a committed dose that are of significance. For consequence codes such as MACCS2, the acute red bone marrow dose may be used to represent the acute whole-body dose. These doses are determined at the site boundary (0.5-mile radius).

The thyroid (site boundary) and whole-body (population) doses are also calculated during the first 24 and 72 hours following the onset of core damage, based on the probabilistic atmospheric dispersion analysis of the dose associated with each release category, coupled with multiple meteorological conditions. The thyroid and population doses are provided for information.

49.2 Conformance with Regulatory Requirements

MACCS2 version 1.12 (Reference 49-2) is used for the analysis. The NRC sponsored the development of this code. The code performs probabilistic estimates of offsite consequences from potential accidental releases in conformance with Chapter 9 of the probabilistic risk assessment (PRA) guidelines described in NUREG/CR-2300 (Reference 49-3).

The analysis was based on the Westinghouse design goals, which are consistent with the guidelines provided in Reference 49-1, as discussed above. This reference document also identifies use of the MACCS2 code for offsite consequence analysis.

49.3 Assumptions

This section discusses the information, including assumptions, required to perform the dose evaluation. The primary information required for the dose evaluation includes the release source terms; the site meteorological data; population distribution data; site economic data; agricultural and land use data; and food uptake, ingestion, and retention factors. Additionally, available data on site emergency plans, such as sheltering and evacuation, and site decontamination and interdiction plans, may be included in the dose evaluation. Since the Westinghouse design goal specifies the site boundary dose as the only consequence of concern, the population, land use, sheltering, evacuation, decontamination, and interdiction data are not required for this calculation.

The advanced light water reactor reference site information described in Reference 49-1 provides the meteorological and population data for the analysis. Since the advanced light water reactor site data does not provide sufficient topographical data to define the MACCS2 site input file, the site land use and crop data are based on representative data from the Surry Plant Site. These data are provided in Reference 49-2. Due to the proximity of the Surry Site to the ocean, those site sectors that are ocean were arbitrarily changed to land. This was done to allow use of the advanced light water reactor reference site population data (without having people assigned to ocean sectors). These changes to the land and crop characteristics are made to provide an acceptable MACCS2 input file. They have no effect on the calculated dose at the plant site boundary.

49.4 Methodology

The dose evaluation uses the MACCS2 accident consequence code to estimate the potential offsite effects of the postulated accidental releases, developed by the Level 2 analysis. The MACCS2 code performs multiple air dispersion analyses, based on the yearly meteorological data, to estimate the air and ground-level concentrations of the released nuclides of concern. Multiple dispersion analyses allow the application of statistical analysis to the full range of results, based on the probability of the meteorological sequences that caused those results. This accounts for the possibility of an accident occurring at any time during the year. The air and ground-level concentrations are then converted to exposure dose, per nuclide, for the following pathways: cloudshine, groundshine, inhalation (direct and resuspended material), and ingestion. For the potential exposure during the initial 24 and 72 hours, the calculated dose does not consider the ingestion pathway.

The MACCS2 code permits evaluation of the effects associated with direct exposure to the radioactive cloud (that is, cloudshine, groundshine, and inhalation), during the period initially following the accident (up to 1 week), and the long-term (over many years) effects due to exposure to contaminated land (ingestion of local farm products, ground shine, resuspension inhalation). It also examines accident costs, which might include permanent relocation and/or decontamination. The code also permits the modeling of the protective effects of sheltering and/or evacuation of the population during the acute exposure phase.

The Westinghouse goal only requires dose determination for exposure resulting from the first 24 hours following the initiation of core damage. Additionally, the Westinghouse goal requires only the total dose to a hypothetical individual located at the site boundary, which is assumed to be one-half mile, directly downwind, during the entire exposure period. Therefore, dose calculations related to the actual site population distribution are not required, nor are calculation of potential health effects, such as deaths and cancers. Finally, the calculation of the site boundary dose ignores any potential mitigating effects, including sheltering and evacuation.

Therefore, the consequence level evaluated in this analysis includes the whole-body effective dose equivalent dose and the acute red bone marrow dose resulting from the first 24-hour exposure versus distance from the reactor.

Statistical evaluation is applied to the multiple dispersion analysis results so that the consequences are presented in terms of a mean value, a peak value, and as complementary cumulative distribution functions. These functions present the value of the consequence level (whole-body effective dose equivalent dose) versus the probability of exceeding this level. The Westinghouse goal and the Reference 49-1 guidelines provide a value for the site boundary dose, not to exceed 25 rems whole-body dose at a frequency not to exceed 1×10^{-6} events per year.

A brief description of the code follows.

The MACCS2 code performs its processing in three steps, or modules: ATMOS, EARLY, and CHRONC. The description of the source term and the dispersion calculations occur in the first module, i.e., the ATMOS module. The EARLY module performs the calculations relating to the initial exposure dose and can also account for sheltering or evacuation schemes. The CHRONC module performs the calculations relating to the long-term exposure dose (for many years) and can account for decontamination or food uptake parameters. Only the ATMOS and EARLY modules are used for this analysis.

The MACCS2 code models the atmospheric transport of fission products that are released from containment, as defined by the source term characteristics, using a Gaussian plume model, and calculates the air and ground-level concentrations for the radionuclides of concern. Vertical plume rise depends on the release energy. Plume motion depends on the available meteorological conditions; that is, wind speed, wind direction, and atmospheric stability. The code includes models for radioactive decay and daughter product buildup, wet and dry deposition of the nuclides due to gravity settling, and washout due to precipitation. Noble gases are not removed by deposition.

The MACCS2 code first reviews the hourly meteorological data for one year and sorts the data into predefined and user-defined meteorological categories. This allows MACCS2 to assess the frequency of occurrence of the different meteorological types, and to provide a realistic representation of a full year of site weather. It does this without overlooking those meteorological conditions that, although infrequent, may be instrumental in producing peak impacts. The probability of each meteorological category is also determined by this analysis.

In performing the dispersion analysis for a specific source term, the code samples each of the meteorological categories several times. The number of sampling per category is specified by the user. Each sample consists of starting the postulated release during one of the 8,760 hours during the year, which is identified with the meteorological type being sampled. This is done for each of the meteorological types. For example, if the user specifies four samples per meteorological category, and there are 30 defined meteorological categories, and if the database has at least four hours of meteorological samples per category, then 4 times 30, or 120, dispersion analyses are performed by MACCS2.

Once the release start time is selected, then the actual meteorological data is used to model the subsequent dispersion. That is, the meteorological data are allowed to change as the material moves downwind. The calculation continues until the material reaches the boundary of the spatial grid (receptor grid) defined by the user. Each dispersion simulation, therefore, results in calculated, integrated air and ground concentrations, (plume centerline, ground level) as a function of downwind distance. Each analysis is then weighted by the probability of occurrence of the meteorological condition. As each calculation is performed, the results are accumulated to provide an average estimate of the downwind integrated air and ground concentrations, including effects from all possible meteorological types. The MACCS2 code also notes the peak downwind concentration at each receptor distance and the associated meteorological condition that produced the peak.

The MACCS2 code then performs conversion calculations to estimate the radiation doses based on the air and ground concentrations. The radiation doses received by individuals are due to the passing radioactive cloud and the material deposited on the ground. Radiation doses received from the cloud result from direct radiation (cloudshine) and inhalation of material suspended in the air (inhalation). These processes occur only during the time that the cloud passes over the affected population. Radiation doses associated with the material deposited to the ground include direct radiation of the nuclides on the ground (groundshine) and inhalation of materials that are resuspended into the air (resuspension). The MACCS2 code simulates these dose paths. Therefore, the code estimates the dose levels for each nuclide and for each dispersion analysis performed.

Six release categories are identified for evaluation of potential offsite doses. These categories are discussed in detail in Chapter 45, and are summarized as the following:

- IC – Containment integrity is maintained throughout the accident, and the release of radiation to the environment is due to nominal leakage.

- BP – Fission products are released from the reactor coolant system to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to onset of core damage.

- CI – Fission product release occurs through a failure of the system or valves that close the penetrations between containment and the environment. Containment failure occurs prior to onset of core damage.

- CFE – Fission product release occurs through a containment failure caused by some dynamic severe accident phenomena occurring after the onset of core damage but prior to core relocation. Such phenomena include: hydrogen detonation, hydrogen diffusion flame, steam explosions, and vessel failures.

- CFI – Fission product release occurs through a containment failure caused by some dynamic severe accident phenomena occurring after core relocation but before 24 hours. Such phenomena include: hydrogen detonation and hydrogen deflagration.

- CFL – Fission product release occurs through a containment failure caused by some dynamic severe accident phenomena occurring after 24 hours. Such phenomena include the failure of containment heat removal (failure of passive containment cooling).

Additionally, one sensitivity evaluation (called DIRECT) is performed. The DIRECT release case is a modification of the IC release category in which no credit is assumed for aerosol nuclide deposition in the middle annulus. This case is conservative.

Based on the analysis described in Chapter 45, source terms are generated for each of the release categories. The source terms provide the necessary parameters to describe the conditions of the release. Tables 49-1 and 49-2 contain listings of the source terms and their parameters considered in this analysis. Table 49-1 summarizes the output provided by the Modular Accident Analysis Program (MAAP) code, while Table 49-2 presents the conversion of this data into MACCS2 input categories.

There are nine source terms, six release categories, and one sensitivity study defined for this analysis. To conservatively estimate the ground-level dose exposure at the site boundary, it is assumed that all the release categories occur at ground level. Finally, it is conservatively assumed that 5 percent of the iodine released from containment is volatile and would not deposit. Reference 49-1 provides a guideline of 3 percent volatile iodine.

Reference 49-1 provides some of the MACCS2 input data, including the site, and meteorological data. The dose data conversion file provided with the MACCS2 PC Code, version 1.12 (Reference 49-2) is used for this analysis. This file is required to convert the predicted nuclide concentrations to dose values.

49.5 Dose Evaluation Results and Discussion

Doses are determined for the early exposure effects resulting from the initial 24 and 72 hours following the core damage initiation. The dose evaluation provides the conditional probability distributions for the consequence measures, which includes the whole-body dose and the acute red bone marrow dose for this analysis. These consequence probability distributions are based on the assumption that the accident that produced the source term has occurred. Therefore, the consequence probability distributions presented result from the variation in dose levels due to the various meteorological conditions. Hence, the actual

probability of the identified dose levels would be the probability of the release category that produced the source term occurring multiplied by the probability of the dose level. The actual probability of the identified dose levels is presented in Section 49.7.

Tables 49-3 through 49-6 present the summary of the dose evaluations (the MACCS2 output) for the six source terms and the DIRECT sensitivity. The information is provided in the following columns: the mean dose; the 50, 90, 95, 99, and 99.5 percent confidence values that the dose will not exceed; and the peak dose produced by any dispersion analysis. The dose (one sievert equals 100 rem) is presented for the following source terms: IC, BP, CFI, CFE, CFL, CI and DIRECT release sensitivity. Table 49-7 summarizes the calculated mean and peak dose values for the source terms evaluated.

Figures 49-1 through 49-56 present plots of the complementary cumulative distribution functions for the population whole-body, the site boundary whole-body effective dose equivalent, the acute red bone marrow and thyroid doses resulting from the following source terms: CFI, CFE, CFL, IC, BP, CI, and the DIRECT release sensitivity study.

Results in Table 49-7 show that for release categories CFL, IC and the DIRECT sensitivity study, the mean whole-body EDE dose at the site boundary in 24 hours is less than 6 rem. For all other release categories – BP, CI, CFE, and CFI~~, and CFL~~ – the mean dose at the site boundary in 24 hours is greater than 25 rem. The sum of the probabilities of the release categories including an intact containment excess leakage category is approximately 2.4×10^{-7} events per year for at power conditions. Therefore, for the CFL, IC and the DIRECT release categories, there is a large margin in both the dose as well as the probability for meeting the Westinghouse design goal of limiting the frequency of exceeding the 25 rem whole-body effective dose equivalent for an individual at the site boundary 24 hours after core damage to 1×10^{-6} events per year, without any emergency protective action. For the other release categories – BP, CI, CFE, and CFI~~, and CFL~~ – there is a large margin in the probability for meeting the Westinghouse design goal.

Results in Table 49-7 also show that for release categories CFL, IC and the DIRECT sensitivity study, the acute red bone marrow dose at the site boundary in 24 hours is less than 1 rem. For all other release categories – BP, CI, CFE, and CFI~~, and CFL~~ – the mean dose at the site boundary in 24 hours is greater than 25 rem. Again, the sum of the probabilities of the release categories including an intact containment excess leakage category is approximately 2.4×10^{-7} events per year for at power conditions. Therefore, for the CFL, IC and the DIRECT release categories, there is a large margin in both the dose as well as the probability for meeting the Westinghouse design goal of limiting the frequency of exceeding the 25 rem whole-body effective dose equivalent for an individual at the site boundary 24 hours after core damage to 1×10^{-6} events per year, without any emergency protective action. For the other release categories – BP, CI, CFE, and CFI~~, and CFL~~ – there is a large margin in the probability for meeting the Westinghouse design goal.

49.6 Quantification of Site Risk

This section documents the calculation of total radiation dose risk at the site boundary and to the surrounding population for internal, at power, initiating events. Results are quantified based on both a 24-hour and a 72-hour exposure.

The dose risks are quantified by multiplying the calculated fission product release category frequency vector by the release category mean dose vectors. The frequencies for each of the six release categories are quantified in Chapter 45, while the mean doses for each release category are identified in this section. The total dose risk for each case is calculated as:

$$D_n = \Sigma_i (R_i \times d_{i,n})$$

where:

- D_n = Total dose risk for case n=1,2,3,4 (site-24-hr, site-72-hr, population-24-hr, population-72-hr),
- R_i = Release frequency for category i,
- $d_{i,n}$ = mean dose for release category i for case n.

As previously described, the six release categories analyzed in this calculation are designated: IC, BP, CI, CFI, CFE, and CFL.

Tables 49-8 through 49-11 present the results of the dose risk calculations. Each table presents the release category identifier, the release frequency (per reactor-year), the mean dose (in rem), and the resulting risk (in rem per reactor-year). In addition, each table presents the total dose risk and the percent that each release category contributes to the total risk.

It is shown that release category CFE presents the largest risk to the site safety in each of the four presented cases.

49.7 Risk Quantification Results

The complementary cumulative distribution function (CCDF) probabilities presented in Figures 49-1 through 49-56 are based on the assumption that the respective release category has occurred. The actual dose probability is equal to the probability of the release multiplied by the CCDF probability. Figure 49-57 summarizes this calculation for the 24 hour, whole-body, site boundary dose for all release categories, excluding the sensitivity study. Figure 49-58 summarizes this calculation for the 24 hour, acute red bone marrow, site boundary dose for all release categories, excluding the sensitivity study. In addition, a total probability-dose curve, which sums all the release categories, is provided. This figure demonstrates compliance with the large release goal (24 hour, whole-body, site boundary dose greater than 25 REM has a frequency of less than 1×10^{-6}).

49.8 References

49-1 "Advanced Light Water Reactor Utility Requirements Document," Volume III, Appendix A to Chapter 1, PRA Key Assumptions and Groundrules," EPRI, Rev. 5 & 6, December 1993.

49-2 Chanin, D., Young, M. L., "Code Manual for MACCS2, User's Guide," NUREG/CR-6613, SAND97-0594, Vol. 1, Sandia National Laboratories, U.S. Nuclear Regulatory Commission.

49-3 "PRA Procedures Guide," NUREG/CR-2300, U.S. Nuclear Regulatory Commission, Vol. 2, Chapter 9, Washington, D.C.

Table 49-1

AP1000 SOURCE TERMS FROM LEVEL 2 ANALYSIS (MAAP)

					Release Fraction (MAAP Group)												Plume Energy (Joules/sec)	
Case No.	Plume No.	Start Time (Seconds)	End Time (Seconds)	Duration (Seconds)	1 Inert	2 CsI	3 TeO2	4 SrO	5 MoO2	6 CsOH	7 BaO	8 La2O3	9 CeO2	10 Sb	11 Te2	12 UO2	(Watts)	Plume Position
CFI	1	2924 ~~2930~~	32590 ~~10100~~	29666 ~~7170~~	5.40E-01 ~~2.88E-01~~	3.19E-03 ~~5.09E-03~~	4.83E-03 ~~7.92E-03~~	2.11E-02 ~~4.77E-03~~	9.11E-03 ~~1.03E-02~~	3.18E-03 ~~5.07E-03~~	1.62E-02 ~~8.12E-03~~	3.53E-03 ~~1.56E-04~~	6.61E-03 ~~1.82E-04~~	6.92E-03 ~~8.89E-03~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Leading ~~Leading~~
	2	32590 ~~10100~~	86420 ~~89850~~	53830 ~~79750~~	2.58E-01 ~~4.69E-01~~	1.35E-04 ~~3.50E-03~~	1.45E-04 ~~4.99E-03~~	6.50E-04 ~~9.23E-03~~	1.68E-04 ~~6.62E-03~~	1.35E-04 ~~3.49E-03~~	3.40E-04 ~~9.18E-03~~	4.53E-03 ~~1.08E-03~~	4.19E-03 ~~1.09E-03~~	2.77E-04 ~~5.96E-03~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	3	86420 ~~89850~~	172800 ~~176200~~	86380 ~~86350~~	8.40E-02 ~~6.98E-02~~	0.00E+00 ~~6.00E-06~~	0.00E+00 ~~1.00E-05~~	0.00E+00 ~~4.20E-04~~	0.00E+00 ~~1.00E-05~~	0.00E+00 ~~6.00E-06~~	0.00E+00 ~~1.70E-04~~	6.00E-03 ~~5.84E-04~~	5.44E-03 ~~5.84E-04~~	7.40E-05 ~~4.00E-05~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	4	172800 ~~176200~~	259200 ~~262600~~	86400 ~~86400~~	3.83E-02 ~~2.35E-02~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~1.00E-05~~	0.00E+00 ~~1.50E-04~~	0.00E+00 ~~4.00E-05~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~8.00E-05~~	5.22E-03 ~~3.22E-04~~	4.74E-03 ~~3.21E-04~~	2.60E-05 ~~4.00E-05~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
CFE	1	3004 ~~3004~~	19810 ~~19810~~	16806 ~~16806~~	4.16E-01 ~~4.16E-01~~	5.53E-02 ~~5.53E-02~~	3.01E-02 ~~3.01E-02~~	3.14E-03 ~~3.14E-03~~	1.16E-02 ~~1.16E-02~~	5.35E-02 ~~5.35E-02~~	4.63E-03 ~~4.63E-03~~	5.57E-05 ~~5.57E-05~~	2.39E-04 ~~2.39E-04~~	2.03E-02 ~~2.03E-02~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Leading ~~Leading~~
	2	19810 ~~19810~~	89970 ~~89970~~	70160 ~~70160~~	4.05E-01 ~~4.05E-01~~	1.26E-03 ~~1.26E-03~~	1.48E-03 ~~1.48E-03~~	3.43E-04 ~~3.43E-04~~	2.58E-03 ~~2.58E-03~~	1.20E-03 ~~1.20E-03~~	6.45E-04 ~~6.45E-04~~	9.66E-06 ~~9.66E-06~~	1.14E-05 ~~1.14E-05~~	2.66E-03 ~~2.66E-03~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Leading ~~Leading~~
	3	89970 ~~89970~~	176300 ~~176300~~	86330 ~~86330~~	1.08E-01 ~~1.08E-01~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	4	176300 ~~176300~~	262700 ~~262700~~	86400 ~~86400~~	3.43E-02 ~~3.43E-02~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	1.00E-05 ~~1.00E-05~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
DIRECT	1	4378 ~~4378~~	84810 ~~84810~~	80432 ~~80432~~	2.95E-03 ~~2.95E-03~~	3.61E-05 ~~3.61E-05~~	2.86E-05 ~~2.86E-05~~	3.22E-05 ~~3.22E-05~~	3.94E-05 ~~3.94E-05~~	3.44E-05 ~~3.44E-05~~	3.61E-05 ~~3.61E-05~~	4.04E-06 ~~4.04E-06~~	4.39E-06 ~~4.39E-06~~	3.99E-05 ~~3.99E-05~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	2	84810 ~~84810~~	134400 ~~134400~~	49590 ~~49590~~	1.48E-03 ~~1.48E-03~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	1.80E-08 ~~1.80E-08~~	2.40E-08 ~~2.40E-08~~	2.40E-07 ~~2.40E-07~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Leading ~~Leading~~
	3	134400 ~~134400~~	177600 ~~177600~~	43200 ~~43200~~	1.18E-03 ~~1.18E-03~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	6.00E-08 ~~6.00E-08~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	4	177600 ~~177600~~	264000 ~~264000~~	86400 ~~86400~~	2.32E-03 ~~2.32E-03~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	3.00E-08 ~~3.00E-08~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
IC	1	4378 ~~4378~~	84810 ~~84810~~	80432 ~~80432~~	9.83E-04 ~~9.83E-04~~	1.20E-05 ~~1.20E-05~~	9.53E-06 ~~9.53E-06~~	1.07E-05 ~~1.07E-05~~	1.31E-05 ~~1.31E-05~~	1.15E-05 ~~1.15E-05~~	1.20E-05 ~~1.20E-05~~	1.35E-06 ~~1.35E-06~~	1.46E-06 ~~1.46E-06~~	1.33E-05 ~~1.33E-05~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	2	84810 ~~84810~~	134400 ~~134400~~	49590 ~~49590~~	4.93E-04 ~~4.93E-04~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	6.00E-09 ~~6.00E-09~~	8.00E-09 ~~8.00E-09~~	8.00E-08 ~~8.00E-08~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Leading ~~Leading~~
	3	134400 ~~134400~~	177600 ~~177600~~	43200 ~~43200~~	3.94E-04 ~~3.94E-~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	2.00E-08 ~~2.00E-~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~

					~~04~~									~~08~~				
	4	177600 ~~177600~~	264000 ~~264000~~	86400 ~~86400~~	7.72E-04 ~~7.72E-04~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	1.00E-08 ~~1.00E-08~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
BP	1	31890 ~~5984~~	46440 ~~22130~~	14550 ~~16146~~	1.00E+00 ~~6.22E-01~~	1.69E-01 ~~1.20E-02~~	2.46E-01 ~~1.85E-02~~	3.57E-03 ~~2.02E-04~~	4.48E-02 ~~4.97E-03~~	1.61E-01 ~~1.10E-02~~	8.93E-03 ~~7.91E-04~~	1.30E-04 ~~1.25E-05~~	7.99E-04 ~~3.52E-05~~	1.04E-01 ~~5.48E-02~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	2	46440 ~~22130~~	86490 ~~108500~~	40050 ~~86370~~	0.00E+00 ~~3.00E-04~~	4.64E-02 ~~5.00E-05~~	6.70E-03 ~~2.00E-05~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	3.21E-02 ~~0.00E+00~~	2.00E-06 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	5.17E-02 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	LeadingMidpoint ~~LeadingMidpoint~~
	3	86490 ~~108500~~	172800 ~~194900~~	86310 ~~86400~~	0.00E+00 ~~4.00E-04~~	2.31E-01 ~~0.00E+00~~	3.40E-03 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	4.41E-02 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	8.80E-02 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	LeadingMidpoint ~~LeadingMidpoint~~
	4	172800 ~~194900~~	259200 ~~267800~~	86400 ~~72900~~	0.00E+00 ~~3.00E-04~~	2.80E-03 ~~0.00E+00~~	4.00E-04 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	1.09E-02 ~~0.00E+00~~	1.00E-06 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	2.59E-02 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
CI	1	101 ~~101~~	50020 ~~50020~~	49919 ~~49919~~	5.73E-01 ~~5.73E-01~~	4.56E-02 ~~4.56E-02~~	2.12E-02 ~~2.12E-02~~	2.03E-02 ~~2.03E-02~~	4.04E-02 ~~4.04E-02~~	1.78E-02 ~~1.78E-02~~	3.16E-02 ~~3.16E-02~~	2.39E-04 ~~2.39E-04~~	7.42E-04 ~~7.42E-04~~	2.71E-02 ~~2.71E-02~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	2	50020 ~~50020~~	136400 ~~136400~~	86380 ~~86380~~	1.13E-01 ~~1.13E-01~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	1.00E-07 ~~1.00E-07~~	0.00E+00 ~~0.00E+00~~	1.90E-04 ~~1.90E-04~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	3	136400 ~~136400~~	211700 ~~211700~~	75300 ~~75300~~	5.66E-02 ~~5.66E-02~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	1.34E-03 ~~1.34E-03~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	4	211700 ~~211700~~	259600 ~~259600~~	47900 ~~47900~~	2.74E-02 ~~2.74E-02~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	2.10E-04 ~~2.10E-04~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
CFL	1	2922 ~~4353~~	26360 ~~23610~~	23438 ~~19257~~	3.36E-04 ~~7.51E-01~~	1.20E-05 ~~3.57E-02~~	8.51E-06 ~~4.36E-02~~	1.57E-05 ~~2.01E-03~~	1.68E-05 ~~9.90E-03~~	1.15E-05 ~~2.74E-02~~	1.61E-05 ~~3.02E-03~~	9.96E-07 ~~3.31E-05~~	1.85E-06 ~~1.60E-04~~	1.66E-05 ~~2.87E-02~~	0.00E+00 ~~1.14E-070.00E+00~~	0.00E+00 ~~7.76E-100.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	2	26360 ~~23610~~	108000 ~~91280~~	81640 ~~67670~~	1.19E-03 ~~1.22E-01~~	5.00E-08 ~~9.20E-04~~	2.60E-08 ~~3.00E-05~~	1.04E-06 ~~0.00E+00~~	2.90E-07 ~~0.00E+00~~	3.00E-08 ~~2.70E-04~~	6.60E-07 ~~0.00E+00~~	1.07E-05 ~~0.00E+00~~	1.01E-05 ~~0.00E+00~~	2.90E-07 ~~2.90E-04~~	0.00E+00 ~~5.70E-090.00E+00~~	0.00E+00 ~~2.00E-130.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	3	108000 ~~91280~~	194400 ~~177600~~	86400 ~~86320~~	9.79E-01 ~~6.93E-02~~	2.13E-05 ~~2.40E-04~~	4.20E-08 ~~3.00E-05~~	2.39E-03 ~~0.00E+00~~	1.26E-03 ~~0.00E+00~~	1.03E-05 ~~1.00E-04~~	2.25E-03 ~~0.00E+00~~	9.75E-02 ~~0.00E+00~~	9.21E-02 ~~0.00E+00~~	4.09E-04 ~~2.00E-04~~	0.00E+00 ~~1.00E-090.00E+00~~	0.00E+00 ~~0.00E+000.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	4	194400 ~~177600~~	259200 ~~264000~~	64800 ~~86400~~	0.00E+00 ~~2.84E-02~~	0.00E+00 ~~1.00E-05~~	0.00E+00 ~~0.00E+00~~	4.42E-04 ~~0.00E+00~~	1.55E-04 ~~0.00E+00~~	2.90E-07 ~~1.00E-05~~	3.46E-04 ~~0.00E+00~~	4.39E-02 ~~0.00E+00~~	4.15E-02 ~~0.00E+00~~	1.98E-04 ~~1.20E-04~~	0.00E+00 ~~3.00E-100.00E+00~~	0.00E+00 ~~0.00E+000.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~

Table 49-2

AP1000 SOURCE TERMS FOR DOSE EVALUATION (MACCS2)

Case No.	Plume No.	Start Time (Seconds)	End Time (Seconds)	Duration (Seconds)	Release Fraction (MACCS2 Group) 1 Inert	2 I	3 Cs	4 Te/Sb	5 Sr	6 Ru	7 La	8 Ce	9 Ba	Plume Energy Joules/sec (Watts)	Plume Position
CFI	1	2924 ~~2930~~	32590 ~~10100~~	29666 ~~7170~~	5.40E-01 ~~2.88E-01~~	3.19E-03 ~~5.09E-03~~	3.18E-03 ~~5.08E-03~~	4.18E-04 ~~5.14E-04~~	2.11E-02 ~~4.77E-03~~	9.11E-03 ~~1.03E-02~~	3.53E-03 ~~1.56E-04~~	2.64E-05 ~~6.04E-07~~	1.62E-02 ~~8.12E-03~~	0.00E+00 ~~0.00E+00~~	Leading ~~Leading~~
	2	32590 ~~10100~~	86420 ~~89850~~	53830 ~~79750~~	2.58E-01 ~~4.69E-01~~	1.35E-04 ~~3.50E-03~~	1.35E-04 ~~3.49E-03~~	1.67E-05 ~~3.45E-04~~	6.50E-04 ~~9.23E-03~~	1.68E-04 ~~6.62E-03~~	4.53E-03 ~~1.08E-03~~	1.68E-05 ~~3.61E-06~~	3.40E-04 ~~9.15E-03~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	3	86420 ~~89850~~	172800 ~~176200~~	86380 ~~86350~~	8.40E-02 ~~6.98E-02~~	0.00E+00 ~~6.00E-06~~	0.00E+00 ~~6.00E-06~~	4.47E-06 ~~2.31E-06~~	0.00E+00 ~~4.20E-04~~	0.00E+00 ~~1.00E-05~~	6.00E-03 ~~5.84E-04~~	2.17E-05 ~~1.94E-06~~	0.00E+00 ~~1.70E-04~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	4	172800 ~~176200~~	259200 ~~262600~~	86400 ~~86400~~	3.83E-02 ~~2.35E-02~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	1.57E-06 ~~2.31E-06~~	0.00E+00 ~~1.50E-04~~	0.00E+00 ~~4.00E-05~~	5.22E-03 ~~3.23E-04~~	1.89E-05 ~~1.06E-06~~	0.00E+00 ~~8.00E-05~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
CFE	1	3004 ~~3004~~	19810 ~~19810~~	16806 ~~16806~~	4.16E-01 ~~4.16E-01~~	5.53E-02 ~~5.53E-02~~	5.37E-02 ~~5.37E-02~~	1.23E-03 ~~1.17E-03~~	3.14E-03 ~~2.14E-03~~	1.16E-02 ~~1.16E-02~~	5.57E-05 ~~5.57E-05~~	9.54E-07 ~~7.91E-07~~	4.63E-03 ~~4.63E-03~~	0.00E+00 ~~0.00E+00~~	Leading ~~Leading~~
	2	19810 ~~19810~~	89970 ~~89970~~	70160 ~~70160~~	4.05E-01 ~~4.05E-01~~	1.26E-03 ~~1.26E-03~~	1.21E-03 ~~1.21E-03~~	1.61E-04 ~~1.54E-04~~	3.43E-04 ~~3.43E-04~~	2.58E-03 ~~2.58E-03~~	9.66E-06 ~~9.66E-06~~	4.56E-08 ~~3.78E-08~~	6.45E-04 ~~6.45E-04~~	0.00E+00 ~~0.00E+00~~	Leading ~~Leading~~
	3	89970 ~~89970~~	176300 ~~176300~~	86330 ~~86330~~	1.08E-01 ~~1.08E-01~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	4	176300 ~~176300~~	262700 ~~262700~~	86400 ~~86400~~	3.43E-02 ~~3.43E-02~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	6.04E-07 ~~5.78E-07~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
DIRECT	1	4378 ~~4378~~	84810 ~~84810~~	80432 ~~80432~~	2.95E-03 ~~2.95E-03~~	3.61E-05 ~~3.61E-05~~	3.46E-05 ~~3.46E-05~~	2.41E-06 ~~2.31E-06~~	3.22E-05 ~~3.22E-05~~	3.94E-05 ~~3.94E-05~~	4.04E-06 ~~4.04E-06~~	1.75E-08 ~~1.45E-08~~	3.61E-05 ~~3.61E-05~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	2	84810 ~~84810~~	134400 ~~134400~~	49590 ~~49590~~	1.48E-03 ~~1.48E-03~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	1.45E-08 ~~1.39E-08~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	1.80E-08 ~~1.80E-08~~	9.59E-11 ~~7.05E-11~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Leading ~~Leading~~
	3	134400 ~~134400~~	177600 ~~177600~~	43200 ~~43200~~	1.18E-03 ~~1.18E-03~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	3.63E-09 ~~3.47E-09~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	4	177600 ~~177600~~	264000 ~~264000~~	86400 ~~86400~~	2.32E-03 ~~2.32E-03~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	1.81E-09 ~~1.73E-09~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
IC	1	4378 ~~4378~~	84810 ~~84810~~	80432 ~~80432~~	9.83E-04 ~~9.83E-04~~	1.20E-05 ~~1.20E-05~~	1.15E-05 ~~1.15E-05~~	8.04E-07 ~~7.69E-07~~	1.07E-05 ~~1.07E-05~~	1.31E-05 ~~1.31E-05~~	1.35E-06 ~~1.35E-06~~	5.85E-09 ~~4.85E-09~~	1.20E-05 ~~1.20E-05~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	2	84810 ~~84810~~	134400 ~~134400~~	49590 ~~49590~~	4.93E-04 ~~4.93E-04~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	4.83E-09 ~~4.62E-09~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	6.00E-09 ~~6.00E-09~~	3.20E-11 ~~2.65E-11~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Leading ~~Leading~~
	3	134400 ~~134400~~	177600 ~~177600~~	43200 ~~43200~~	3.94E-04 ~~3.94E-04~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	1.21E-09 ~~1.16E-09~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~
	4	177600 ~~177600~~	264000 ~~264000~~	86400 ~~86400~~	7.72E-04 ~~7.72E-~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	6.04E-10 ~~5.78E-10~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	0.00E+00 ~~0.00E+00~~	Midpoint ~~Midpoint~~

					~~04~~									
BP	1	31890~~5984~~	46440~~22130~~	14550~~16146~~	1.00E+00~~6.22E-01~~	1.69E-01~~1.20E-02~~	1.62E-01~~1.11E-02~~	6.27E-03~~3.17E-03~~	3.57E-03~~3.02E-04~~	4.48E-02~~4.97E-03~~	1.30E-04~~1.25E-05~~	3.19E-06~~1.17E-07~~	8.93E-03~~7.91E-04~~	0.00E+00~~0.00E+00~~	Midpoint~~Midpoint~~
	2	46440~~22130~~	86490~~108500~~	40050~~86370~~	0.00E+00~~3.00E-04~~	4.64E-02~~5.00E-05~~	3.38E-02~~5.73E-06~~	3.12E-03~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	2.00E-06~~0.00E+00~~	0.00E+00~~0.00E+00~~	Leading~~Midpoint~~
	3	86490~~108500~~	172800~~194900~~	86310~~86400~~	0.00E+00~~4.00E-04~~	2.31E-01~~0.00E+00~~	6.60E-02~~0.00E+00~~	5.32E-03~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	Leading~~Midpoint~~
	4	172800~~194900~~	259200~~267800~~	86400~~72900~~	0.00E+00~~7.00E-04~~	2.80E-03~~0.00E+00~~	9.96E-03~~0.00E+00~~	1.57E-03~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	1.00E-06~~0.00E+00~~	0.00E+00~~0.00E+00~~	Midpoint~~Midpoint~~
CI	1	100.8~~101~~	50020~~50020~~	49919.2~~49910~~	5.73E-01~~5.73E-01~~	4.56E-02~~4.56E-02~~	2.10E-02~~2.10E-02~~	1.64E-03~~1.57E-03~~	2.03E-02~~2.03E-02~~	4.04E-02~~4.04E-02~~	2.39E-04~~2.39E-04~~	2.97E-06~~2.46E-06~~	3.16E-02~~3.16E-02~~	0.00E+00~~0.00E+00~~	Midpoint~~Midpoint~~
	2	50020~~50020~~	136400~~136400~~	86380~~86380~~	1.13E-01~~1.13E-01~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	1.15E-05~~1.10E-05~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	1.00E-07~~1.00E-07~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	Midpoint~~Midpoint~~
	3	136400~~126400~~	211700~~211700~~	75300~~75300~~	5.66E-02~~5.66E-02~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	8.10E-05~~7.74E-05~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	Midpoint~~Midpoint~~
	4	211700~~211700~~	259600~~259600~~	47900~~47900~~	2.74E-02~~2.74E-02~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	1.27E-05~~1.21E-05~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	0.00E+00~~0.00E+00~~	Midpoint~~Midpoint~~
CFL	1	2922~~4353~~	26360~~23610~~	23438~~19257~~	3.36E-04~~7.51E-01~~	1.20E-05~~3.57E-02~~	1.15E-05~~2.84E-02~~	1.00E-06~~1.66E-03~~	1.57E-05~~2.01E-03~~	1.68E-05~~9.90E-03~~	9.96E-07~~3.31E-05~~	7.41E-09~~5.30E-07~~	1.61E-05~~3.02E-03~~	0.00E+00~~0.00E+00~~	Midpoint~~Midpoint~~
	2	26360~~23610~~	108000~~91280~~	81640~~67670~~	1.19E-03~~1.22E-01~~	5.00E-08~~9.30E-04~~	3.23E-08~~3.46E-04~~	1.75E-08~~1.68E-05~~	1.04E-06~~0.00E+00~~	2.90E-07~~0.00E+00~~	1.07E-05~~0.00E+00~~	4.05E-08~~1.99E-13~~	6.60E-07~~0.00E+00~~	0.00E+00~~0.00E+00~~	Midpoint~~Midpoint~~
	3	108000~~91280~~	194400~~177600~~	86400~~86720~~	9.79E-01~~6.93E-02~~	2.13E-05~~2.40E-04~~	1.16E-05~~1.16E-04~~	2.47E-05~~1.16E-05~~	2.39E-03~~0.00E+00~~	1.26E-03~~0.00E+00~~	9.75E-02~~0.00E+00~~	3.68E-04~~0.00E+00~~	2.25E-03~~0.00E+00~~	0.00E+00~~0.00E+00~~	Midpoint~~Midpoint~~
	4	194400~~177600~~	259200~~264000~~	64800~~86400~~	0.00E+00~~2.84E-02~~	0.00E+00~~1.00E-05~~	2.56E-07~~1.00E-05~~	1.20E-05~~6.93E-06~~	4.42E-04~~0.00E+00~~	1.55E-04~~0.00E+00~~	4.39E-02~~0.00E+00~~	1.66E-04~~0.00E+00~~	3.46E-04~~0.00E+00~~	0.00E+00~~0.00E+00~~	Midpoint~~Midpoint~~

Table 49-3

SITE BOUNDARY WHOLE BODY DOSE [EFFECTIVE DOSE EQUIVALENT (EDE)], SIEVERTS

24-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	2.59E+01~~3.25E+01~~	1.71E+01~~2.61E+01~~	7.33E+01~~7.11E+01~~	8.07E+01~~8.48E+01~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	8.48E+01~~9.29E+01~~
CFE	4.23E+01~~4.23E+01~~	2.22E+01~~2.22E+01~~	1.03E+02~~1.03E+02~~	1.35E+02~~1.35E+02~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	1.55E+02~~1.55E+02~~
DIRECT	5.48E-02~~5.48E-02~~	3.37E-02~~3.37E-02~~	1.36E-01~~1.36E-01~~	1.77E-01~~1.77E-01~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	2.37E-01~~2.37E-01~~
IC	1.82E-02~~1.82E-02~~	1.08E-02~~1.08E-02~~	4.84E-02~~4.84E-02~~	6.24E-02~~6.24E-02~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	7.89E-02~~7.88E-02~~
BP	1.37E+02~~1.15E+01~~	9.21E+01~~7.72E+00~~	3.17E+02~~3.15E+01~~	4.12E+02~~3.92E+01~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	4.72E+02~~4.39E+01~~
CI	5.10E+01~~5.10E+01~~	3.53E+01~~3.53E+01~~	1.24E+02~~1.24E+02~~	1.55E+02~~1.55E+02~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	2.27E+02~~2.27E+02~~
CFL	3.84E-02~~2.58E+01~~	3.08E-02~~2.10E+01~~	8.70E-02~~5.39E+01~~	1.01E-01~~6.47E+01~~	1.06E-01~~7.52E+01~~	1.08E-01~~7.81E+01~~	1.17E-01~~9.23E+01~~
72-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	

CFI	3.72E+01~~3.49E+01~~	3.16E+01~~3.01E+01~~	7.65E+01~~7.12E+01~~	9.13E+01~~8.74E+01~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	1.01E+02~~9.71E+01~~
CFE	4.60E+01~~4.60E+01~~	2.31E+01~~2.31E+01~~	1.16E+02~~1.16E+02~~	1.47E+02~~1.47E+02~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	1.67E+02~~1.67E+02~~
DIRECT	5.84E-02~~6.09E-02~~	3.54E-02~~3.67E-02~~	1.63E-01~~1.66E-01~~	2.03E-01~~2.27E-01~~	2.15E-01~~NOT-FOUND~~	2.21E-01~~NOT-FOUND~~	2.50E-01~~2.56E-01~~
IC	1.94E-02~~2.21E-02~~	1.11E-02~~1.31E-02~~	5.07E-02~~5.73E-02~~	6.31E-02~~7.96E-02~~	7.29E-02~~NOT-FOUND~~	7.45E-02~~NOT-FOUND~~	8.30E-02~~8.97E-02~~
BP	1.84E+02~~1.23E+01~~	1.39E+02~~8.02E+00~~	3.68E+02~~3.18E+01~~	4.80E+02~~4.09E+01~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	5.31E+02~~4.66E+01~~
CI	5.40E+01~~5.40E+01~~	3.75E+01~~3.75E+01~~	1.39E+02~~1.39E+02~~	1.89E+02~~1.89E+02~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	2.38E+02~~2.38E+02~~
CFL	1.33E+02~~2.80E+01~~	8.42E+01~~2.13E+01~~	3.32E+02~~6.32E+01~~	4.02E+02~~7.12E+01~~	NOT-FOUND~~7.86E+01~~	NOT-FOUND~~8.20E+01~~	4.42E+02~~9.92E+01~~

Table 49-4

SITE BOUNDARY THYROID DOSE, SIEVERTS

24-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	2.22E+01~~5.67E+01~~	1.24E+01~~5.09E+01~~	7.03E+01~~1.14E+02~~	7.50E+01~~1.44E+02~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	7.75E+01~~1.62E+02~~
CFE	3.59E+02~~3.59E+02~~	2.02E+02~~2.02E+02~~	1.01E+03~~1.01E+03~~	1.18E+03~~1.18E+03~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	1.28E+03~~1.28E+03~~
DIRECT	1.34E-01~~1.34E-01~~	8.58E-02~~8.58E-02~~	3.55E-01~~3.55E-01~~	4.48E-01~~4.48E-01~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	5.23E-01~~5.23E-01~~
IC	4.47E-02~~4.47E-02~~	3.08E-02~~3.08E-02~~	1.20E-01~~1.20E-01~~	1.54E-01~~1.54E-01~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	1.74E-01~~1.74E-01~~
BP	1.56E+03~~7.71E+01~~	1.18E+03~~5.40E+01~~	3.14E+03~~2.04E+02~~	3.89E+03~~2.23E+02~~	NOT-FOUND~~2.73E+02~~	NOT-FOUND~~NOT-FOUND~~	4.35E+03~~2.73E+02~~
CI	1.82E+02~~1.82E+02~~	1.23E+02~~1.23E+02~~	4.58E+02~~4.58E+02~~	5.74E+02~~5.74E+02~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	7.30E+02~~7.30E+02~~
CFL	7.28E-02~~2.20E+02~~	5.22E-02~~1.68E+02~~	1.47E-01~~5.34E+02~~	1.81E-01~~6.39E+02~~	2.09E-01~~7.07E+02~~	2.14E-01~~7.11E+02~~	2.38E-01~~7.30E+02~~
72-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	

CFI	2.47E+01~~6.14E+01~~	1.43E+01~~5.26E+01~~	7.07E+01~~1.23E+02~~	7.92E+01~~1.54E+02~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	8.39E+01~~1.73E+02~~
CFE	3.81E+02~~3.81E+02~~	2.11E+02~~2.11E+02~~	1.02E+03~~1.02E+03~~	1.23E+03~~1.23E+03~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	1.36E+03~~1.36E+03~~
DIRECT	1.50E-01~~1.56E-01~~	1.03E-01~~1.05E-01~~	4.10E-01~~4.20E-01~~	5.39E-01~~5.47E-01~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	5.79E-01~~5.94E-01~~
IC	4.97E-02~~5.65E-021.67E+00~~	3.38E-02~~3.86E-021.19E+00~~	1.25E-01~~1.33E-013.62E+00~~	1.67E-01~~1.76E-014.21E+00~~	NOT-FOUND~~NOT-FOUND5.30E+00~~	NOT-FOUND~~NOT-FOUND5.57E+00~~	1.92E-01~~2.09E-016.67E+00~~
BP	2.36E+03~~8.19E+01~~	2.22E+03~~5.64E+01~~	4.43E+03~~2.10E+02~~	5.09E+03~~2.60E+02~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	5.19E+03~~2.89E+02~~
CI	1.98E+02~~1.98E+02~~	1.40E+02~~1.40E+02~~	4.70E+02~~4.70E+02~~	5.79E+02~~5.79E+02~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	7.90E+02~~7.90E+02~~
CFL	7.72E+00~~2.35E+02~~	4.67E+00~~2.08E+02~~	1.97E+01~~5.34E+02~~	2.28E+01~~6.39E+02~~	NOT-FOUND~~7.18E+02~~	NOT-FOUND~~7.29E+02~~	2.43E+01~~7.76E+02~~

Table 49-5

POPULATION WHOLE BODY DOSE [EFFECTIVE DOSE EQUIVALENT (EDE)], 0-80.5 KM PERSON-SIEVERTS

24-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	7.03E+03~~7.88E+03~~	5.33E+03~~6.11E+03~~	1.31E+04~~1.47E+04~~	1.82E+04~~2.01E+04~~	3.11E+04~~3.21E+04~~	3.59E+04~~3.51E+04~~	5.07E+04~~5.34E+04~~
CFE	8.51E+03~~8.51E+03~~	6.25E+03~~6.25E+03~~	1.62E+04~~1.62E+04~~	2.31E+04~~2.31E+04~~	4.13E+04~~4.13E+04~~	5.06E+04~~5.06E+04~~	6.40E+04~~6.40E+04~~
DIRECT	2.16E+01~~2.16E+01~~	1.20E+01~~1.20E+01~~	4.78E+01~~4.78E+01~~	8.13E+01~~8.13E+01~~	1.14E+02~~1.14E+02~~	1.23E+02~~1.23E+02~~	1.68E+02~~1.68E+02~~
IC	7.19E+00~~7.19E+00~~	4.21E+00~~4.21E+00~~	1.71E+01~~1.71E+01~~	2.95E+01~~2.95E+01~~	3.56E+01~~3.56E+01~~	3.84E+01~~3.84E+01~~	5.60E+01~~5.60E+01~~
BP	3.23E+04~~2.91E+03~~	2.10E+04~~1.74E+03~~	6.40E+04~~5.90E+03~~	1.03E+05~~1.00E+04~~	1.54E+05~~1.52E+04~~	1.82E+05~~1.81E+04~~	2.64E+05~~2.58E+04~~
CI	2.01E+04~~2.01E+04~~	1.13E+04~~1.13E+04~~	4.71E+04~~4.71E+04~~	6.60E+04~~6.60E+04~~	1.23E+05~~1.23E+05~~	1.48E+05~~1.48E+05~~	1.61E+05~~1.61E+05~~
CFL	7.37E+01~~5.32E+03~~	1.00E+01~~3.87E+03~~	1.62E+02~~1.04E+04~~	5.91E+02~~1.35E+04~~	9.76E+02~~2.32E+04~~	1.11E+03~~2.77E+04~~	2.56E+03~~4.35E+04~~
72-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	1.13E+04~~8.89E+03~~	9.02E+03~~6.89E+03~~	2.12E+04~~1.63E+04~~	2.63E+04~~2.21E+04~~	4.09E+04~~3.42E+04~~	4.89E+04~~3.84E+04~~	6.18E+04~~5.73E+04~~
CFE	9.36E+03~~9.36E+03~~	6.89E+03~~6.89E+03~~	1.89E+04~~1.88E+04~~	2.54E+04~~2.54E+04~~	4.25E+04~~4.25E+04~~	5.12E+04~~5.12E+04~~	6.77E+04~~6.77E+04~~

DIRECT	2.36E+01~~2.45E+01~~	1.35E+01~~1.43E+01~~	5.28E+01~~5.50E+01~~	8.32E+01~~8.33E+01~~	1.15E+02~~1.16E+02~~	1.25E+02~~1.26E+02~~	1.75E+02~~1.78E+02~~
IC	7.87E+00~~8.80E+00~~	4.75E+00~~5.57E+00~~	1.85E+01~~1.98E+01~~	3.00E+01~~3.14E+01~~	3.79E+01~~4.41E+01~~	4.20E+01~~5.03E+01~~	5.83E+01~~6.33E+01~~
BP	4.17E+04~~3.11E+03~~	2.94E+04~~1.85E+03~~	7.99E+04~~6.31E+03~~	1.16E+05~~1.03E+04~~	2.20E+05~~1.54E+04~~	2.61E+05~~1.82E+04~~	2.87E+05~~2.69E+04~~
CI	2.14E+04~~2.14E+04~~	1.25E+04~~1.25E+04~~	4.90E+04~~4.90E+04~~	7.40E+04~~7.40E+04~~	1.27E+05~~1.27E+05~~	1.53E+05~~1.53E+05~~	1.67E+05~~1.67E+05~~
CFL	4.79E+04~~5.84E+03~~	3.11E+04~~4.32E+03~~	9.57E+04~~1.12E+04~~	1.57E+05~~1.48E+04~~	2.62E+05~~2.53E+04~~	3.01E+05~~3.04E+04~~	4.14E+05~~4.62E+04~~

Table 49-6

SITE BOUNDARY RED MARROW DOSE (TOTAL ACUTE), SIEVERTS

24-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	3.41E+00~~2.80E+00~~	2.14E+00~~1.89E+00~~	1.01E+01~~7.07E+00~~	1.11E+01~~7.93E+00~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	1.17E+01~~8.41E+00~~
CFE	7.81E+00~~7.81E+00~~	4.83E+00~~4.83E+00~~	2.01E+01~~2.01E+01~~	2.07E+01~~2.07E+01~~	2.22E+01~~2.22E+01~~	2.29E+01~~2.29E+01~~	2.71E+01~~2.70E+01~~
DIRECT	2.47E-03~~2.47E-03~~	1.49E-03~~1.49E-03~~	6.40E-03~~6.40E-03~~	8.69E-03~~8.69E-03~~	1.01E-02~~1.01E-02~~	1.01E-02~~1.01E-02~~	1.03E-02~~1.03E-02~~
IC	8.23E-04~~8.22E-04~~	5.34E-04~~5.34E-04~~	2.09E-03~~2.09E-03~~	2.67E-03~~2.67E-03~~	3.10E-03~~3.10E-03~~	3.15E-03~~3.15E-03~~	3.42E-03~~3.42E-03~~
BP	1.10E+01~~1.53E+00~~	8.47E+00~~1.09E+00~~	2.69E+01~~3.48E+00~~	3.27E+01~~4.77E+00~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	3.49E+01~~5.32E+00~~
CI	2.47E+00~~2.46E+00~~	1.67E+00~~1.67E+00~~	6.13E+00~~6.13E+00~~	7.72E+00~~7.72E+00~~	1.00E+01~~1.00E+01~~	NOT-FOUND~~NOT-FOUND~~	1.00E+01~~1.00E+01~~
CFL	2.20E-03~~3.01E+00~~	1.70E-03~~2.28E+00~~	4.91E-03~~6.38E+00~~	5.16E-03~~7.12E+00~~	5.58E-03~~7.86E+00~~	5.77E-03~~8.20E+00~~	6.70E-03~~9.90E+00~~
72-hour Case Source Term	**Quantiles**						**Peak Consequence**
	Mean	**50th**	**90th**	**95th**	**99th**	**99.5th**	
CFI	4.48E+00~~3.70E+00~~	3.05E+00~~3.14E+00~~	1.01E+01~~7.52E+00~~	1.09E+01~~9.09E+00~~	1.33E+01~~1.04E+01~~	1.44E+01~~1.07E+01~~	1.86E+01~~1.19E+01~~

CFE	1.02E+01~~1.02E+01~~	5.68E+00~~5.68E+00~~	3.01E+01~~3.01E+01~~	3.11E+01~~3.11E+01~~	3.36E+01~~3.36E+01~~	3.48E+01~~3.48E+01~~	4.17E+01~~4.17E+01~~
DIRECT	4.48E-03~~5.70E-03~~	3.08E-03~~4.48E-03~~	1.02E-02~~1.06E-02~~	1.09E-02~~1.13E-02~~	1.26E-02~~1.33E-02~~	1.34E-02~~1.43E-02~~	1.80E-02~~2.09E-02~~
IC	1.49E-03~~2.78E-03~~	1.03E-03~~2.44E-03~~	3.58E-03~~5.59E-03~~	4.49E-03~~7.00E-03~~	5.22E-03~~7.52E-03~~	5.34E-03~~7.76E-03~~	5.98E-03~~9.02E-03~~
BP	2.69E+01~~2.04E+00~~	2.44E+01~~1.28E+00~~	5.35E+01~~5.30E+00~~	5.99E+01~~6.75E+00~~	NOT-FOUND~~NOT-FOUND~~	NOT-FOUND~~NOT-FOUND~~	6.35E+01~~7.00E+00~~
CI	4.23E+00~~4.23E+00~~	3.13E+00~~3.13E+00~~	1.01E+01~~1.01E+01~~	1.08E+01~~1.08E+01~~	1.26E+01~~1.26E+01~~	1.34E+01~~1.34E+01~~	1.70E+01~~1.70E+01~~
CFL	5.11E+00~~4.34E+00~~	2.88E+00~~3.31E+00~~	1.19E+01~~9.35E+00~~	1.46E+01~~1.03E+01~~	NOT-FOUND~~1.13E+01~~	NOT-FOUND~~1.18E+01~~	1.61E+01~~1.42E+01~~

Table 49-7

DOSE SUMMARY

	Population Dose (Sieverts)				Site Boundary Whole Body Dose (Sieverts)				Site Boundary Thyroid Dose (Sieverts)				Site Boundary Red Marrow Dose (Sieverts)			
	24-Hour		72-Hour		24-Hour		72-Hour		24-Hour		72-Hour		24-Hour		72-Hour	
	Mean	Peak	Mean	Peak	Mean	Peak	Mean	Peak	Mean	Peak	Mean	Peak	Mean	Peak	Mean	Peak
CFI	7.03E+03 ~~7.88E+03~~	5.07E+04 ~~5.34E+04~~	1.13E+04 ~~8.89E+03~~	6.18E+04 ~~5.73E+04~~	2.59E+01 ~~3.25E+01~~	8.48E+01 ~~9.29E+01~~	3.72E+01 ~~3.49E+01~~	1.01E+02 ~~9.71E+01~~	2.22E+01 ~~5.67E+01~~	7.75E+01 ~~1.62E+02~~	2.47E+01 ~~6.14E+01~~	8.39E+01 ~~1.73E+02~~	3.41E+00 ~~2.80E+00~~	1.17E+01 ~~8.41E+00~~	4.48E+00 ~~3.70E+00~~	1.86E+01 ~~1.19E+01~~
CFE	8.51E+03 ~~8.51E+03~~	6.40E+04 ~~6.40E+04~~	9.36E+03 ~~9.36E+03~~	6.77E+04 ~~6.77E+04~~	4.23E+01 ~~4.23E+01~~	1.55E+02 ~~1.55E+02~~	4.60E+01 ~~4.60E+01~~	1.67E+02 ~~1.67E+02~~	3.59E+02 ~~3.59E+02~~	1.28E+03 ~~1.28E+03~~	3.81E+02 ~~3.81E+02~~	1.36E+03 ~~1.36E+03~~	7.81E+00 ~~7.81E+00~~	2.71E+01 ~~2.70E+01~~	1.02E+01 ~~1.02E+01~~	4.17E+01 ~~4.17E+01~~
DIRECT	2.16E+01 ~~2.16E+01~~	1.68E+02 ~~1.68E+02~~	2.36E+01 ~~2.45E+01~~	1.75E+02 ~~1.78E+02~~	5.48E-02 ~~5.48E-02~~	2.37E-01 ~~2.37E-01~~	5.84E-02 ~~6.09E-02~~	2.50E-01 ~~2.56E-01~~	1.34E-01 ~~1.34E-01~~	5.23E-01 ~~5.23E-01~~	1.50E-01 ~~1.56E-01~~	5.79E-01 ~~5.94E-01~~	2.47E-03 ~~2.47E-03~~	1.03E-02 ~~1.03E-02~~	4.48E-03 ~~5.70E-03~~	1.80E-02 ~~2.09E-02~~
IC	7.19E+00 ~~7.19E+00~~	5.60E+01 ~~5.60E+01~~	7.87E+00 ~~8.80E+00~~	5.83E+01 ~~6.33E+01~~	1.82E-02 ~~1.82E-02~~	7.89E-02 ~~7.88E-02~~	1.94E-02 ~~2.21E-02~~	8.30E-02 ~~8.97E-02~~	4.47E-02 ~~4.47E-02~~	1.74E-01 ~~1.74E-01~~	4.97E-02 ~~5.65E-02~~	1.92E-01 ~~2.09E-01~~	8.23E-04 ~~8.22E-04~~	3.42E-03 ~~3.42E-03~~	1.49E-03 ~~2.78E-03~~	5.98E-03 ~~9.02E-03~~
BP	3.23E+04 ~~2.91E+03~~	2.64E+05 ~~2.58E+04~~	4.17E+04 ~~3.11E+03~~	2.87E+05 ~~2.69E+04~~	1.37E+02 ~~1.15E+01~~	4.72E+02 ~~4.39E+01~~	1.84E+02 ~~1.23E+01~~	5.31E+02 ~~4.66E+01~~	1.56E+03 ~~7.71E+01~~	4.35E+03 ~~2.73E+02~~	2.36E+03 ~~8.19E+01~~	5.19E+03 ~~2.89E+02~~	1.10E+01 ~~1.53E+00~~	3.49E+01 ~~5.32E+00~~	2.69E+01 ~~2.04E+00~~	6.35E+01 ~~7.00E+00~~
CI	2.01E+04 ~~2.01E+04~~	1.61E+05 ~~1.61E+05~~	2.14E+04 ~~2.14E+04~~	1.67E+05 ~~1.67E+05~~	5.10E+01 ~~5.10E+01~~	2.27E+02 ~~2.27E+02~~	5.40E+01 ~~5.40E+01~~	2.38E+02 ~~2.38E+02~~	1.82E+02 ~~1.82E+02~~	7.30E+02 ~~7.30E+02~~	1.98E+02 ~~1.98E+02~~	7.90E+02 ~~7.90E+02~~	2.47E+00 ~~2.46E+00~~	1.00E+01 ~~1.00E+01~~	4.23E+00 ~~4.23E+00~~	1.70E+01 ~~1.70E+01~~
CFL	7.37E+01 ~~5.32E+03~~	2.56E+03 ~~4.35E+04~~	4.79E+04 ~~5.84E+03~~	4.14E+05 ~~4.62E+04~~	3.84E-02 ~~2.58E+01~~	1.17E-01 ~~9.23E+01~~	1.33E+02 ~~2.80E+01~~	4.42E+02 ~~9.92E+01~~	7.28E-02 ~~2.20E+02~~	2.38E-01 ~~7.30E+02~~	7.72E+00 ~~2.35E+02~~	2.43E+01 ~~7.76E+02~~	2.20E-03 ~~3.01E+00~~	6.70E-03 ~~9.90E+00~~	5.11E+00 ~~4.34E+00~~	1.61E+01 ~~1.42E+01~~

Table 49-8

SITE BOUNDARY WHOLE BODY EDE DOSE RISK – 24 HOURS

Release Category	Release Frequency (/Reactor Year)	Mean Dose (Sieverts)	Dose (REM)	Risk (REM/Reactor Year)	Percentage Contribution to Total Risk
CFI	1.89E-10	2.59E+01~~3.25E+01~~	2.59E+03~~3.25E+03~~	4.90E-07~~6.14E-07~~	0.3%~~1.2%~~
CFE	7.47E-09	4.23E+01~~4.23E+01~~	4.23E+03~~4.23E+03~~	3.16E-05~~3.16E-05~~	17.3%~~61.4%~~
IC	2.21E-07	1.82E-02~~1.82E-02~~	1.82E+00~~1.82E+00~~	4.02E-07~~4.02E-07~~	0.2%~~0.8%~~
BP	1.05E-08	1.37E+02~~1.15E+01~~	1.37E+04~~1.15E+03~~	1.44E-04~~1.21E-05~~	78.6%~~23.5%~~
CI	1.33E-09	5.10E+01~~5.10E+01~~	5.10E+03~~5.10E+03~~	6.78E-06~~6.78E-06~~	3.7%~~13.2%~~
CFL	3.45E-13	3.84E-02~~2.58E+01~~	3.84E+00~~2.58E+03~~	1.32E-12~~8.90E-10~~	0.0%~~0.0%~~
			Total Risk =~~Total Risk =~~	1.83E-04~~5.150E-05~~	100.0%~~100.0%~~

Table 49-9

SITE BOUNDARY WHOLE BODY EDE DOSE RISK – 72 HOURS

Release Category	Release Frequency (/Reactor Year)	Mean Dose (Sieverts)	Dose (REM)	Risk (REM/Reactor Year)	Percentage Contribution to Total Risk
CFI	1.89E-10	3.72E+01~~3.49E+01~~	3.72E+03~~3.49E+03~~	7.03E-07~~6.60E-07~~	0.3%~~1.2%~~
CFE	7.47E-09	4.60E+01~~4.60E+01~~	4.60E+03~~4.60E+03~~	3.44E-05~~3.44E-05~~	14.6%~~61.8%~~
IC	2.21E-07	1.94E-02~~2.21E-02~~	1.94E+00~~2.21E+00~~	4.29E-07~~4.88E-07~~	0.2%~~0.9%~~
BP	1.05E-08	1.84E+02~~1.23E+01~~	1.84E+04~~1.23E+03~~	1.93E-04~~1.29E-05~~	81.9%~~23.2%~~
CI	1.33E-09	5.40E+01~~5.40E+01~~	5.40E+03~~5.40E+03~~	7.18E-06~~7.18E-06~~	3.0%~~12.9%~~
CFL	3.45E-13	1.33E+02~~2.80E+01~~	1.33E+04~~2.80E+03~~	4.59E-09~~9.66E-10~~	0.0%~~0.0%~~
			Total Risk =~~Total Risk=~~	2.36E-04~~5.56E-05~~	100.0%~~100.0%~~

Table 49-10

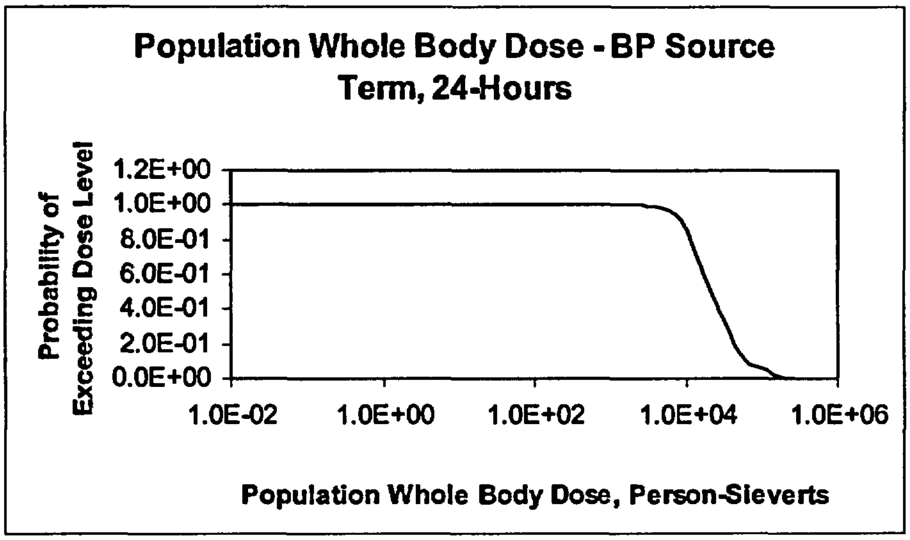
POPULATION WHOLE BODY EDE DOSE RISK – 24 HOURS

Release Category	Release Frequency (/Reactor Year)	Mean Dose (Person-Sieverts)	Dose (Person-REM)	Risk (Person-REM/ Reactor Year)	Percentage Contribution to Total Risk
CFI	1.89E-10	7.03E+03~~7.88E+03~~	7.03E+05~~7.88E+05~~	1.33E-04~~1.49E-04~~	0.3%~~1.2%~~
CFE	7.47E-09	8.51E+03~~8.51E+03~~	8.51E+05~~8.51E+05~~	6.36E-03~~6.36E-03~~	14.7%~~51.3%~~
IC	2.21E-07	7.19E+00~~7.19E+00~~	7.19E+02~~7.19E+02~~	1.59E-04~~1.59E-04~~	0.4%~~1.3%~~
BP	1.05E-08	3.23E+04~~2.91E+03~~	3.23E+06~~2.91E+05~~	3.39E-02~~3.06E-03~~	78.4%~~24.7%~~
CI	1.33E-09	2.01E+04~~2.01E+04~~	2.01E+06~~2.01E+06~~	2.67E-03~~2.67E-03~~	6.2%~~21.6%~~
CFL	3.45E-13	7.37E+01~~5.32E+03~~	7.37E+03~~5.32E+05~~	2.54E-09~~1.84E-07~~	0.0%~~0.0%~~
			Total Risk =~~Total Risk =~~	4.32E-02~~1.24E-02~~	100.0%~~100.0%~~

Table 49-11

POPULATION WHOLE BODY EDE DOSE RISK – 72 HOURS

Release Category	Release Frequency (/Reactor Year)	Mean Dose (Person-Sieverts)	Dose (Person-REM)	Risk (Person-REM/ Reactor Year)	Percentage Contribution to Total Risk
CFI	1.89E-10	1.13E+04~~8.89E+03~~	1.13E+06~~8.89E+05~~	2.14E-04~~1.68E-04~~	0.4%~~1.2%~~
CFE	7.47E-09	9.36E+03~~9.36E+03~~	9.36E+05~~9.36E+05~~	6.99E-03~~6.99E-03~~	12.9%~~51.9%~~
IC	2.21E-07	7.87E+00~~8.80E+00~~	7.87E+02~~8.80E+02~~	1.74E-04~~1.94E-04~~	0.3%~~1.4%~~
BP	1.05E-08	4.17E+04~~3.11E+03~~	4.17E+06~~3.11E+05~~	4.38E-02~~3.27E-03~~	81.1%~~24.2%~~
CI	1.33E-09	2.14E+04~~2.14E+04~~	2.14E+06~~2.14E+06~~	2.85E-03~~2.85E-03~~	5.3%~~21.1%~~
CFL	3.45E-13	4.79E+04~~5.84E+03~~	4.79E+06~~5.84E+05~~	1.65E-06~~2.01E-07~~	0.0%~~0.0%~~
			Total Risk =~~Total Risk =~~	5.40E-02~~1.35E-02~~	100.0%~~100.0%~~



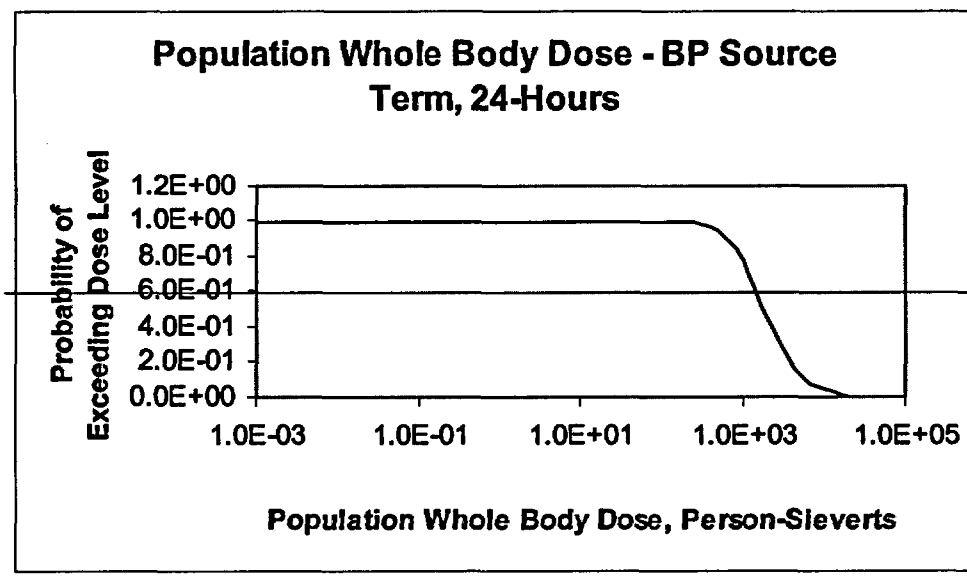
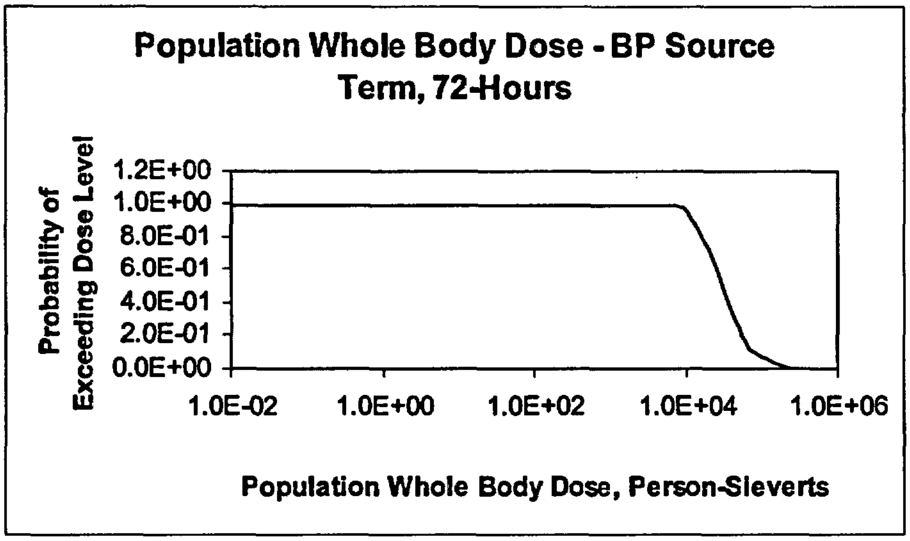


Figure 49-1

Population Whole Body Dose – BP Source Term, 24 Hours



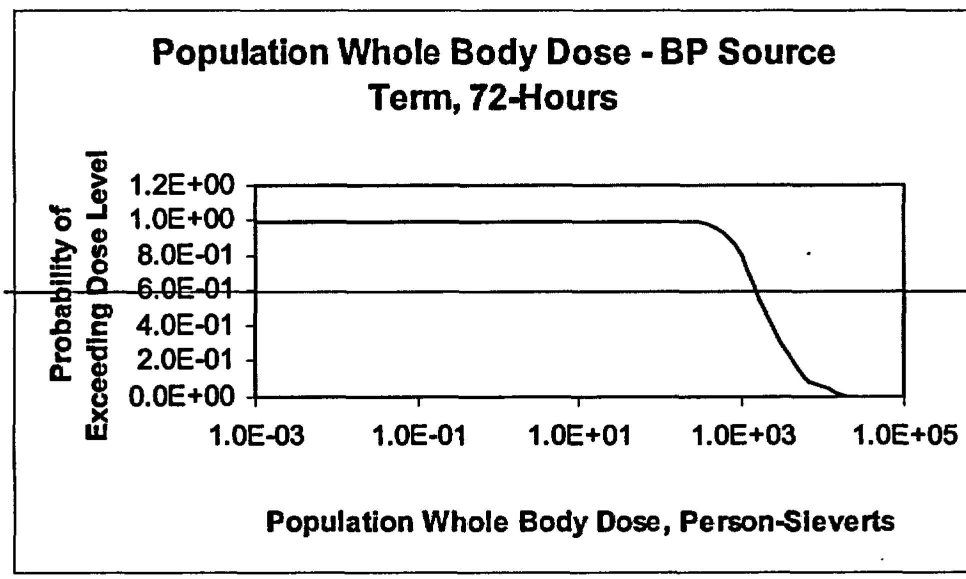
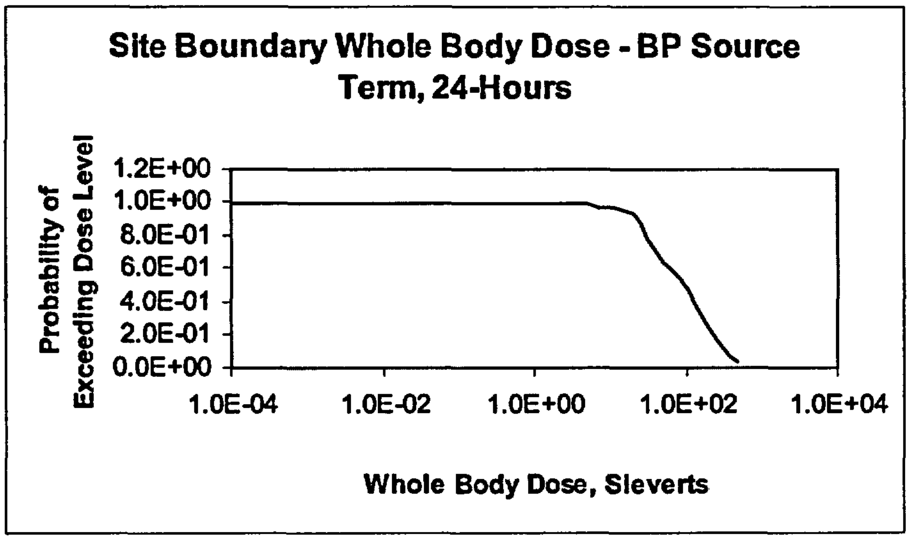


Figure 49-2

Population Whole Body Dose – BP Source Term 72 Hours



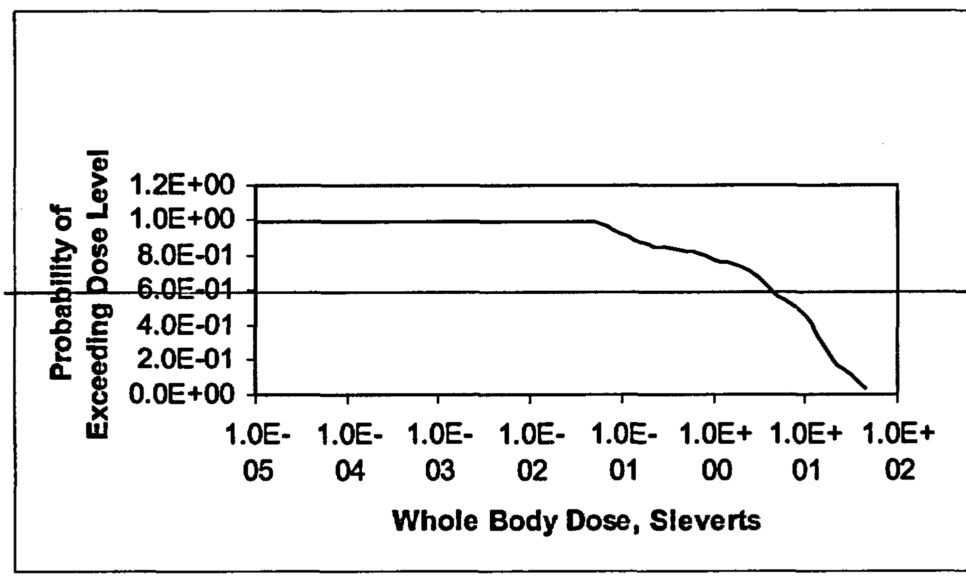
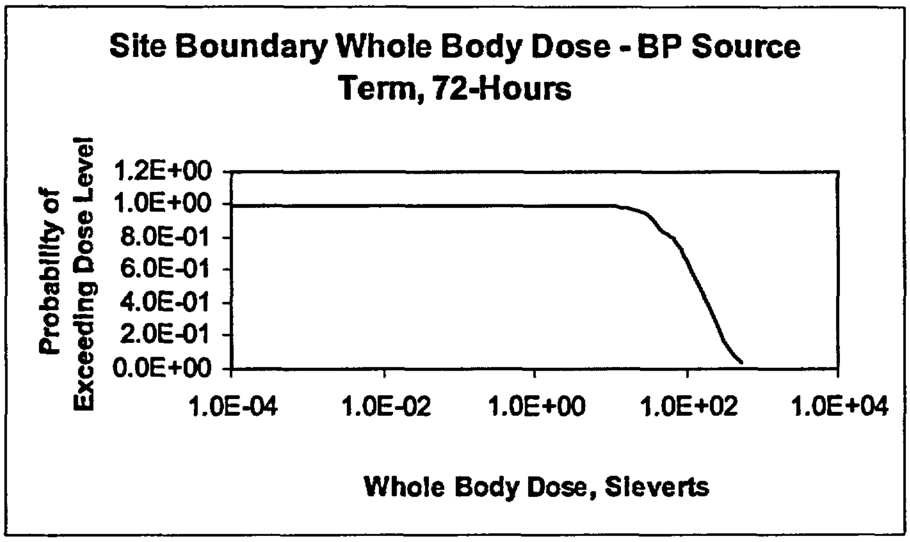


Figure 49-3

Site Boundary Whole Body Dose – BP Source Term, 24 Hours



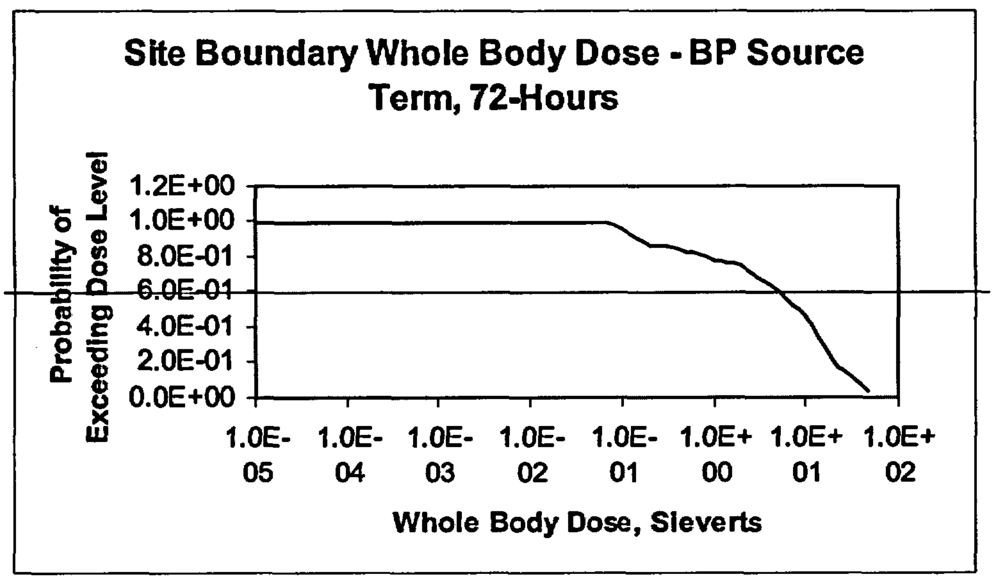
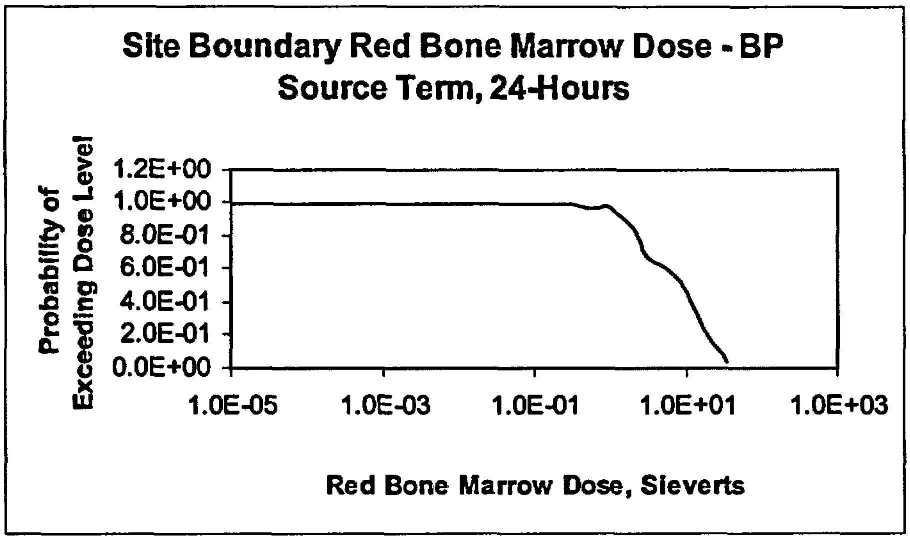


Figure 49-4

Site Boundary Whole Body Dose – BP Source Term, 72 Hours



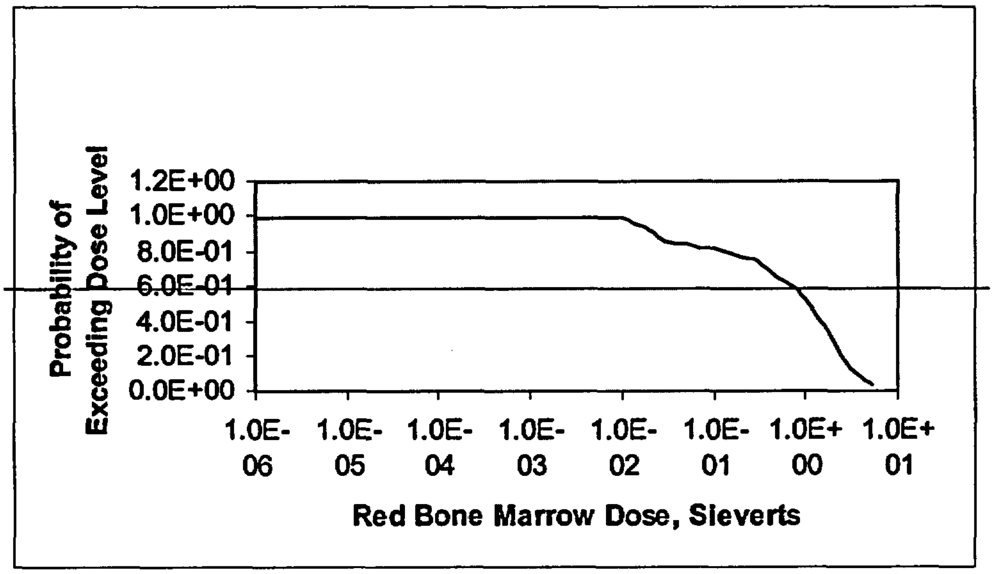
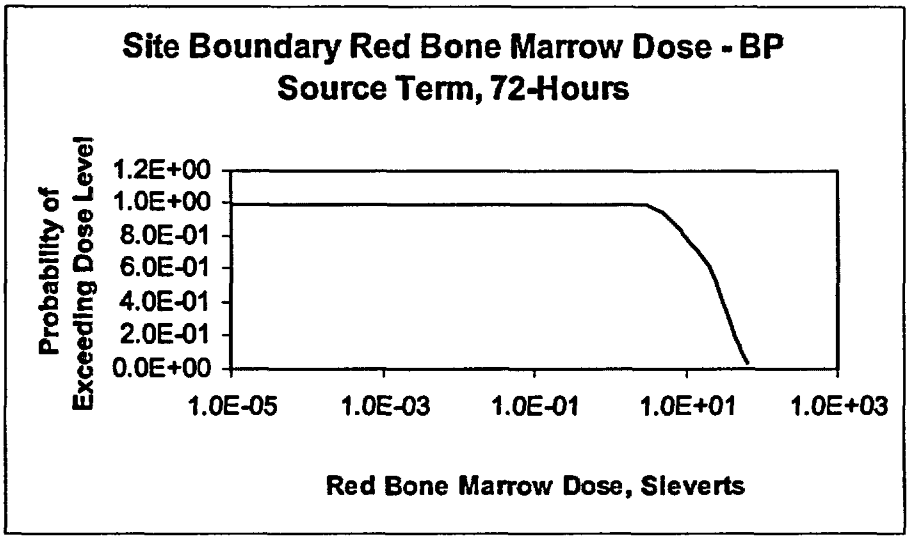


Figure 49-5

Site Boundary Red Bone Marrow Dose – BP Source Term, 24 Hours



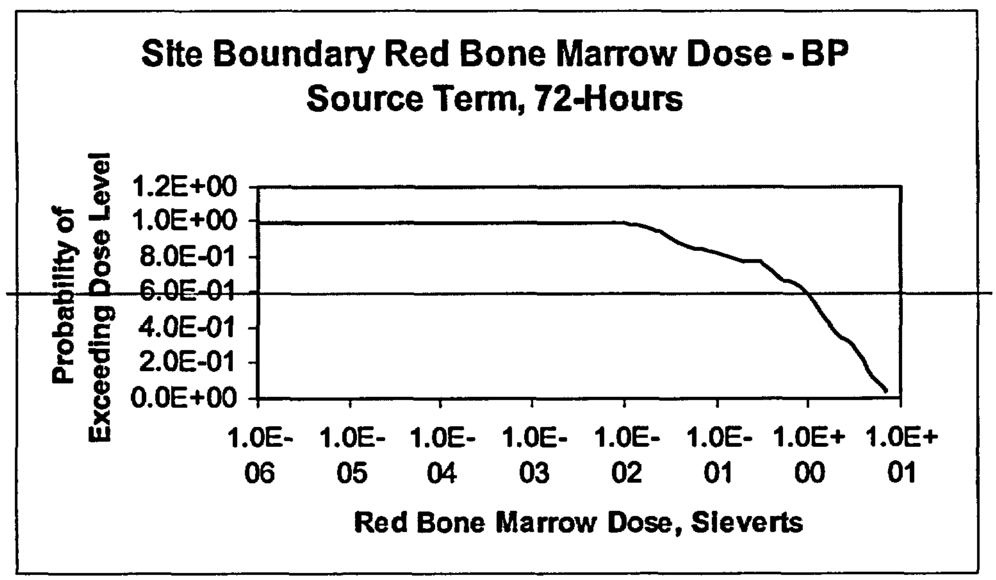
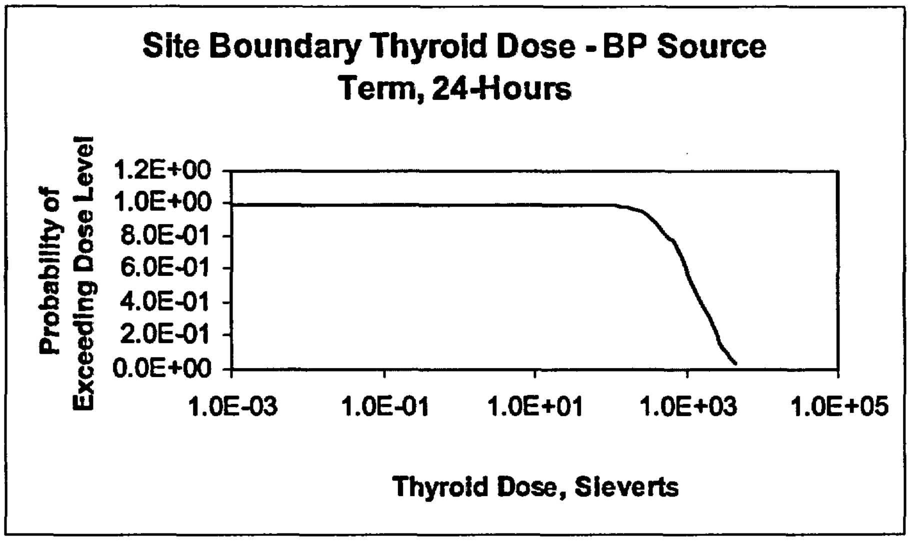


Figure 49-6

Site Boundary Red Bone Marrow Dose – BP Source Term, 72 Hours



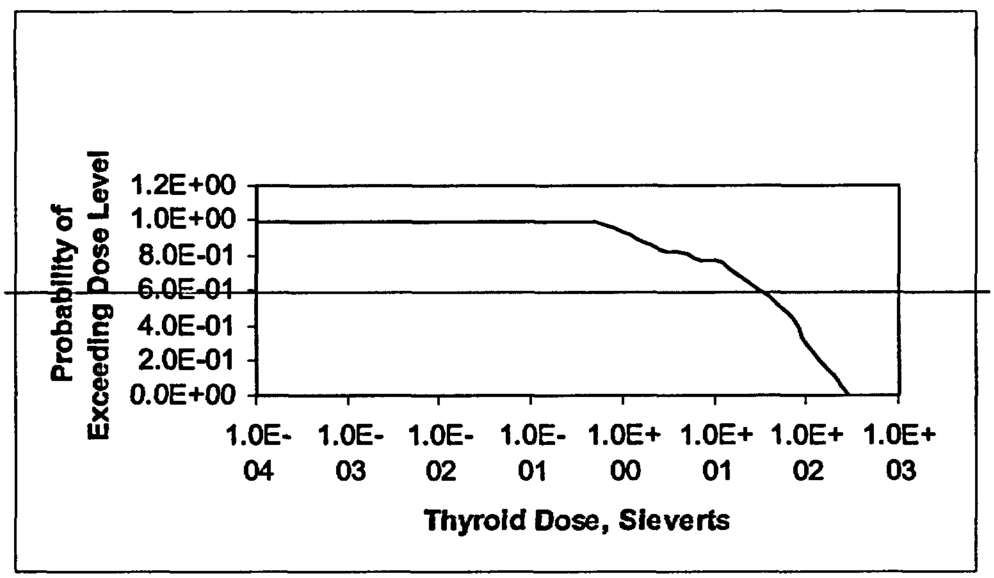
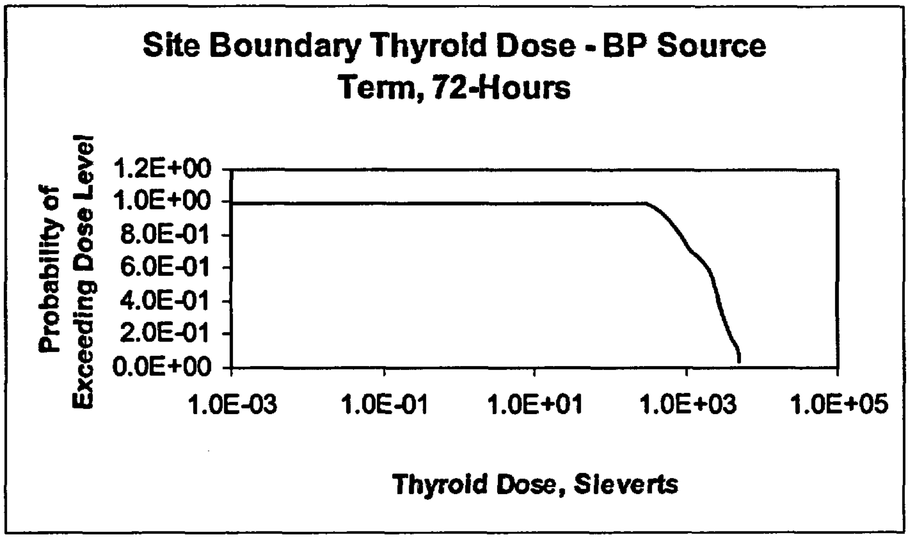


Figure 49-7

Site Boundary Thyroid Dose – BP Source Term, 24 Hours



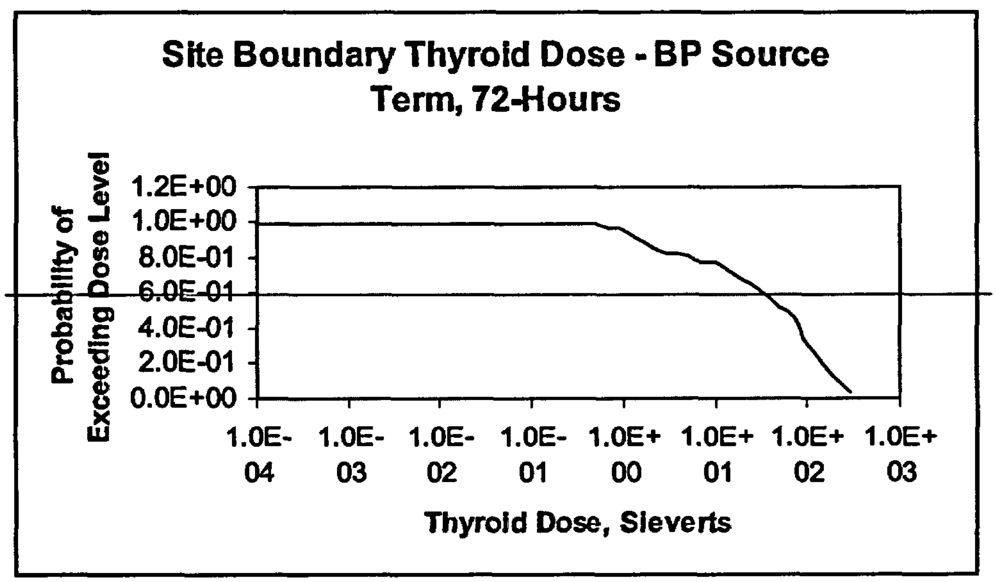
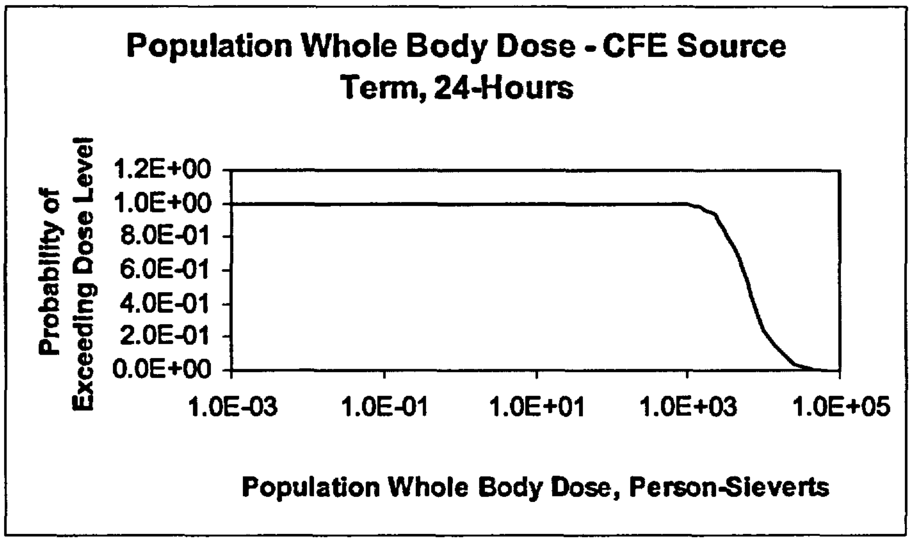


Figure 49-8

Site Boundary Thyroid Dose – BP Source Term, 72 Hours



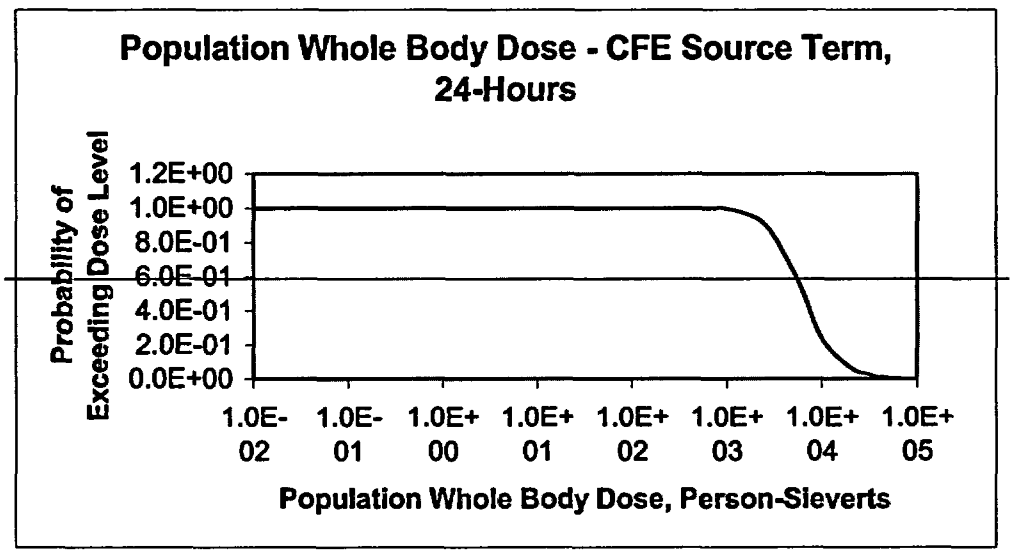
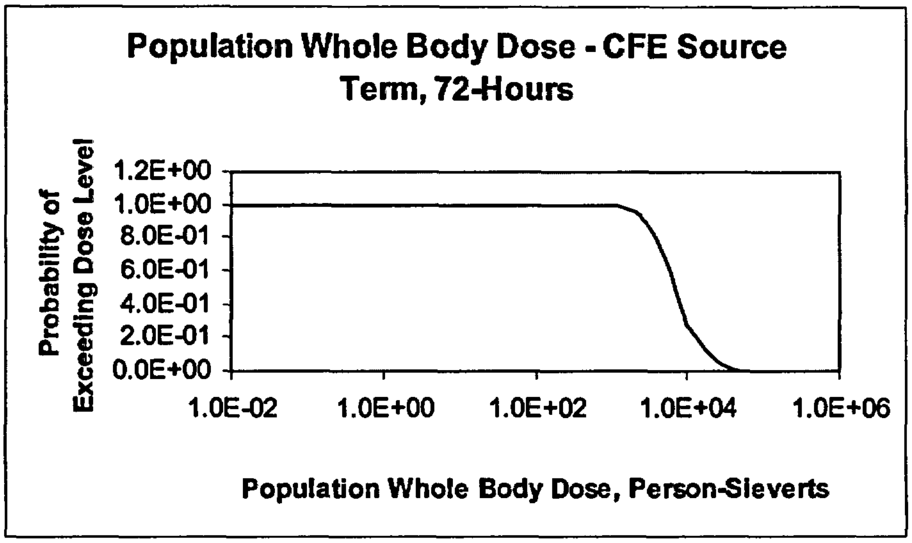


Figure 49-9

Population Whole Body Dose – CFE Source Term, 24 Hours



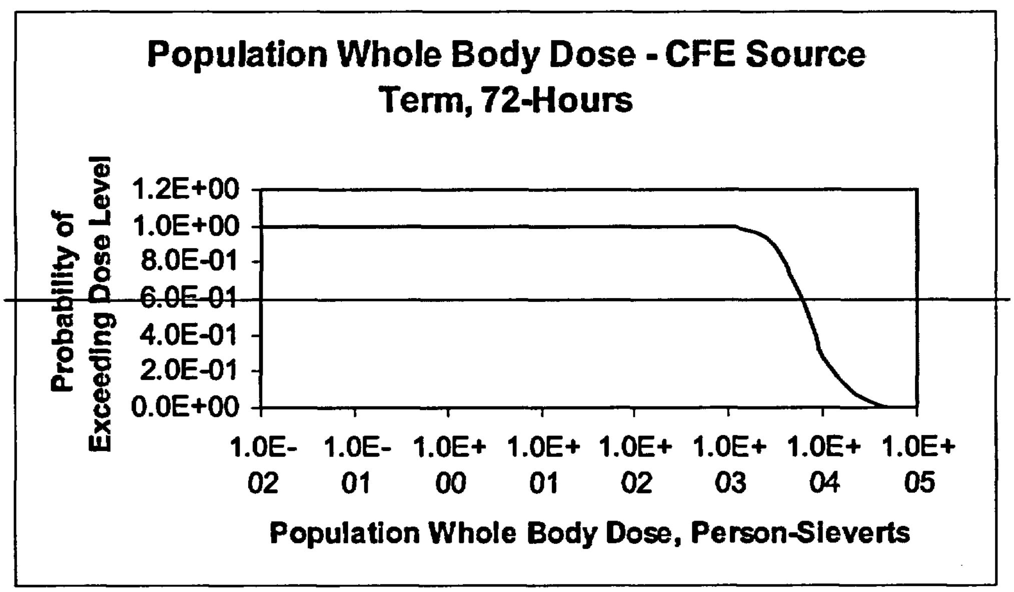
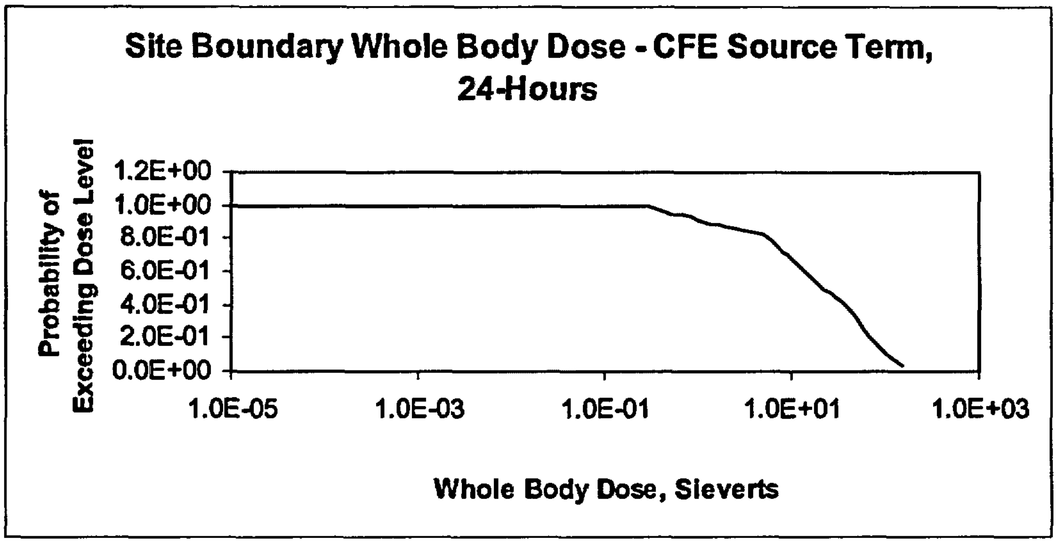


Figure 49-10

Population Whole Body Dose – CFE Source Term, 72 Hours



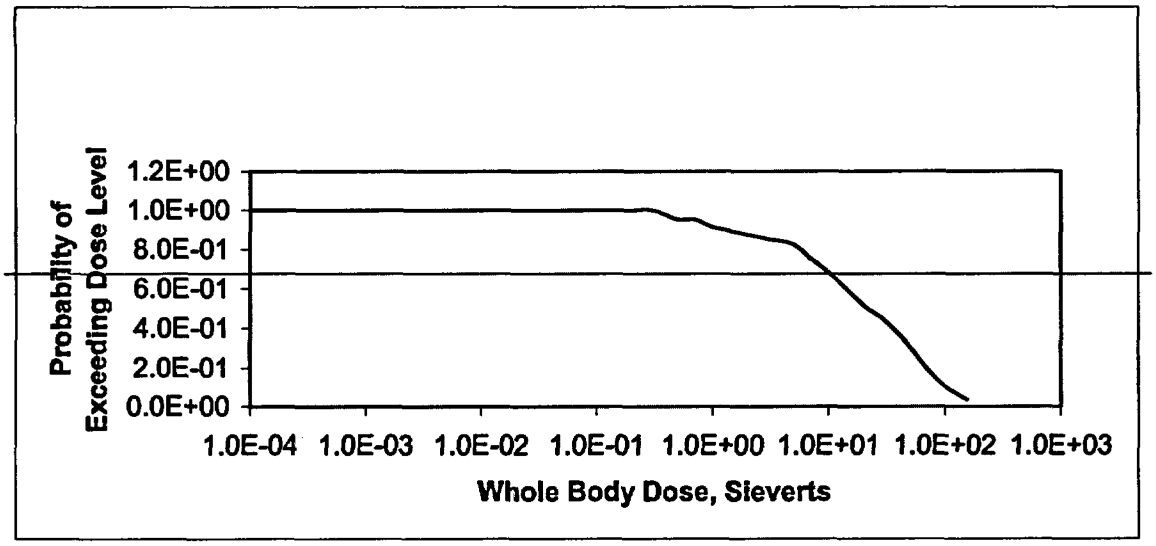
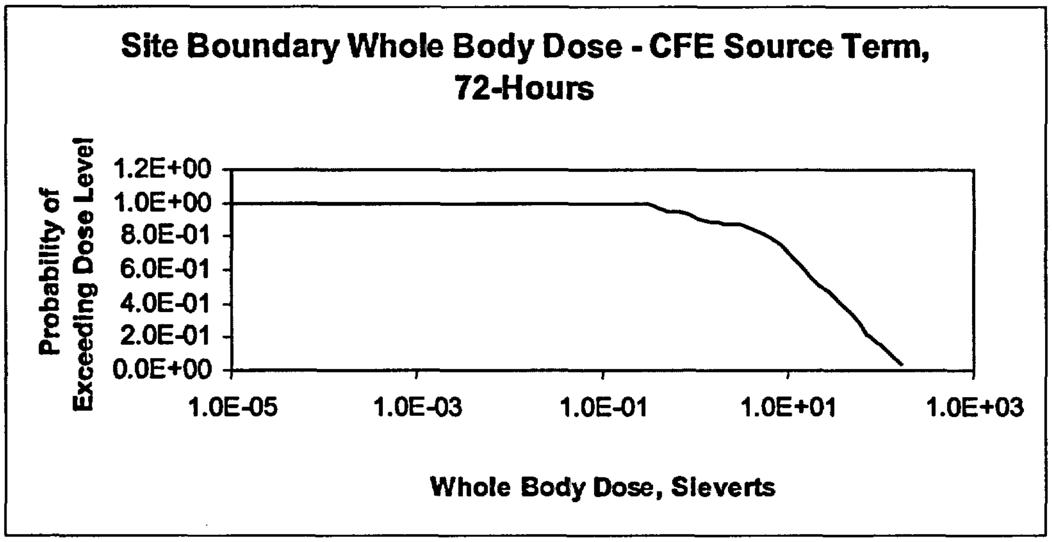


Figure 49-11

Site Boundary Whole Body Dose – CFE Source Term, 24 Hours



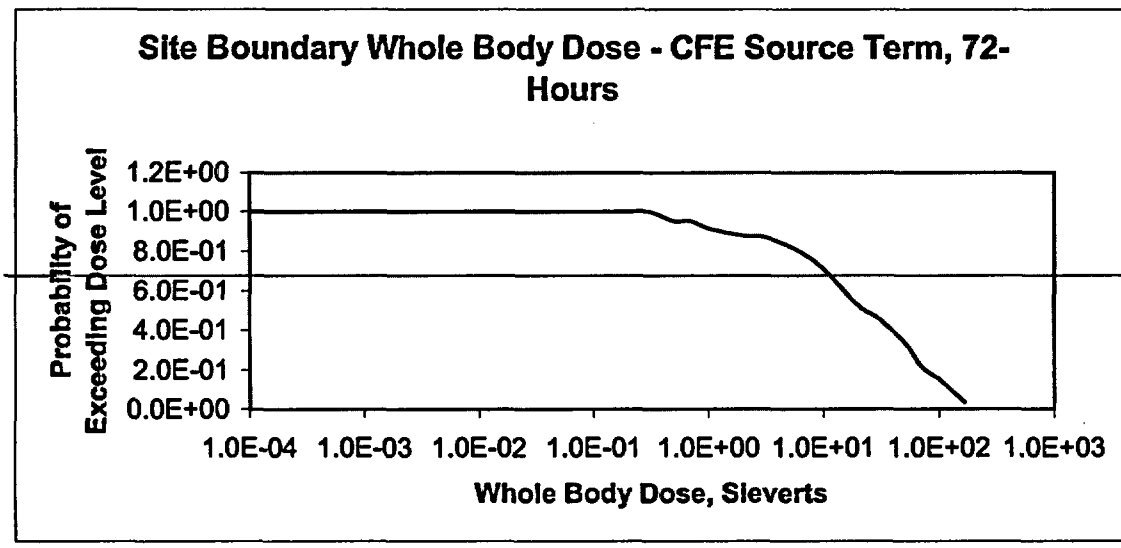
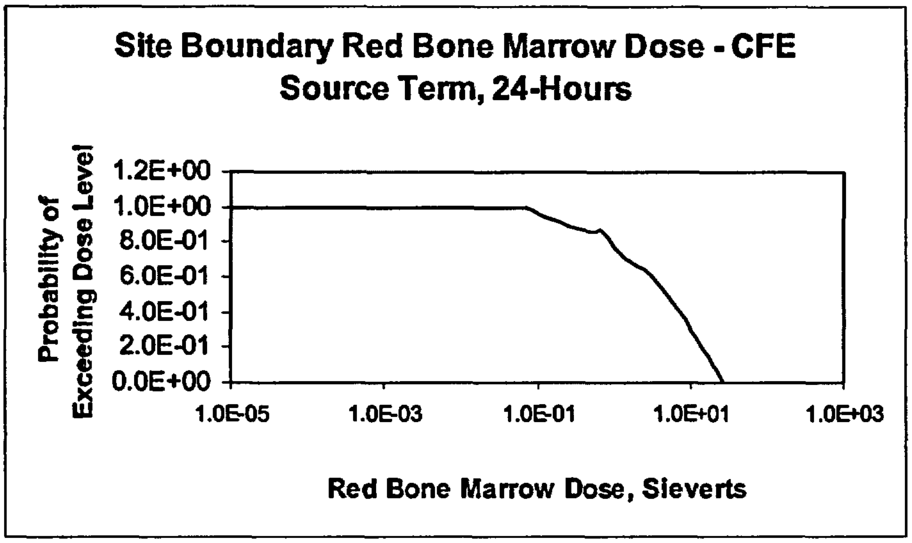


Figure 49-12

Site Boundary Whole Body Dose – CFE Source Term, 72 Hours



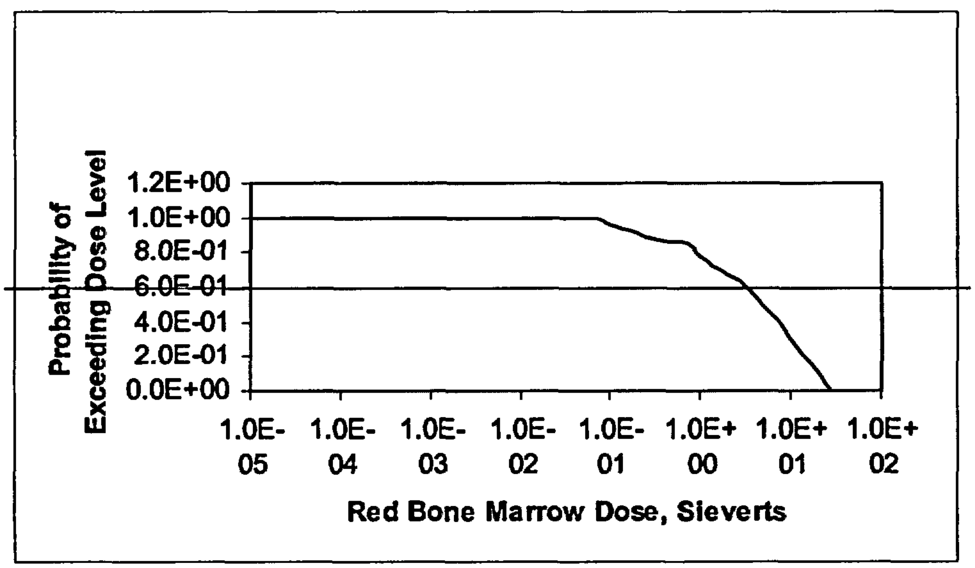
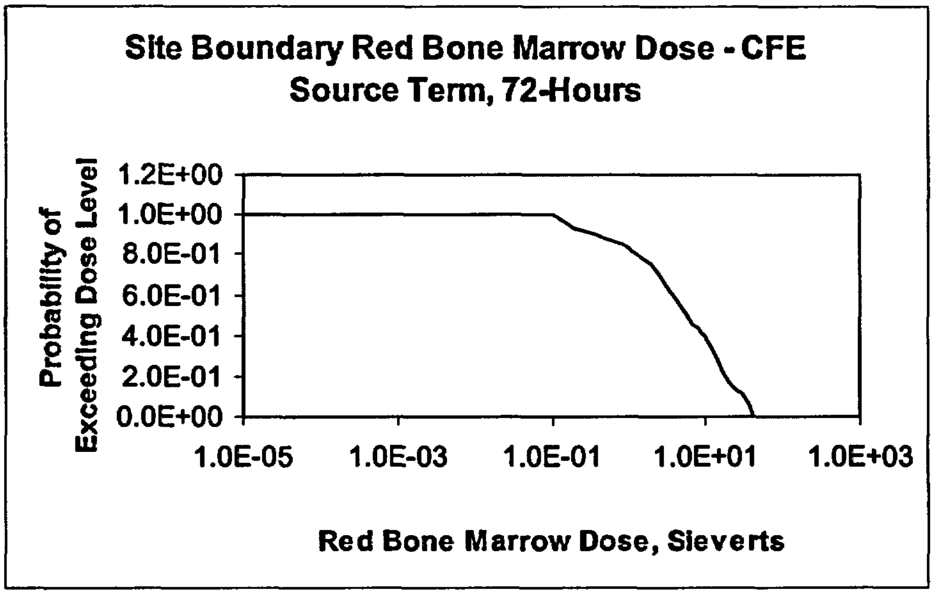


Figure 49-13

Site Boundary Red Bone Marrow Dose – CFE Source Term, 24 Hours



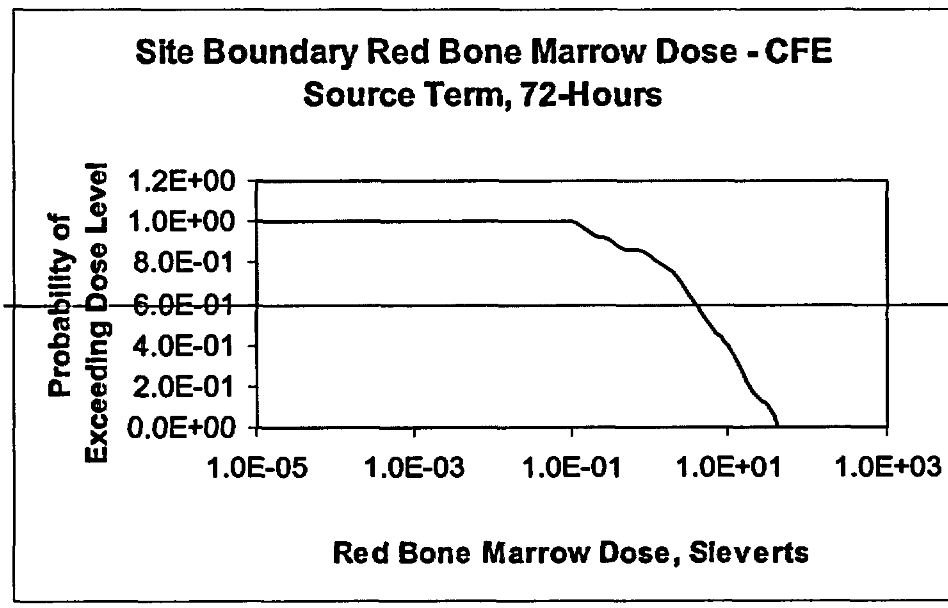
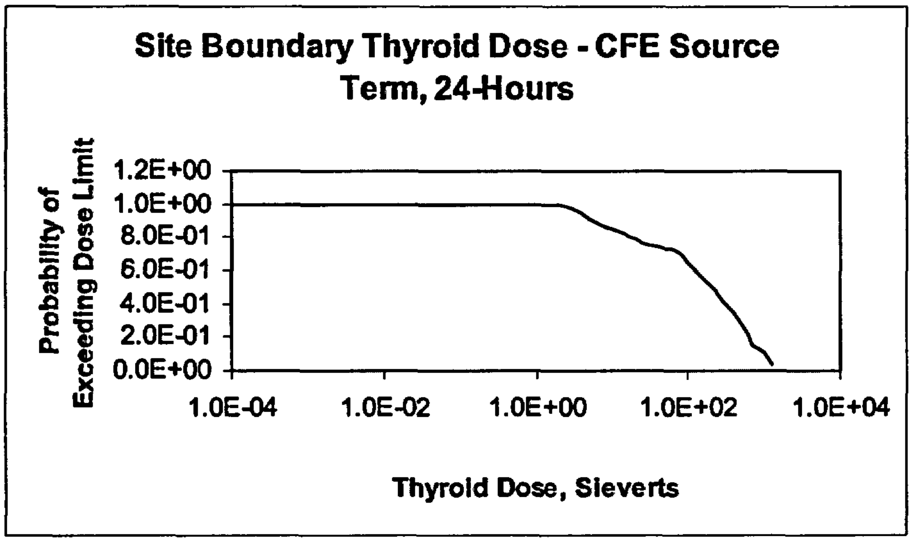


Figure 49-14

Site Boundary Red Bone Marrow Dose – CFE Source Term, 72 Hours



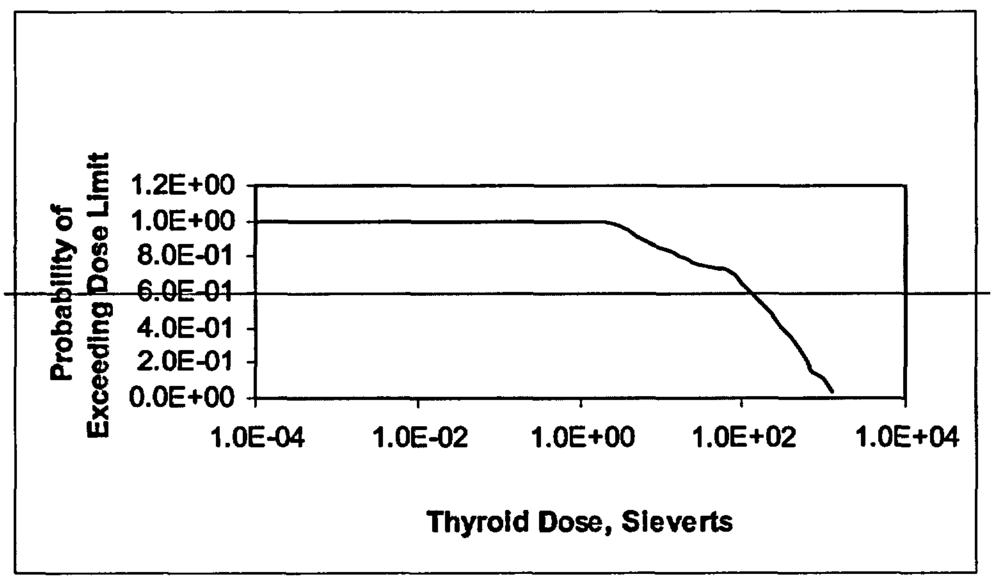
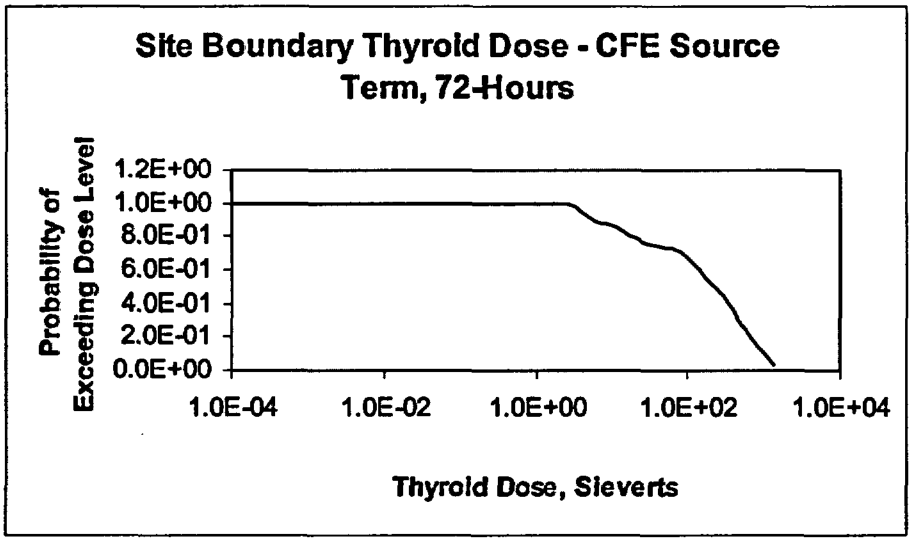


Figure 49-15

Site Boundary Thyroid Dose – CFE Source Term, 24 Hours



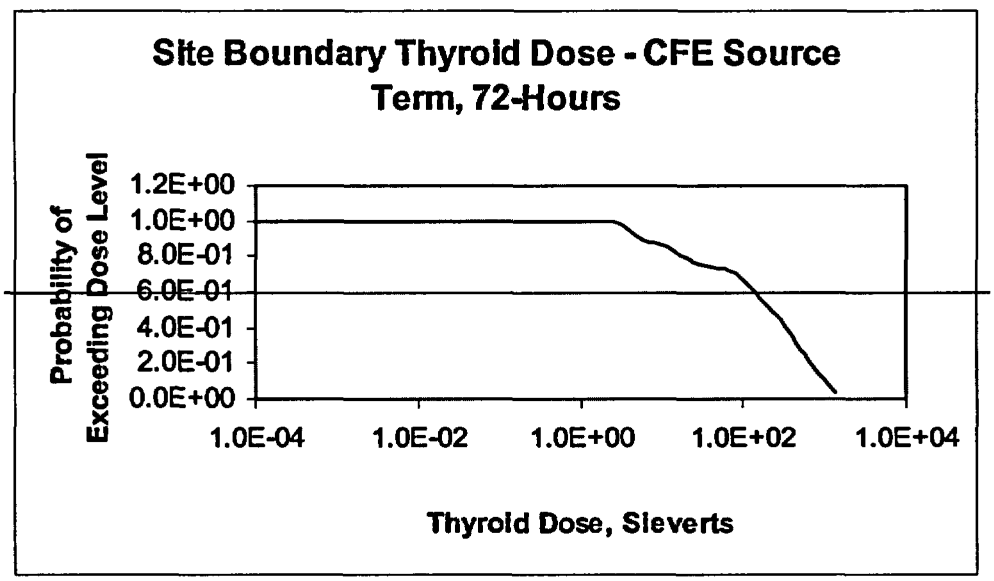
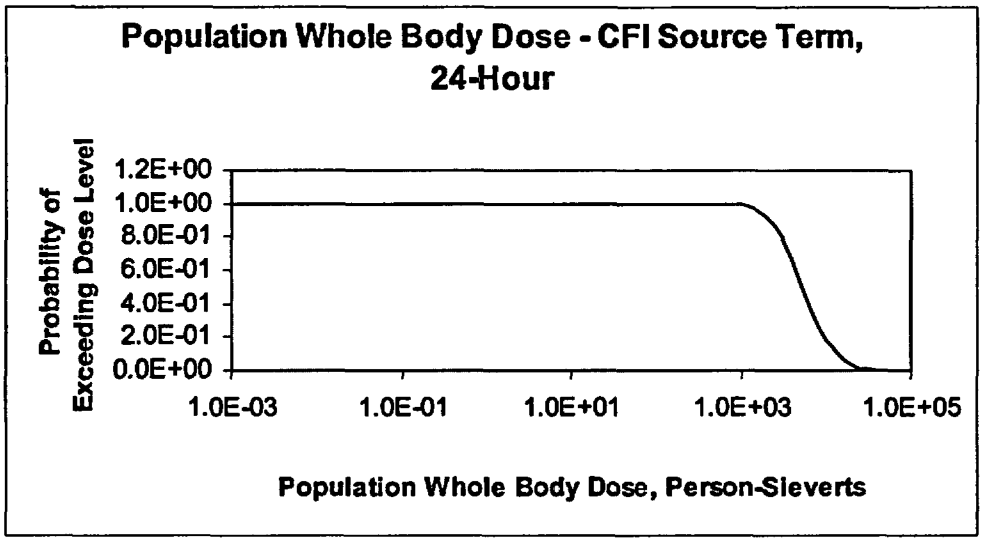


Figure 49-16

Site Boundary Thyroid Dose – CFE Source Term, 72 Hours



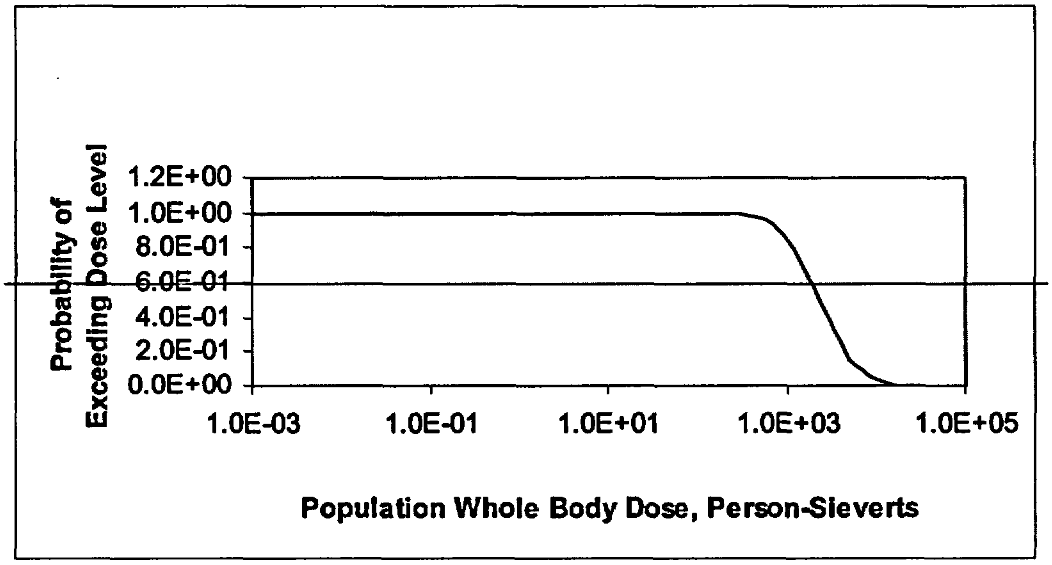
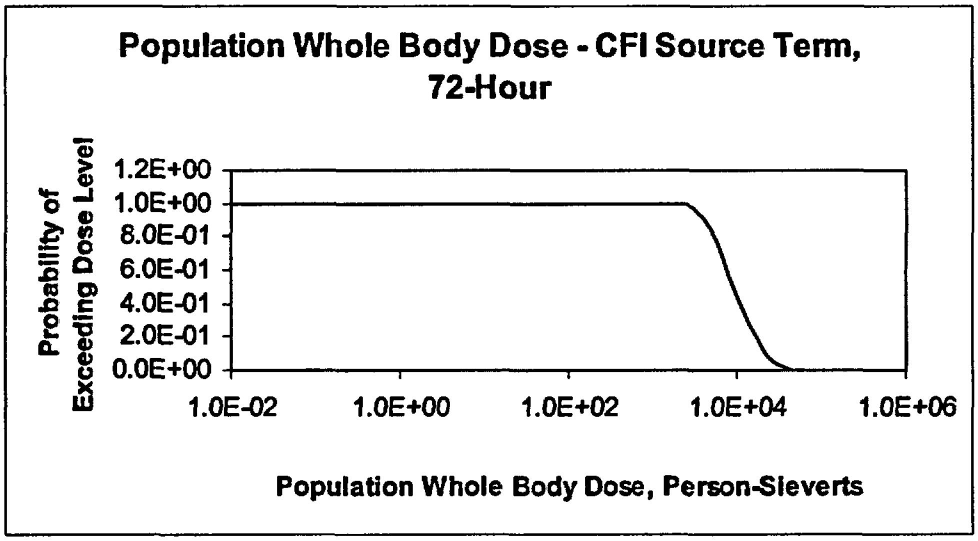


Figure 49-17

Population Whole Body Dose – CFI Source Term, 24 Hours



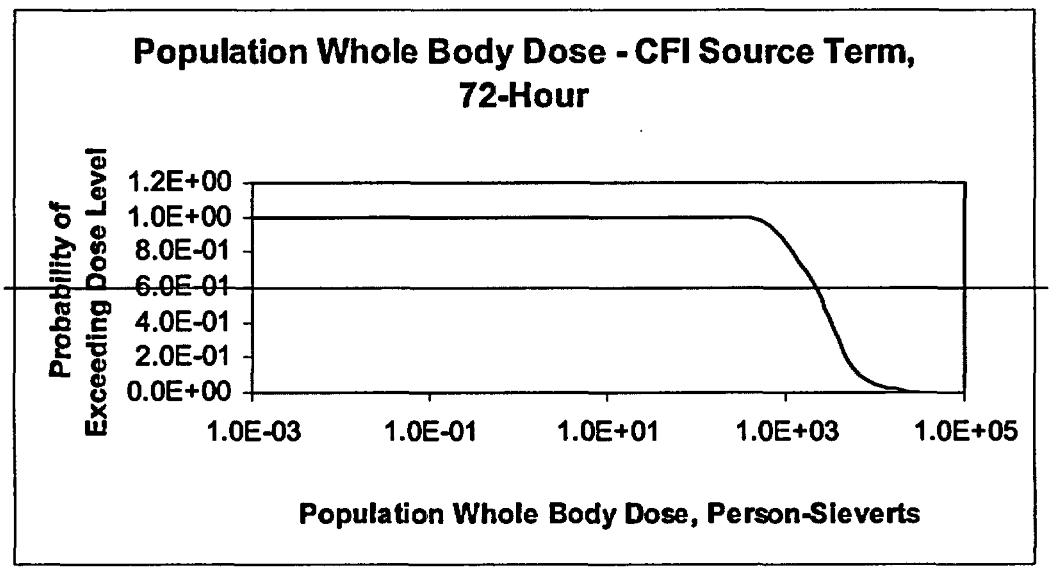
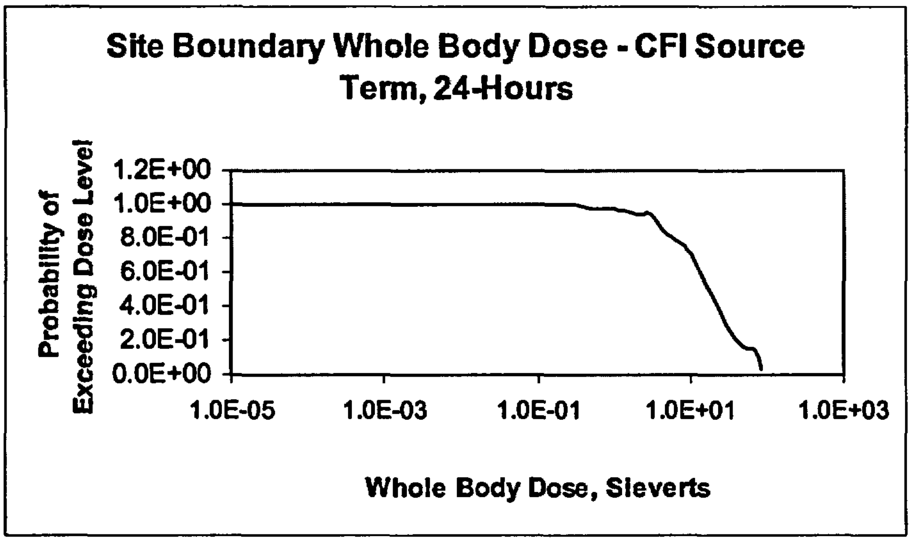


Figure 49-18

Population Whole Body Dose – CFI Source Term, 72 Hours



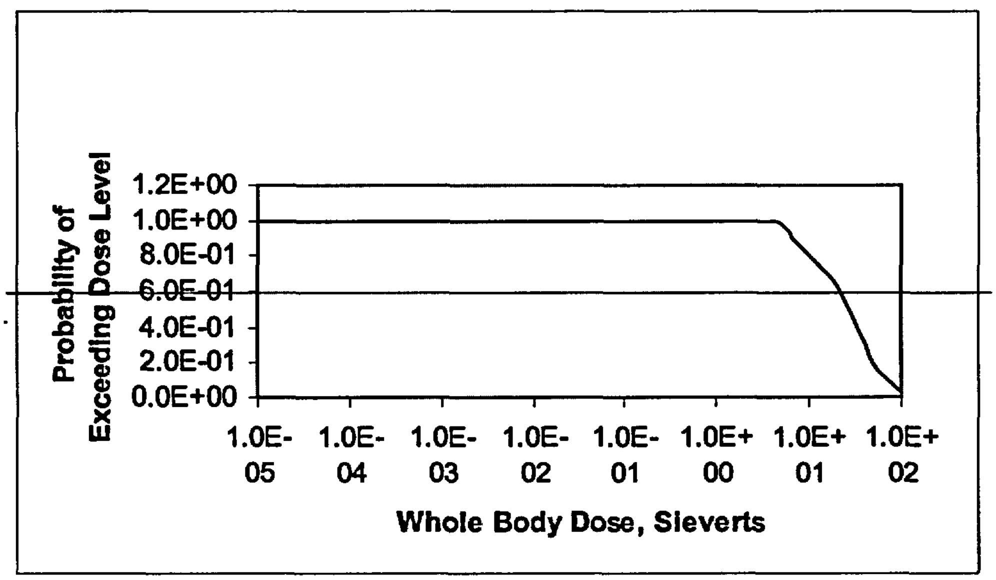
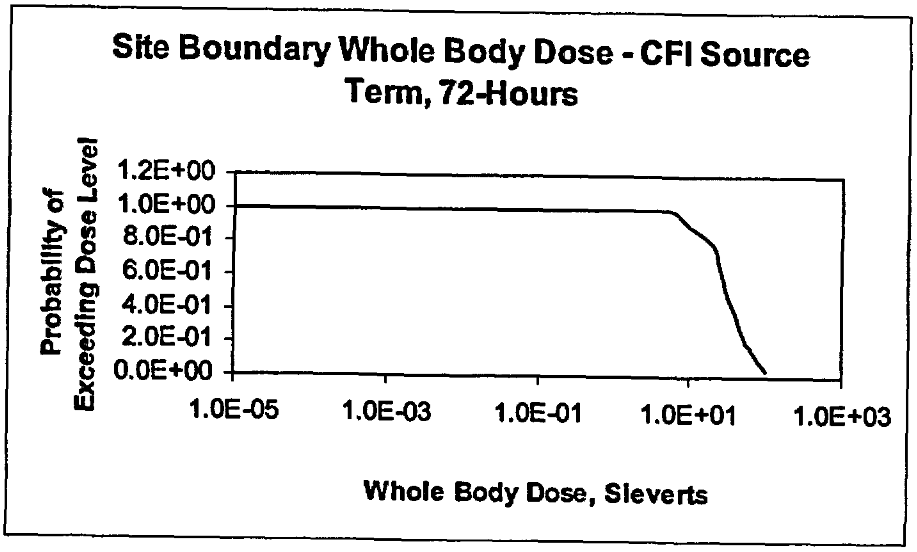


Figure 49-19

Site Boundary Whole Body Dose – CFI Source Term, 24 Hours



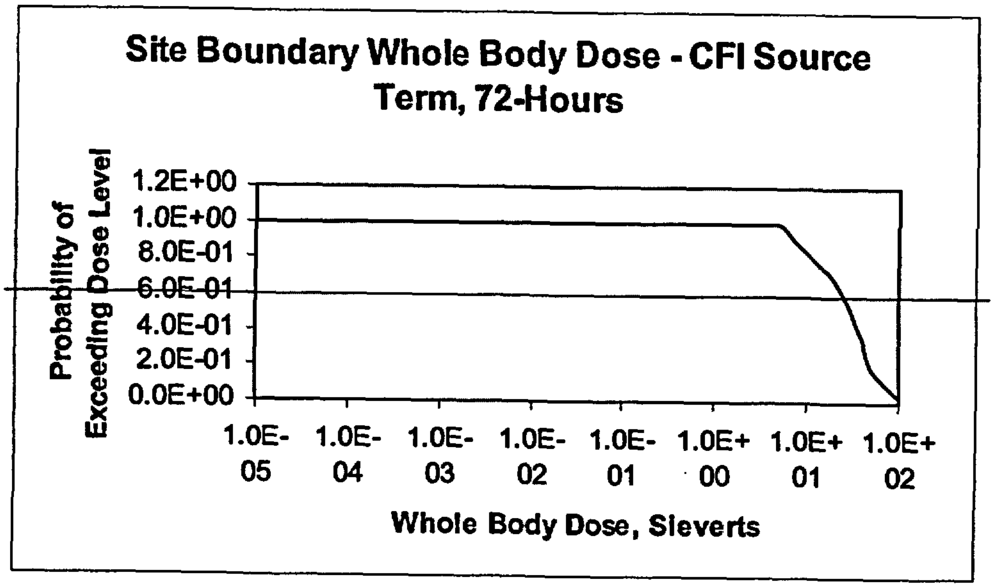
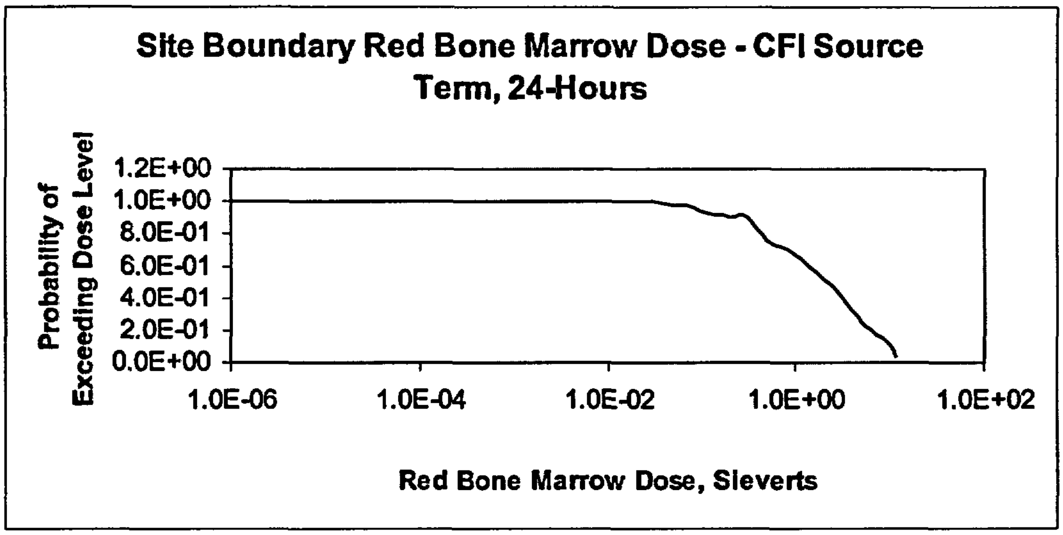


Figure 49-20

Site Boundary Whole Body Dose – CFI Source Term, 72 Hours



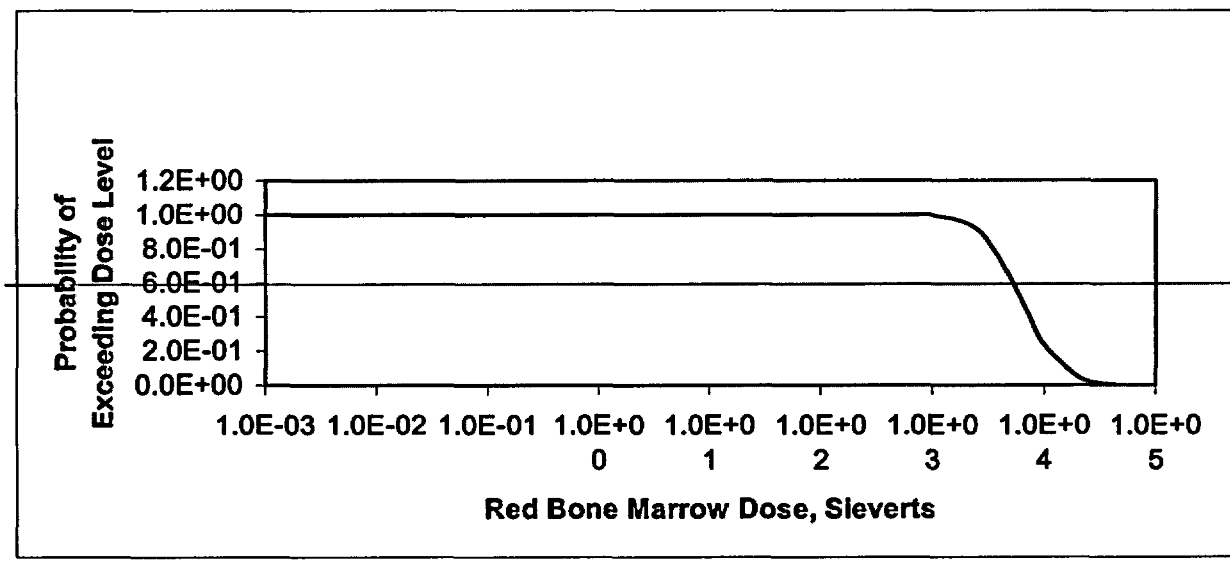
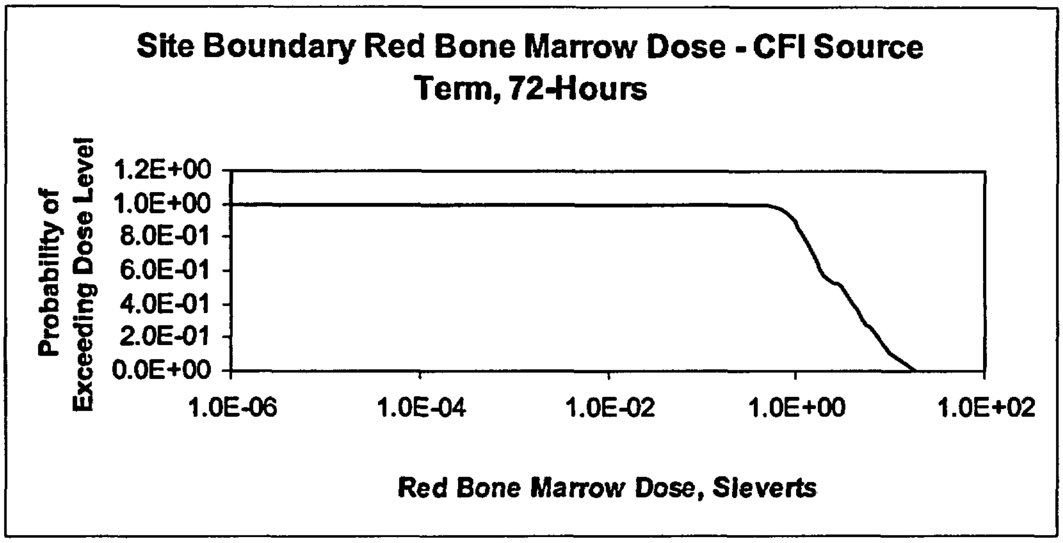


Figure 49-21

Site Boundary Red Bone Marrow Dose – CFI Source Term, 24 Hours



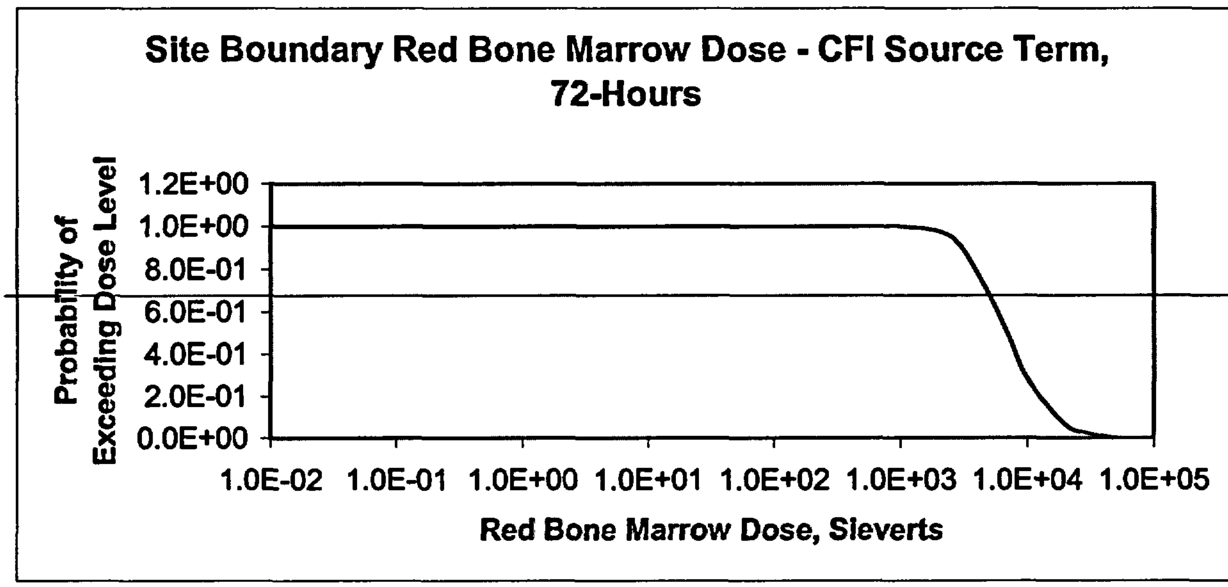
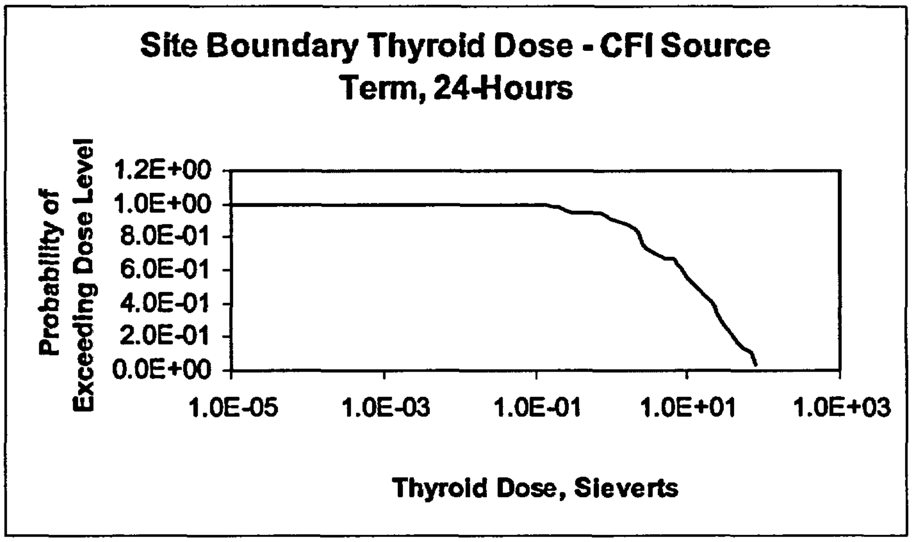


Figure 49-22

Site Boundary Red Bone Marrow Dose – CFI Source Term, 72 Hours



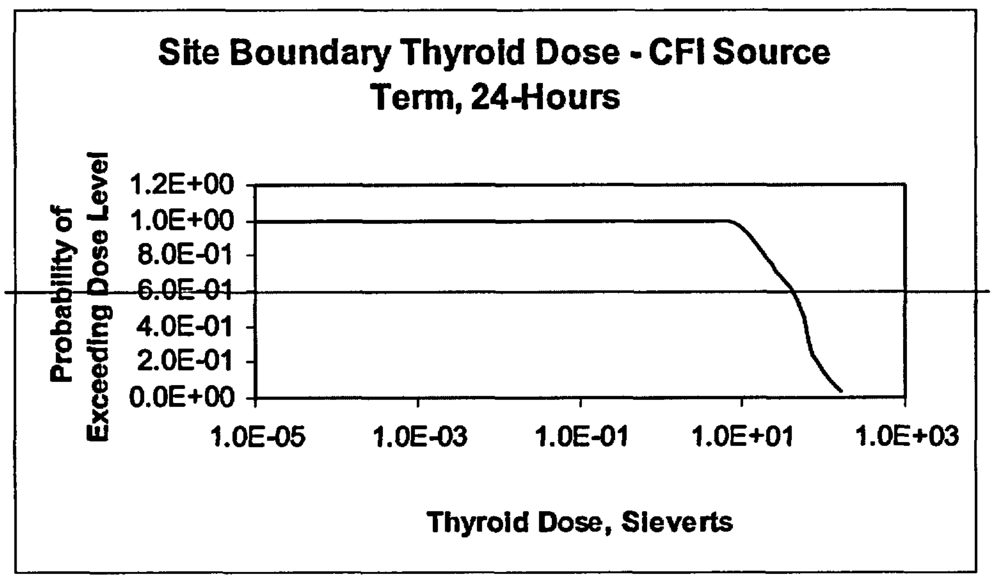
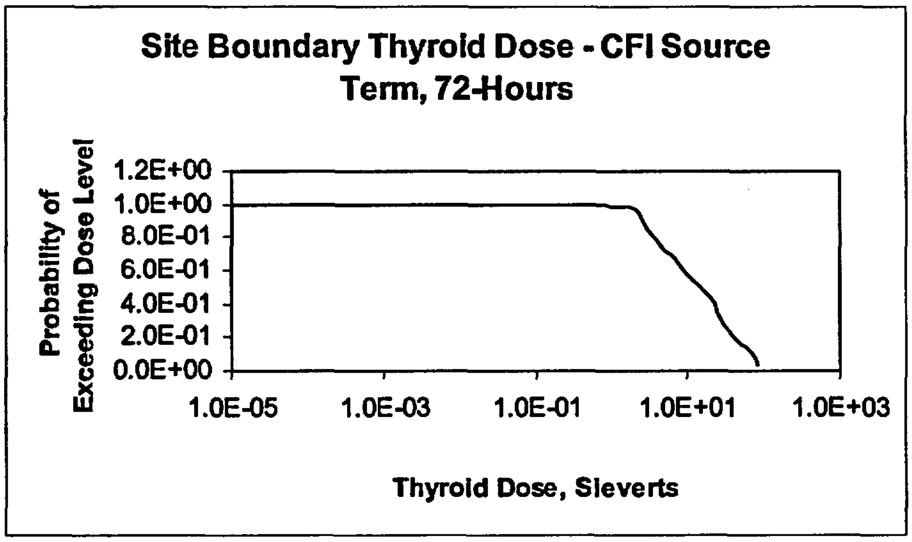


Figure 49-23

Site Boundary Thyroid Dose – CFI Source Term, 24 Hours



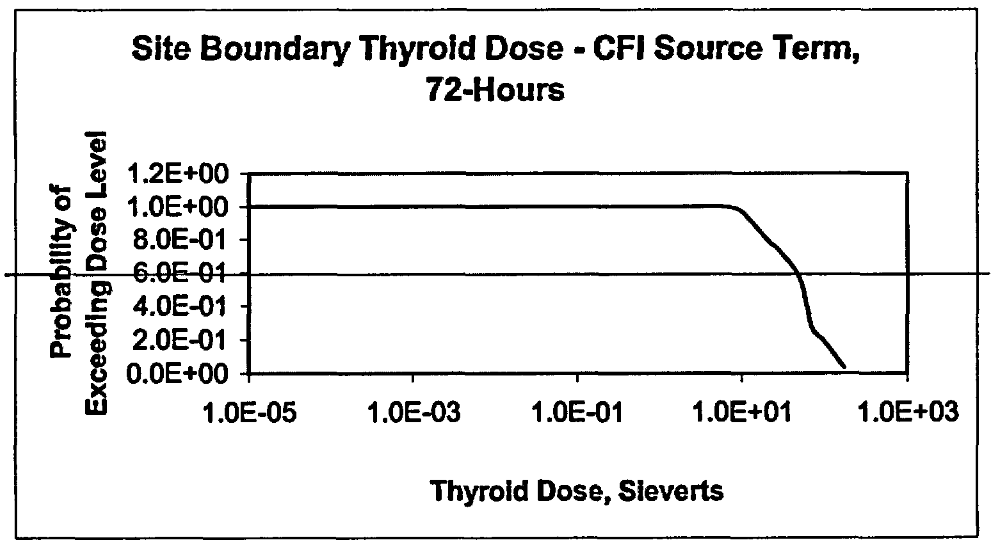
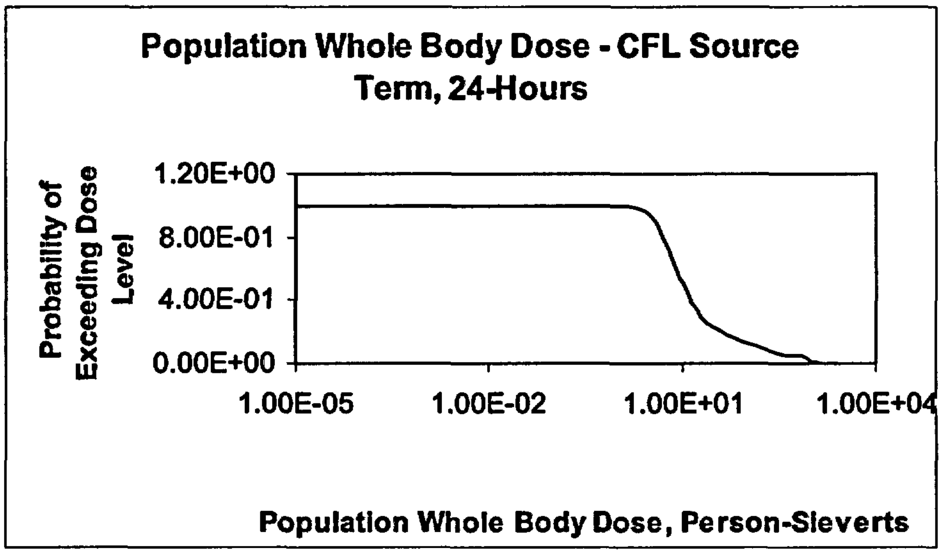


Figure 49-24

Site Boundary Thyroid Dose – CFI Source Term, 72 Hours



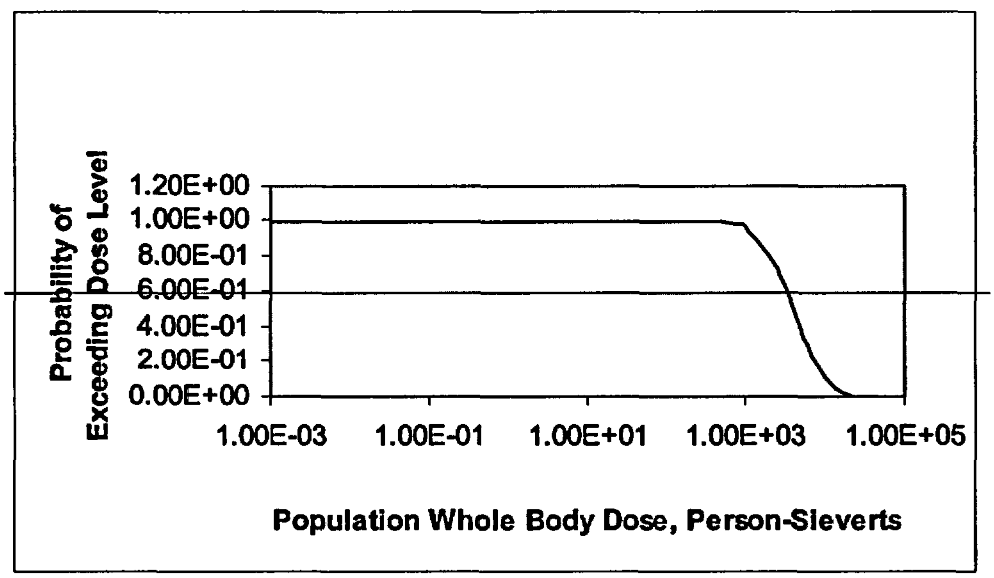
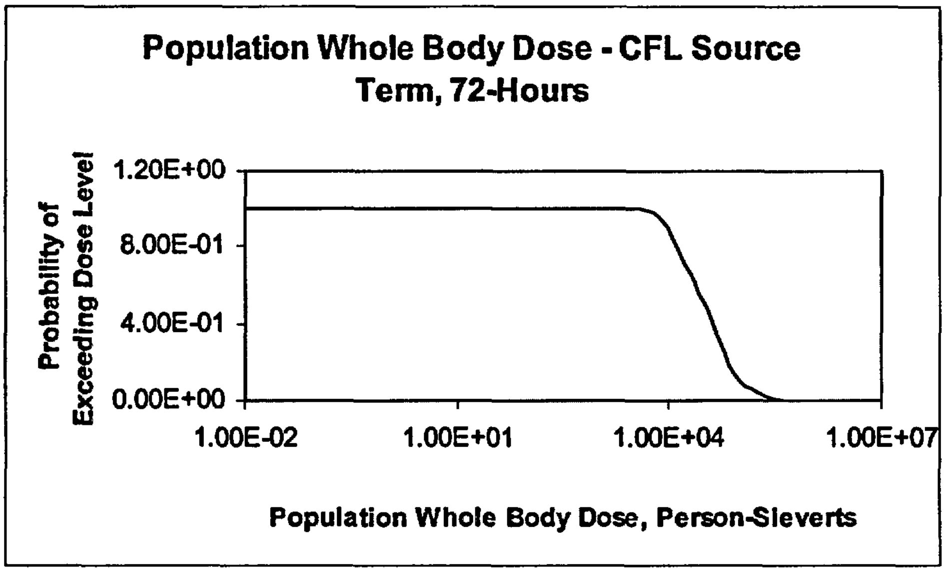


Figure 49-25

Population Whole Body Dose – CFL Source Term, 24 Hours



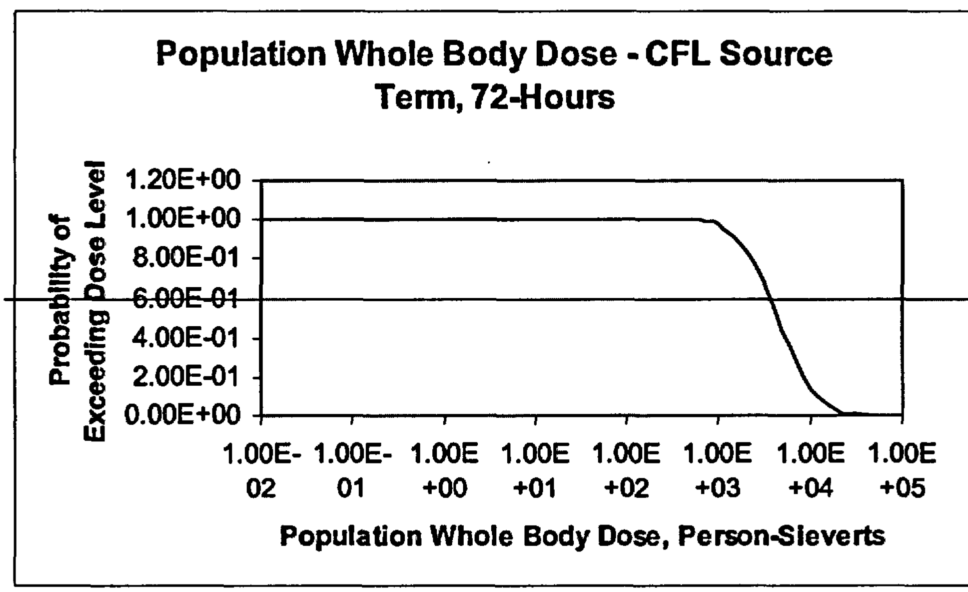
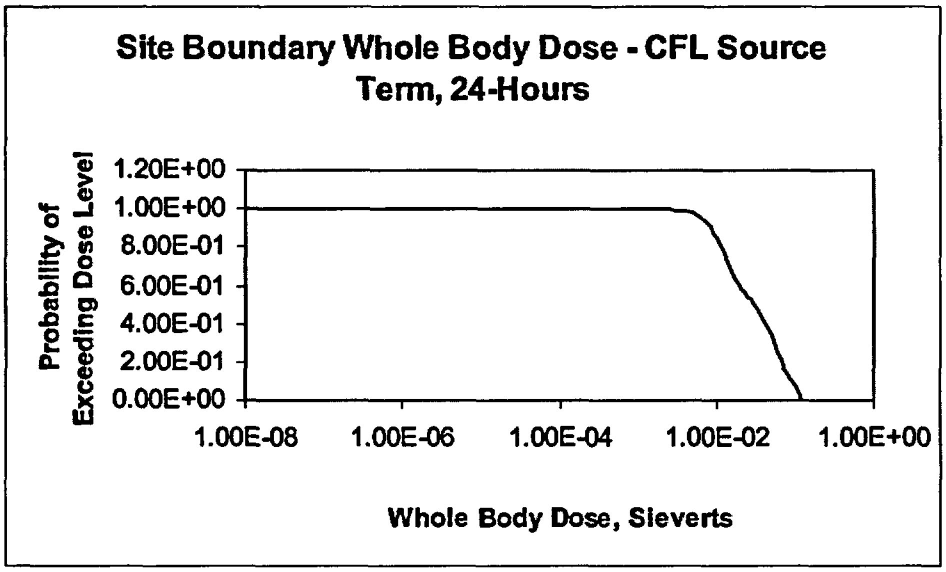


Figure 49-26

Population Whole Body Dose – CFL Source Term, 72 Hours



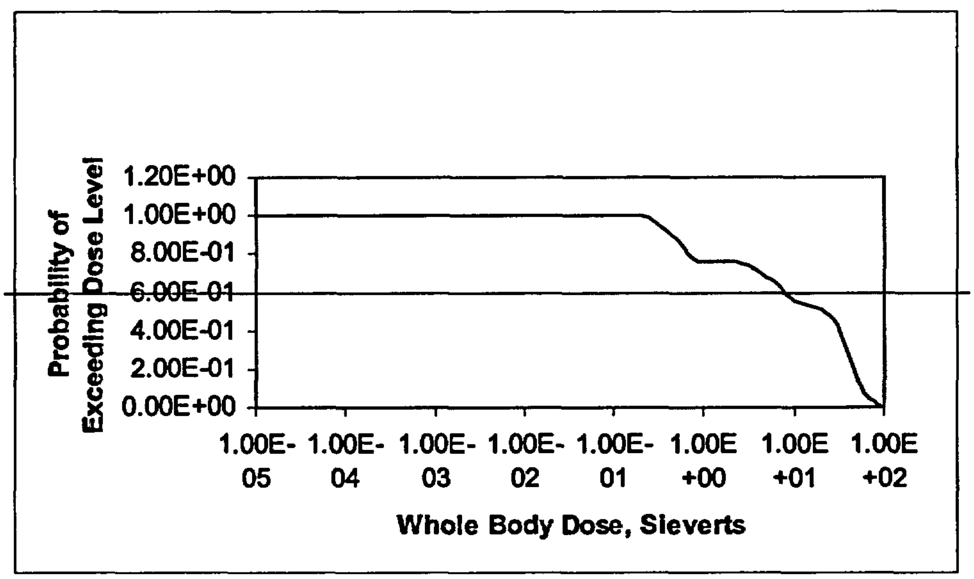
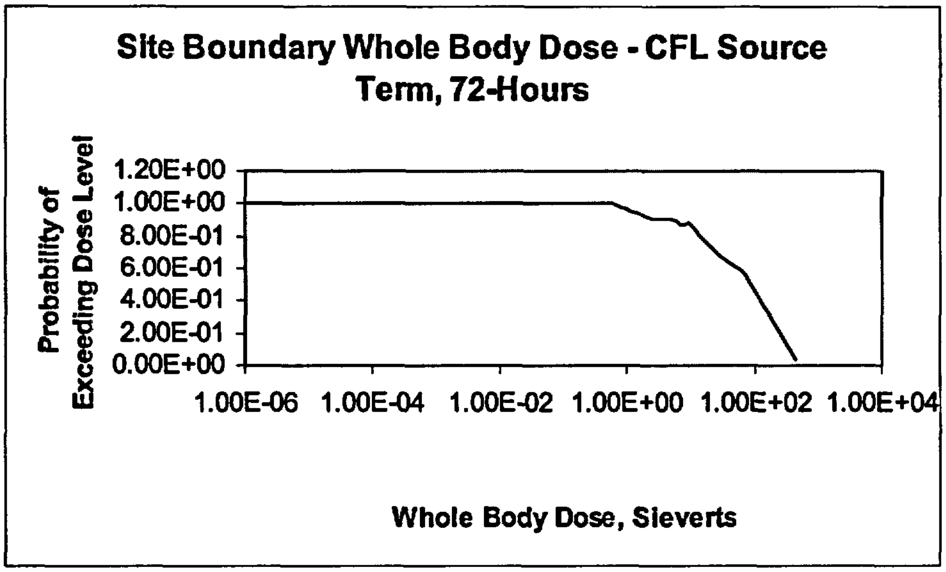


Figure 49-27

Site Boundary Whole Body Dose – CFL Source Term, 24 Hours



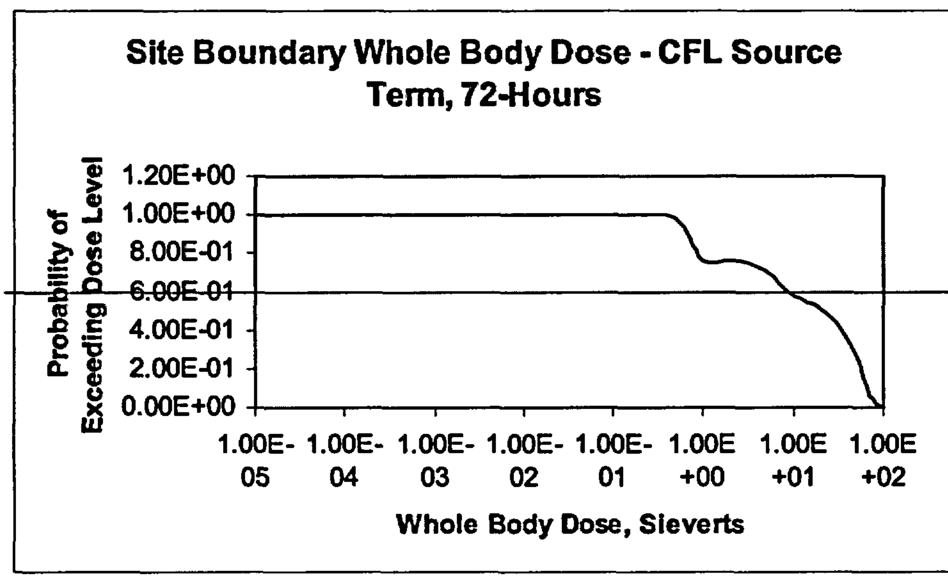
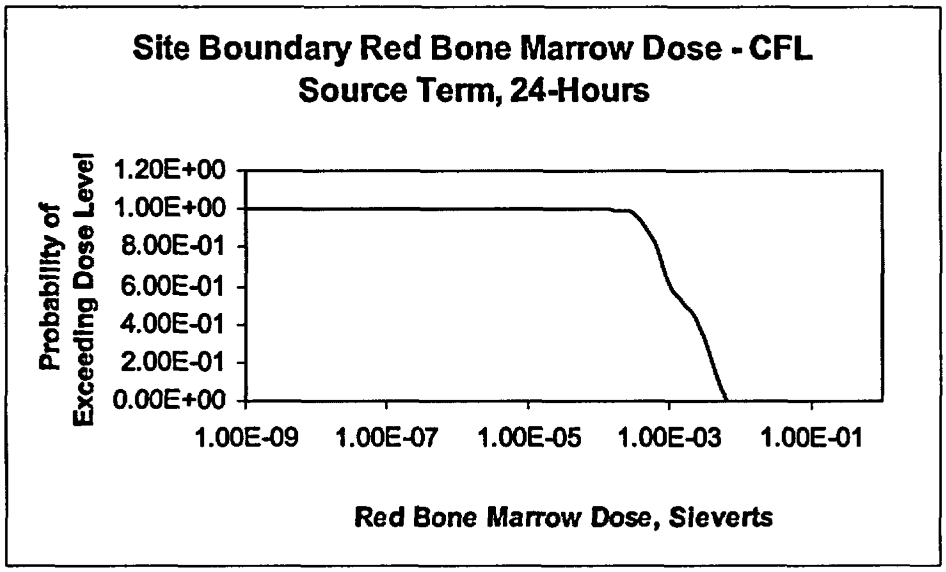


Figure 49-28

Site Boundary Whole Body Dose – CFL Source Term, 72 Hours



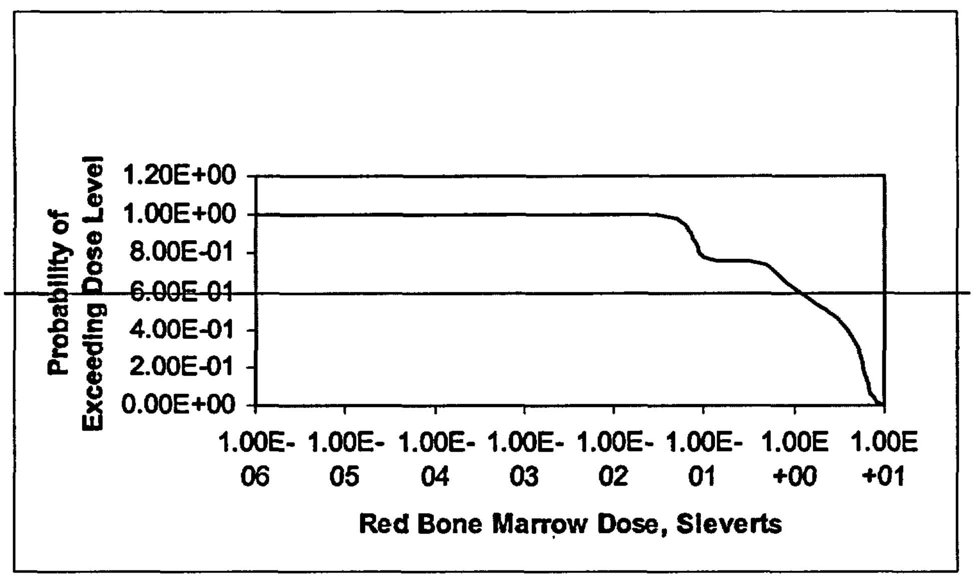
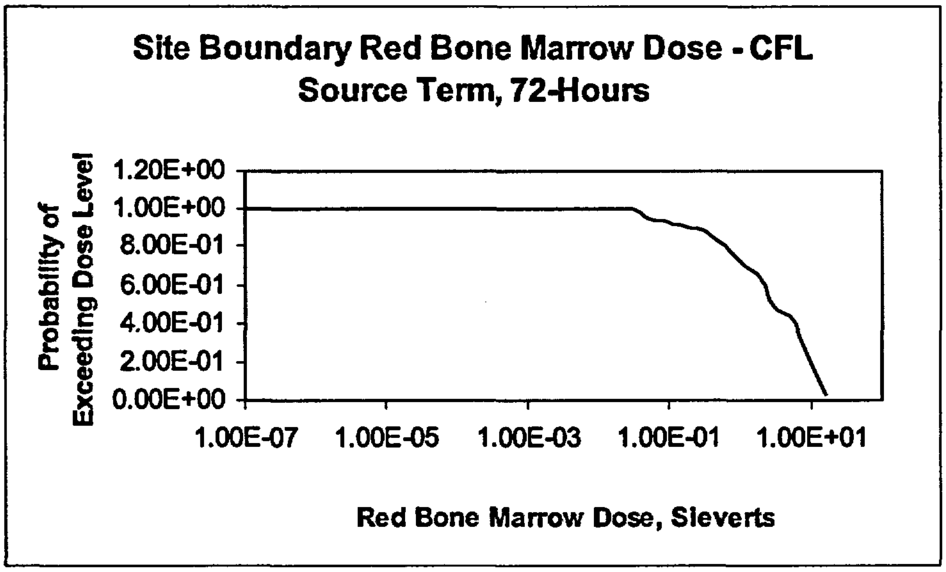


Figure 49-29

Site Boundary Red Bone Marrow Dose – CFL Source Term, 24 Hours



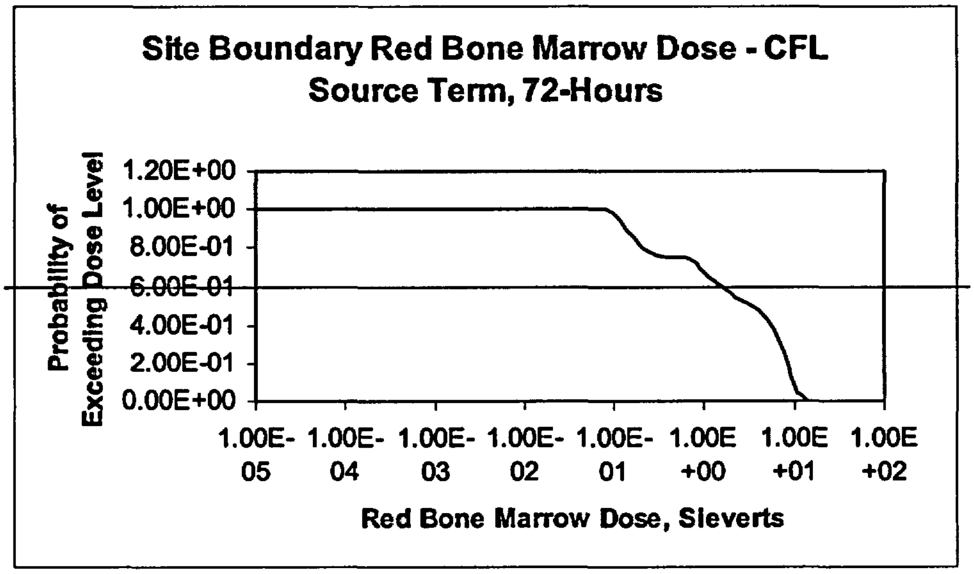
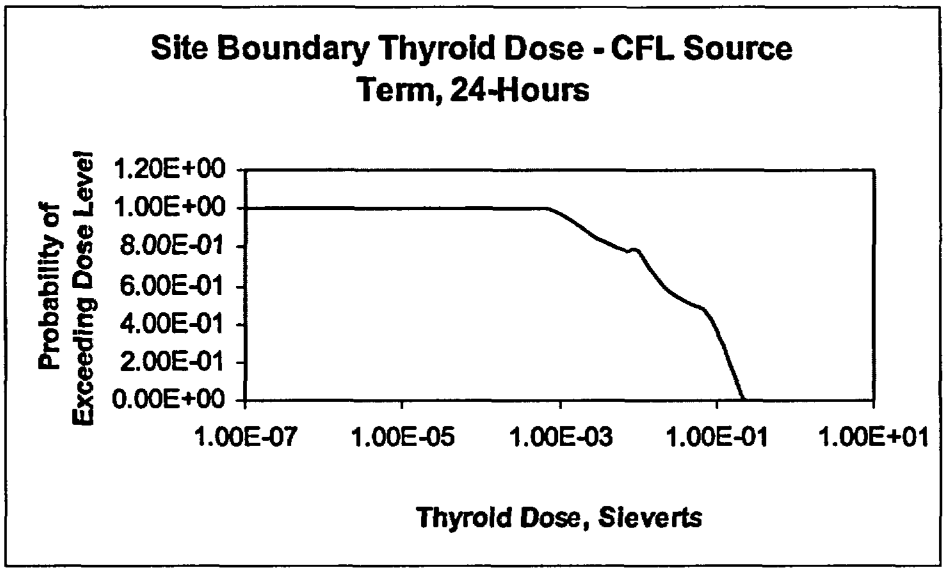


Figure 49-30

Site Boundary Red Bone Marrow Dose – CFL Source Term, 72 Hours



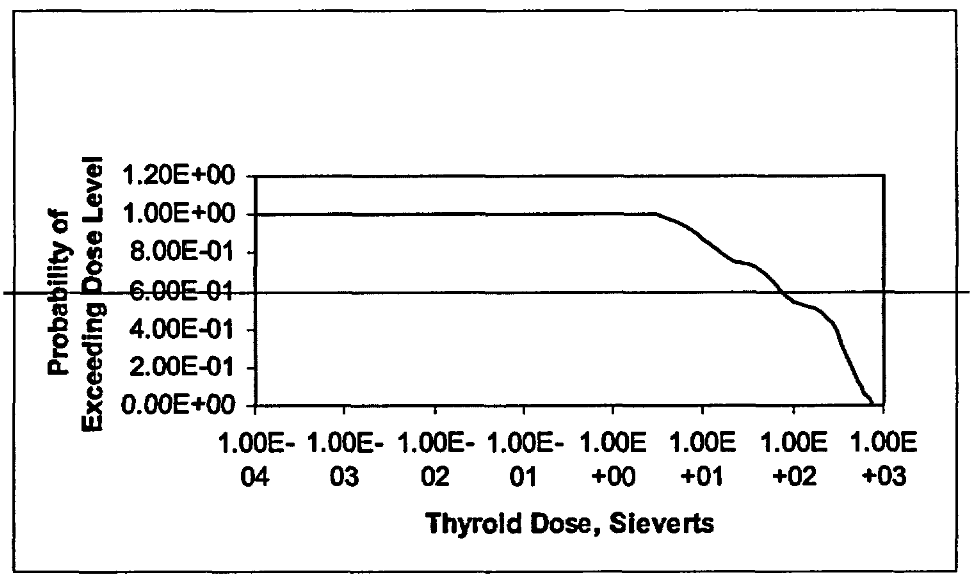
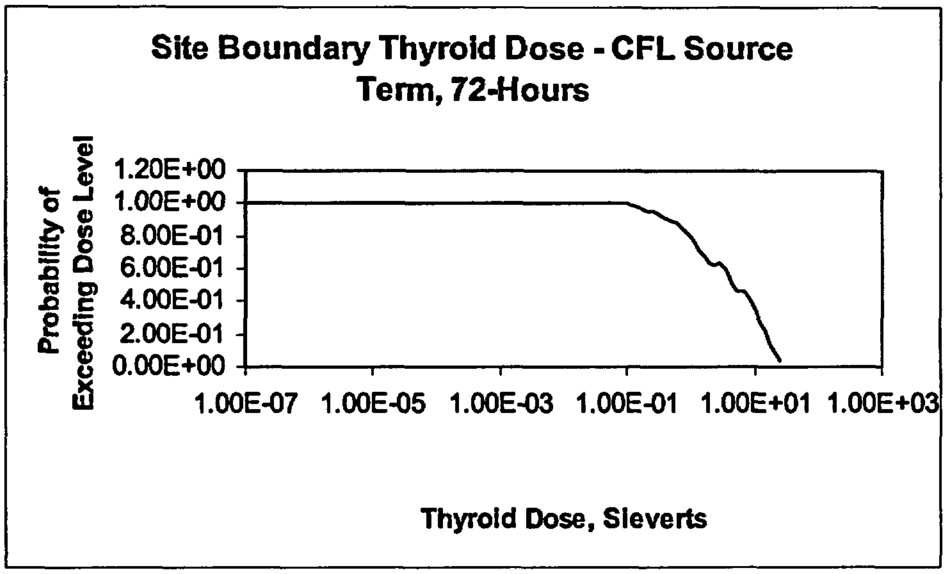


Figure 49-31

Site Boundary Thyroid Dose – CFL Source Term, 24 Hours



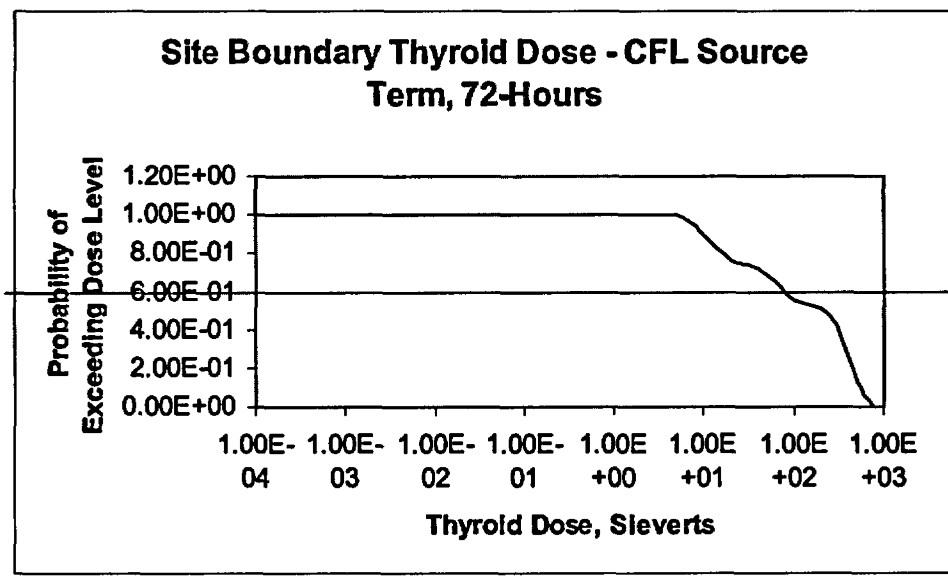
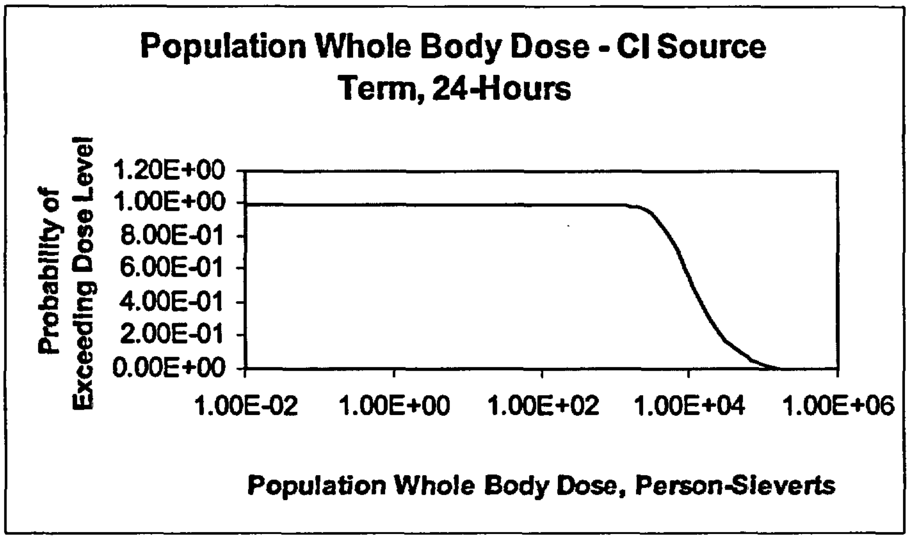


Figure 49-32

Site Boundary Thyroid Dose – CFL Source Term, 72 Hours



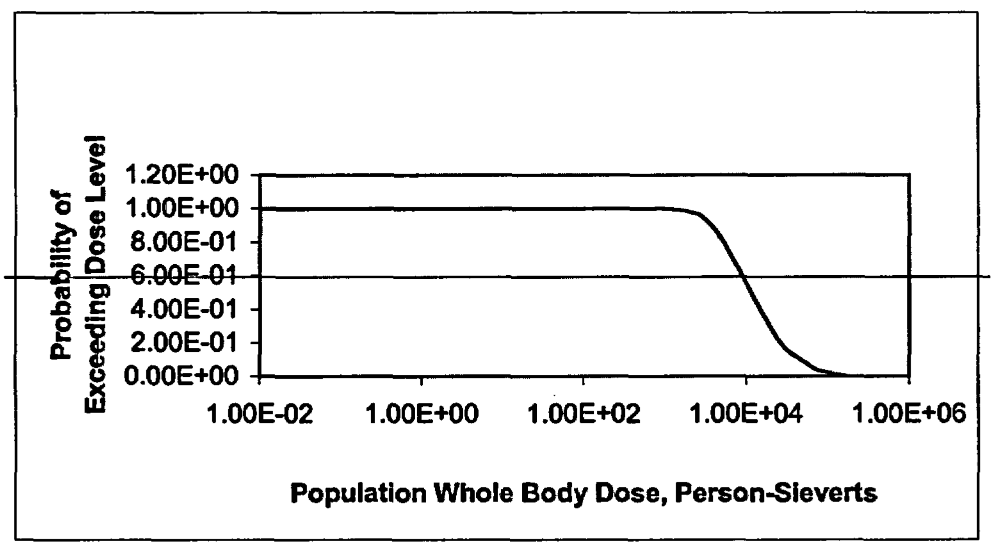
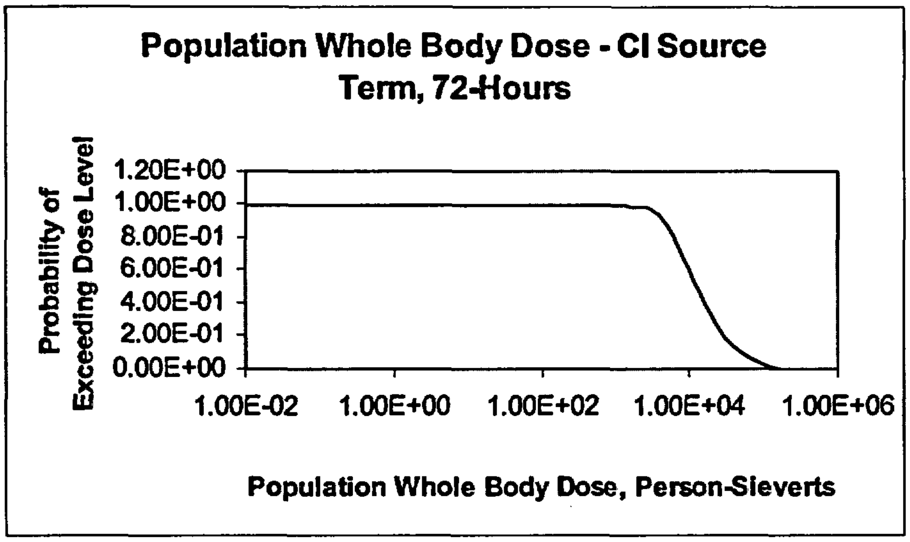


Figure 49-33

Population Whole Body Dose – CI Source Term, 24 Hours



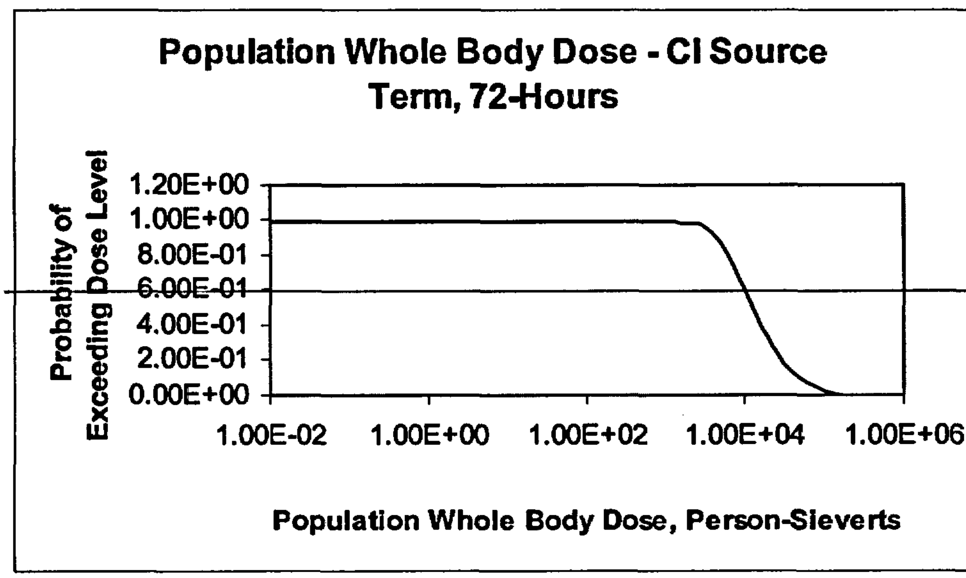
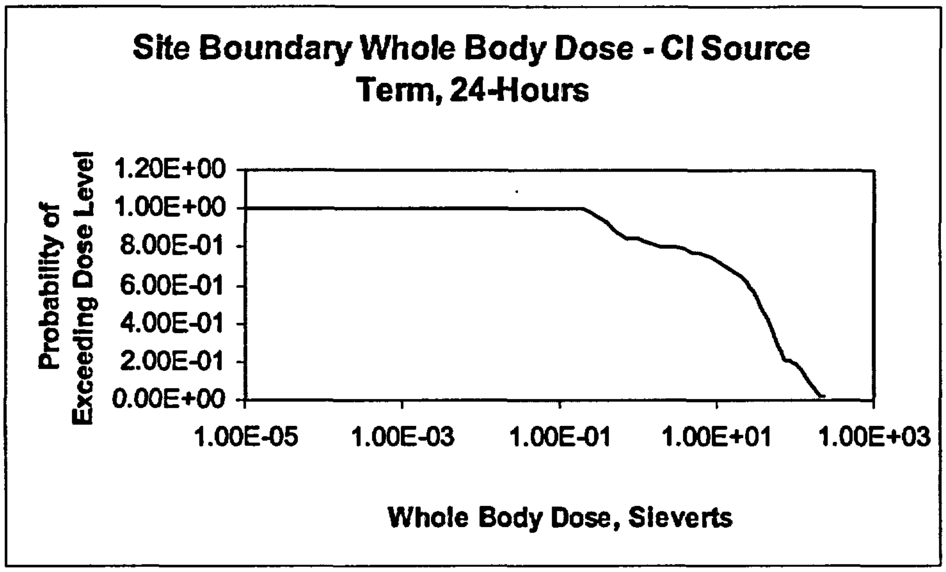


Figure 49-34

Population Whole Body Dose – CI Source Term, 72 Hours



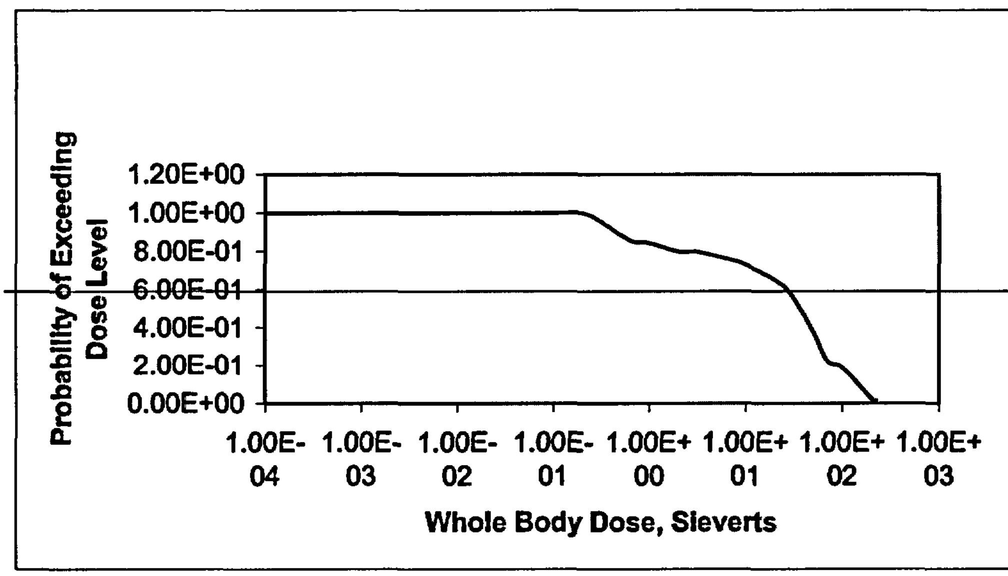
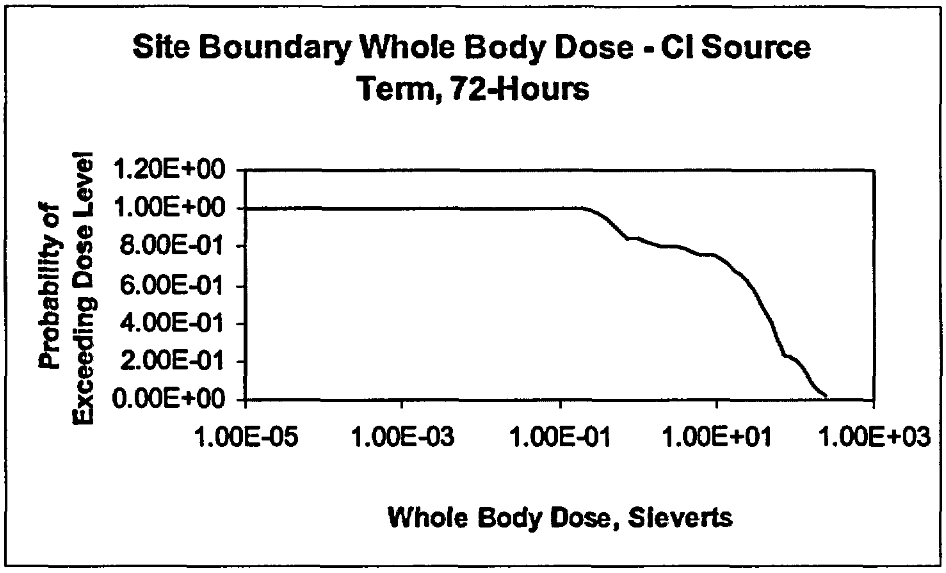


Figure 49-35

Site Boundary Whole Body Dose – CI Source Term, 24 Hours



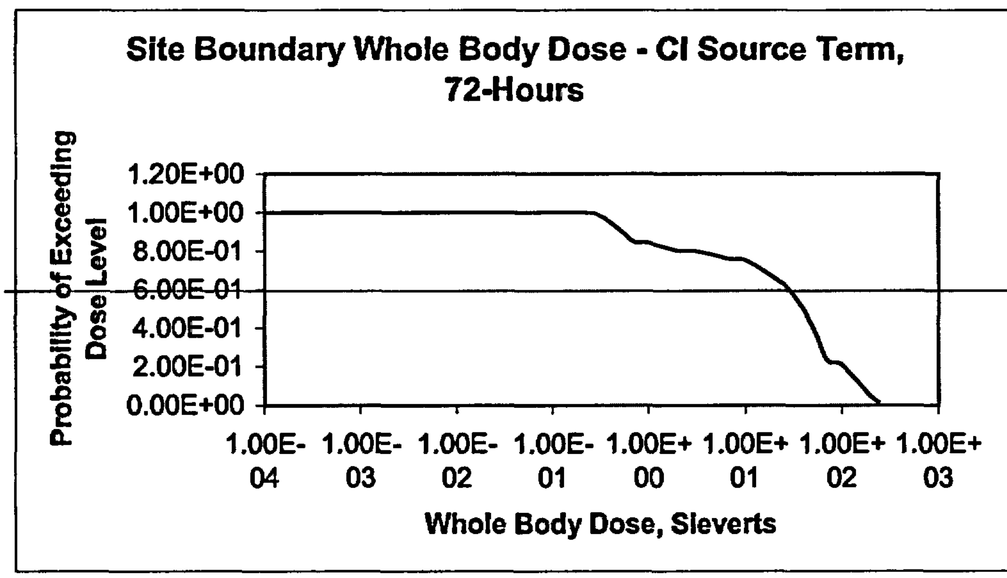
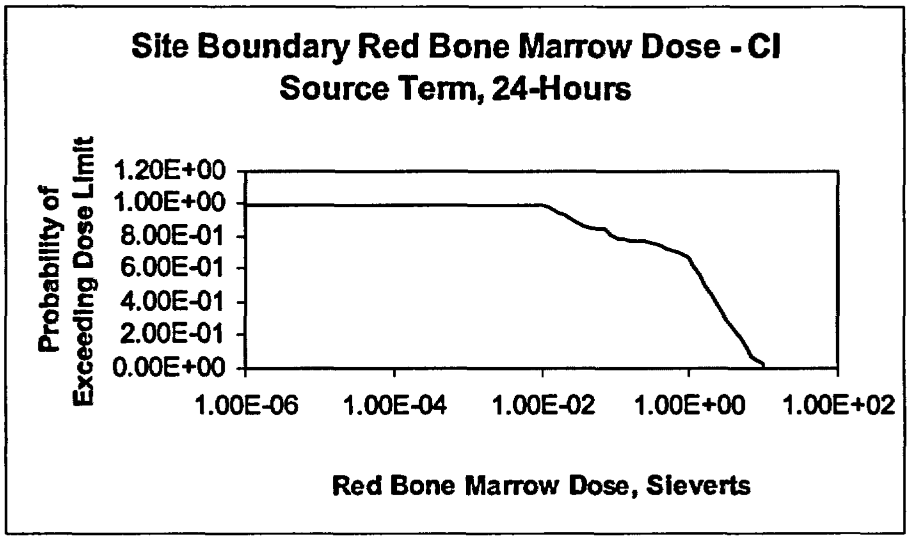


Figure 49-36

Site Boundary Whole Body Dose – CI Source Term, 72 Hours



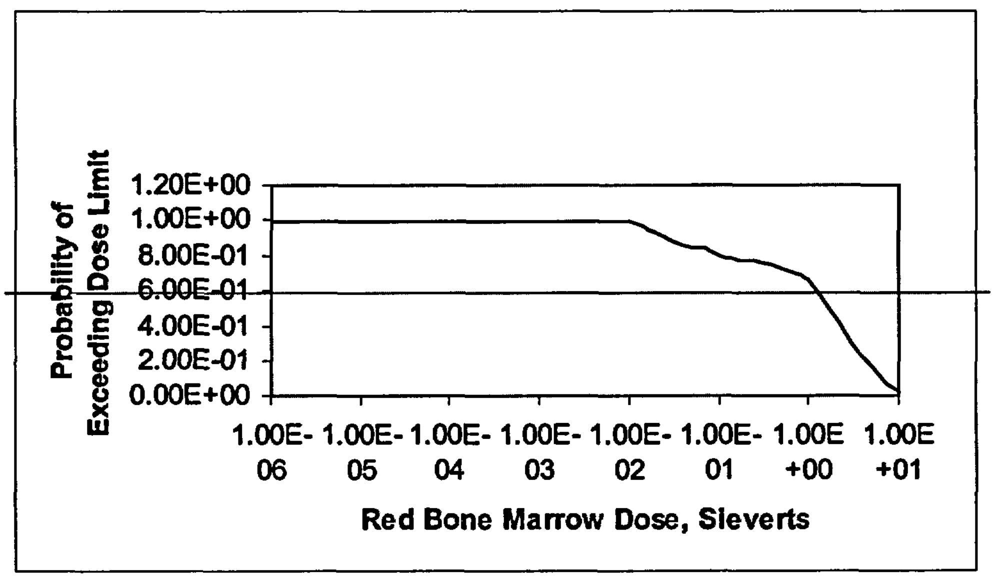
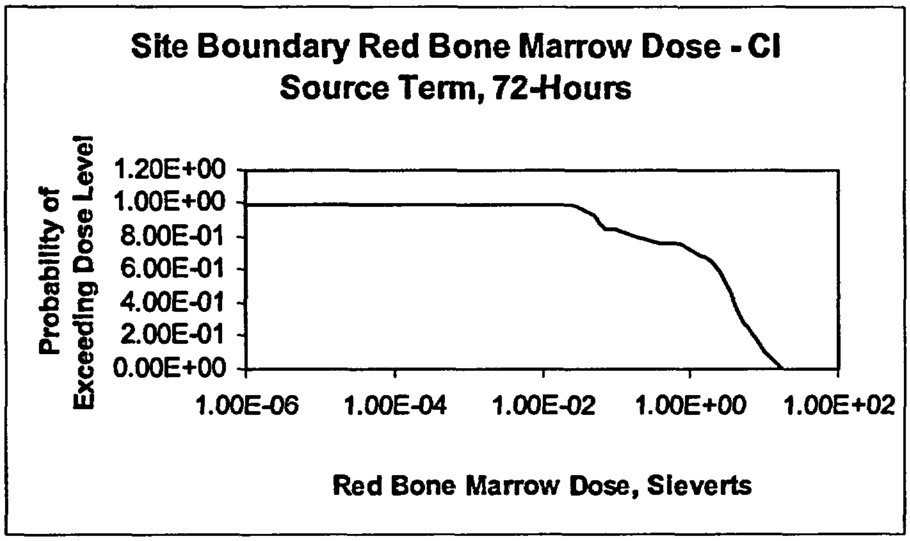


Figure 49-37

Site Boundary Red Bone Marrow Dose – CI Source Term, 24 Hours



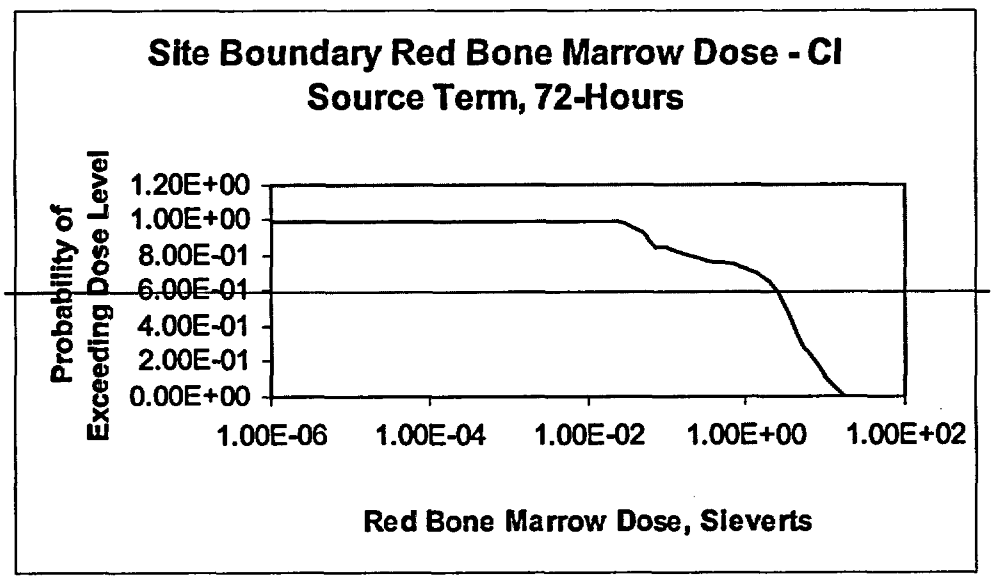
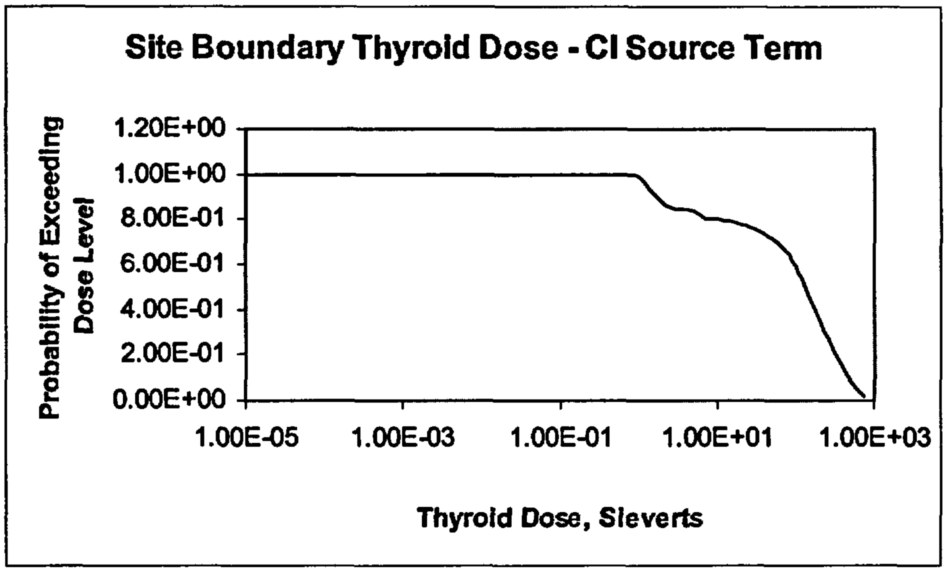


Figure 49-38

Site Boundary Red Bone Marrow Dose – CI Source Term, 72 Hours



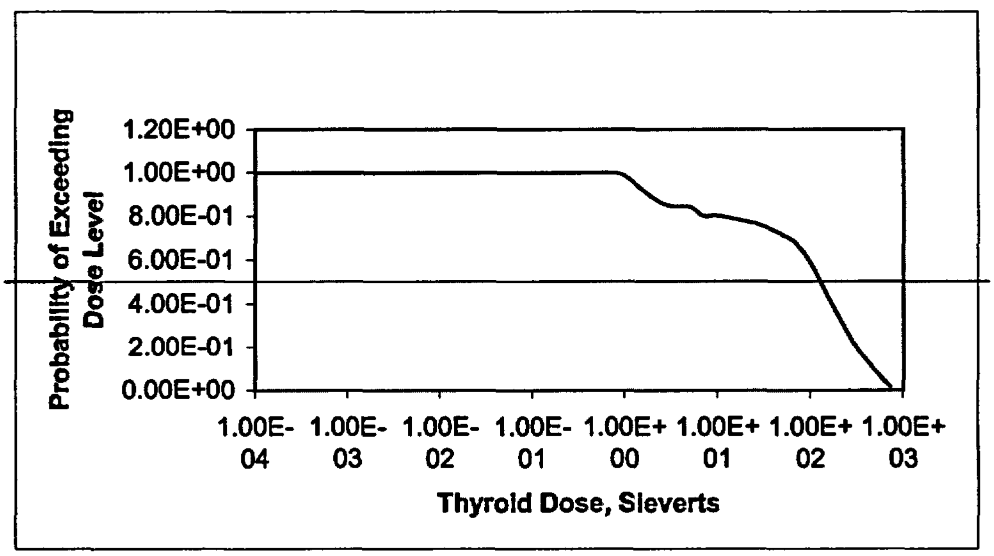
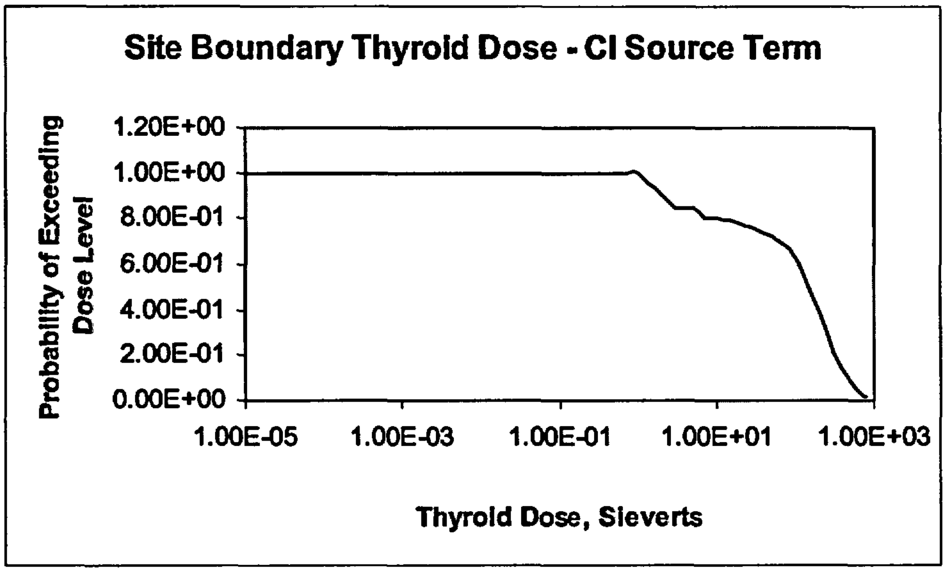


Figure 49-39

Site Boundary Thyroid Dose – CI Source Term, 24 Hours



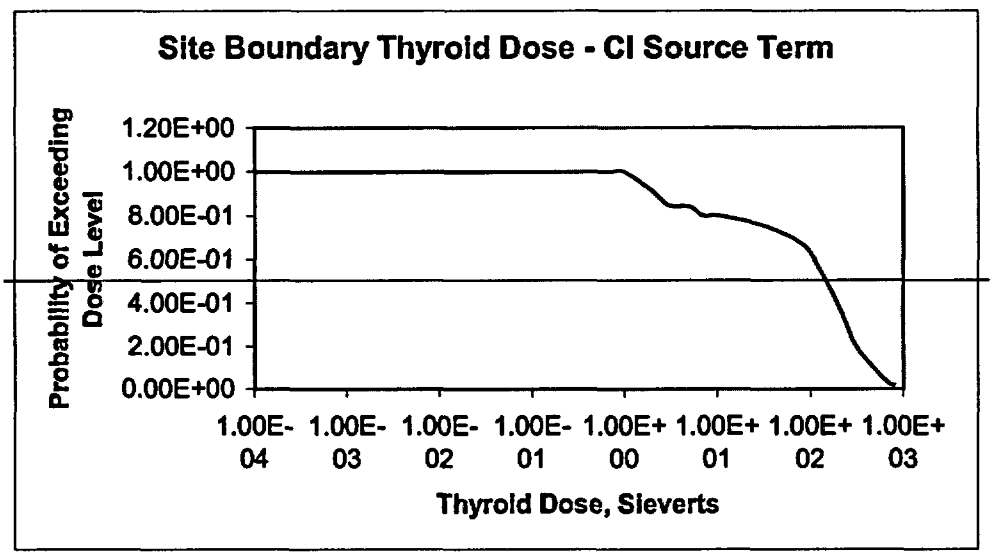
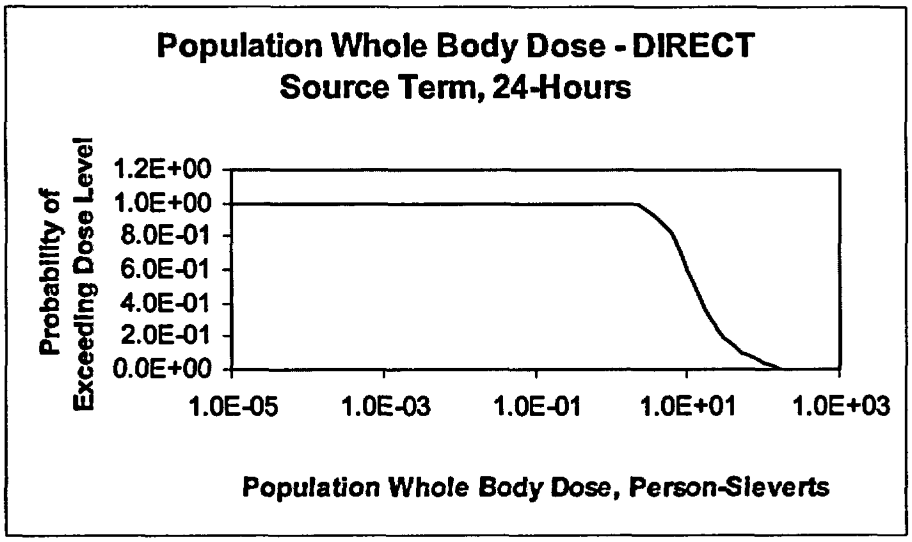


Figure 49-40

Site Boundary Thyroid Dose – CI Source Term, 72 Hours



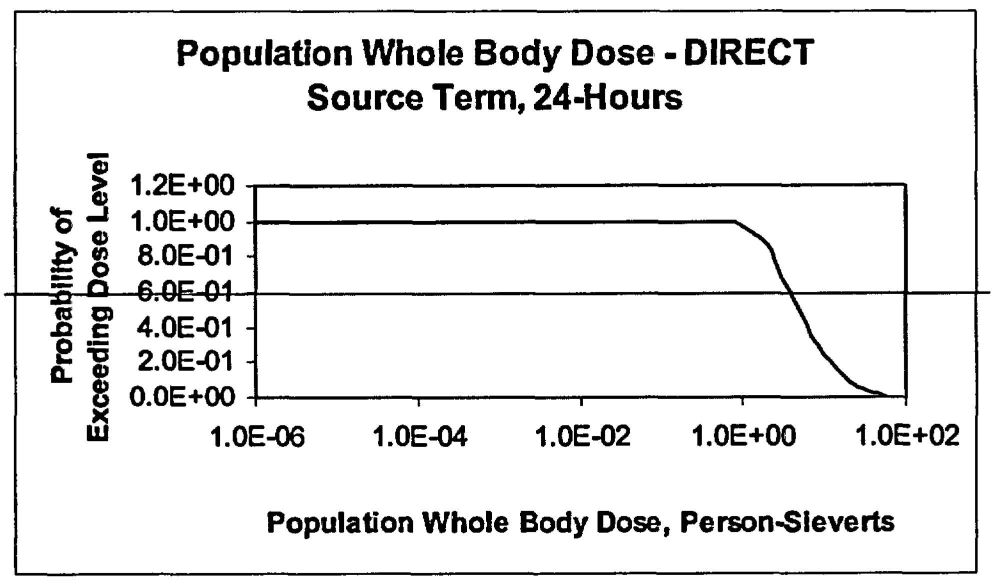
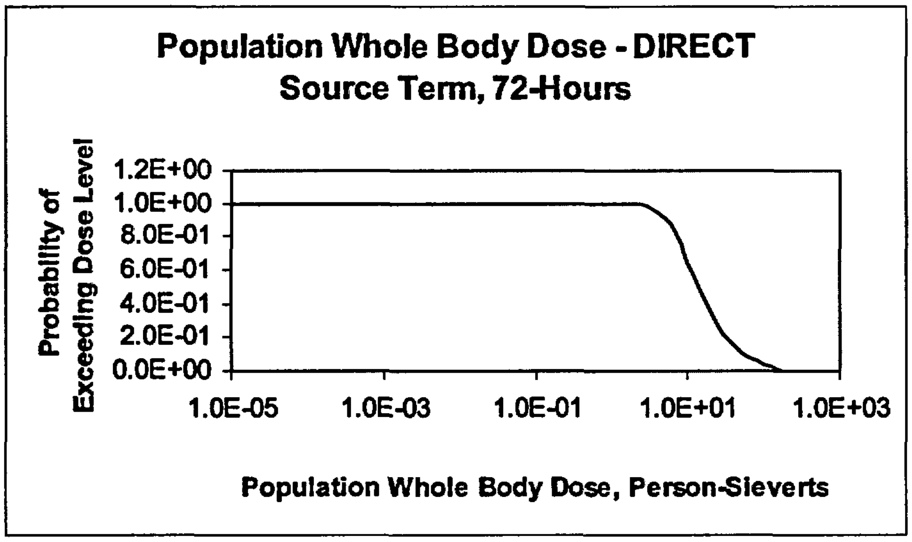


Figure 49-41

Population Whole Body Dose – DIRECT Source Term, 24 Hours



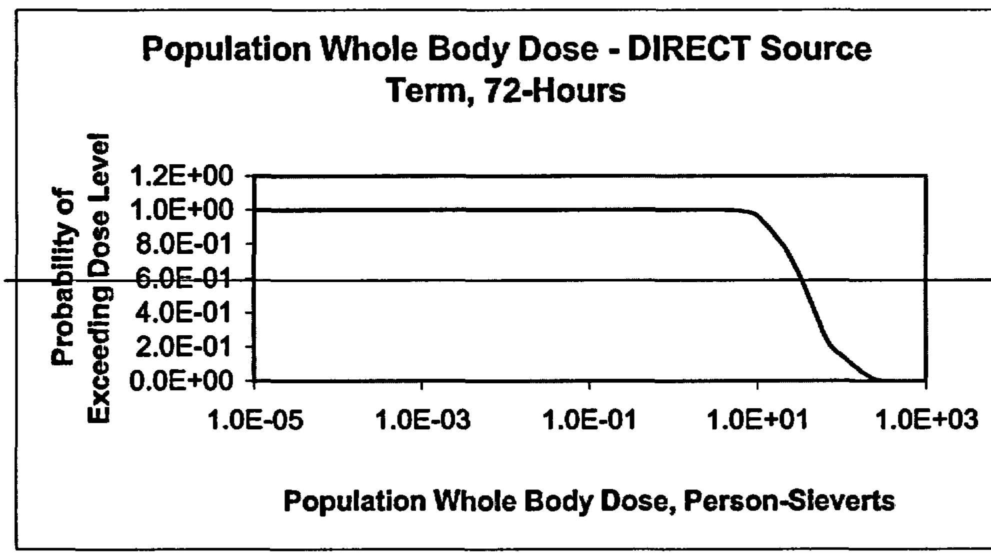
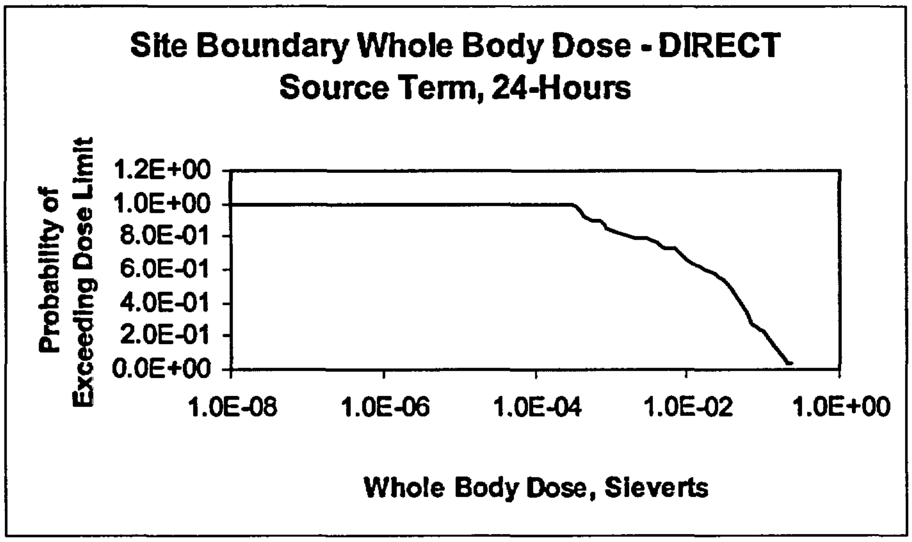


Figure 49-42

Population Whole Body Dose – DIRECT Source Term, 72 Hours



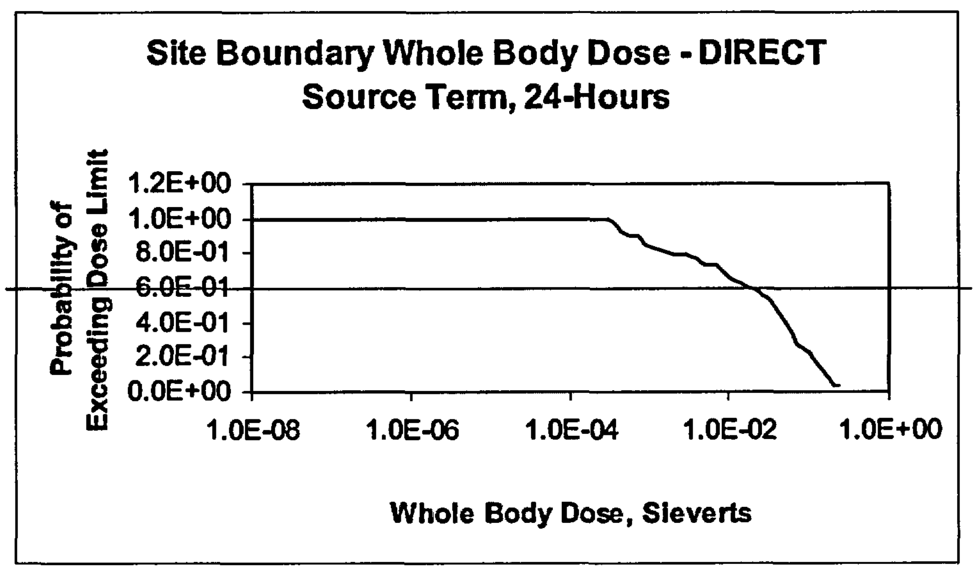
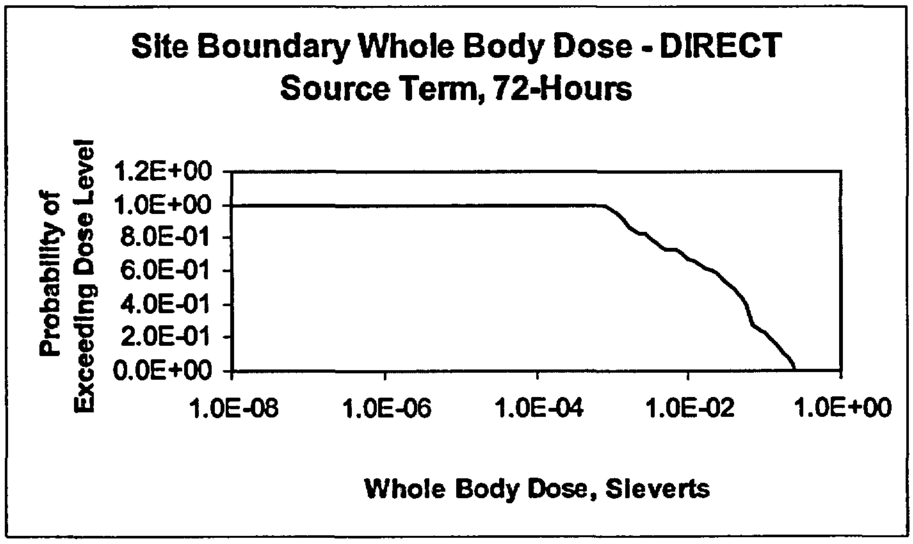


Figure 49-43

Site Boundary Whole Body Dose – DIRECT Source Term, 24 Hours



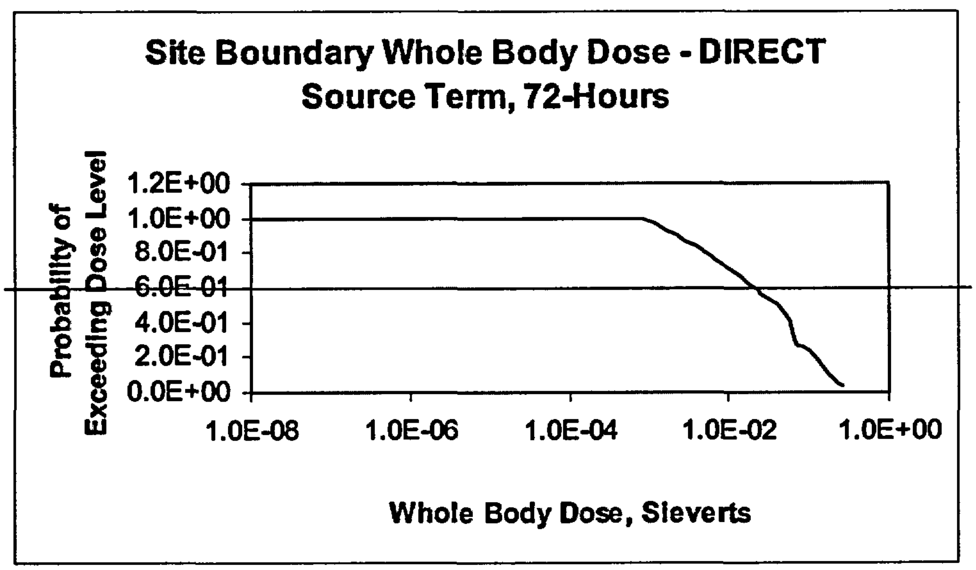
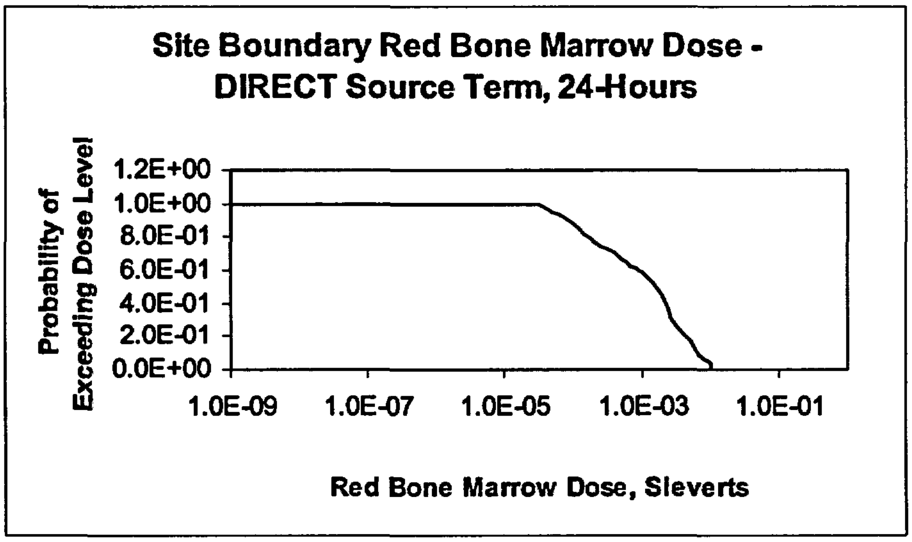


Figure 49-44

Site Boundary Whole Body Dose – DIRECT Source Term, 72 Hours



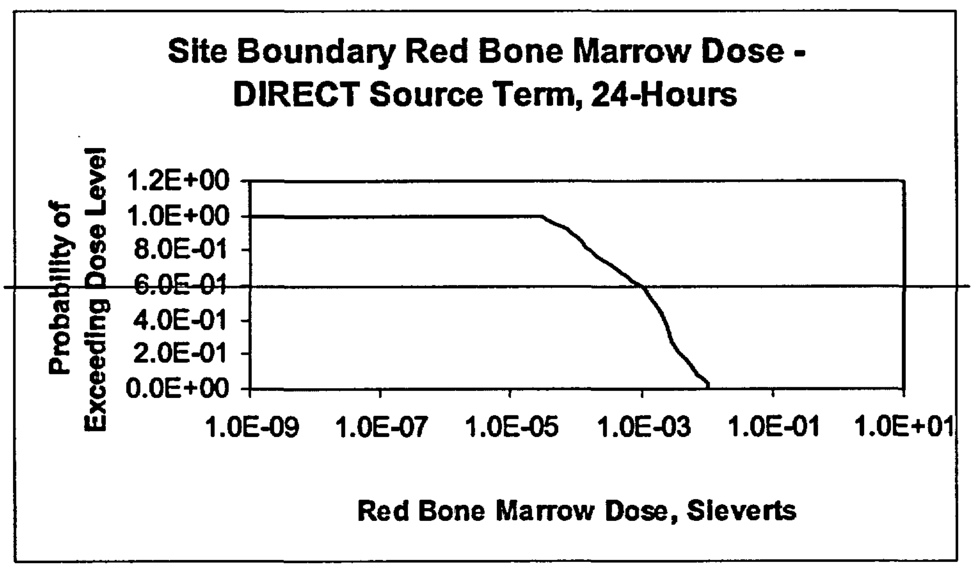
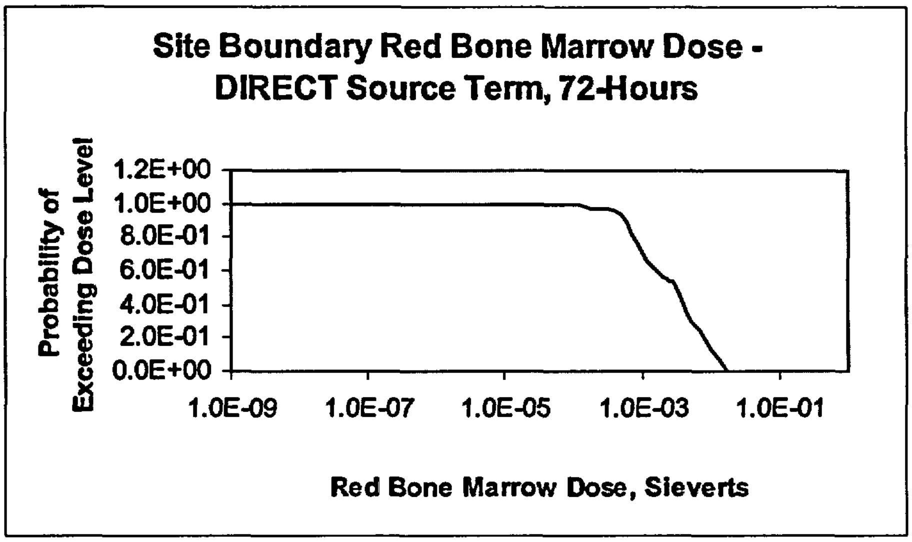


Figure 49-45

Site Boundary Red Bone Marrow Dose – DIRECT Source Term, 24 Hours



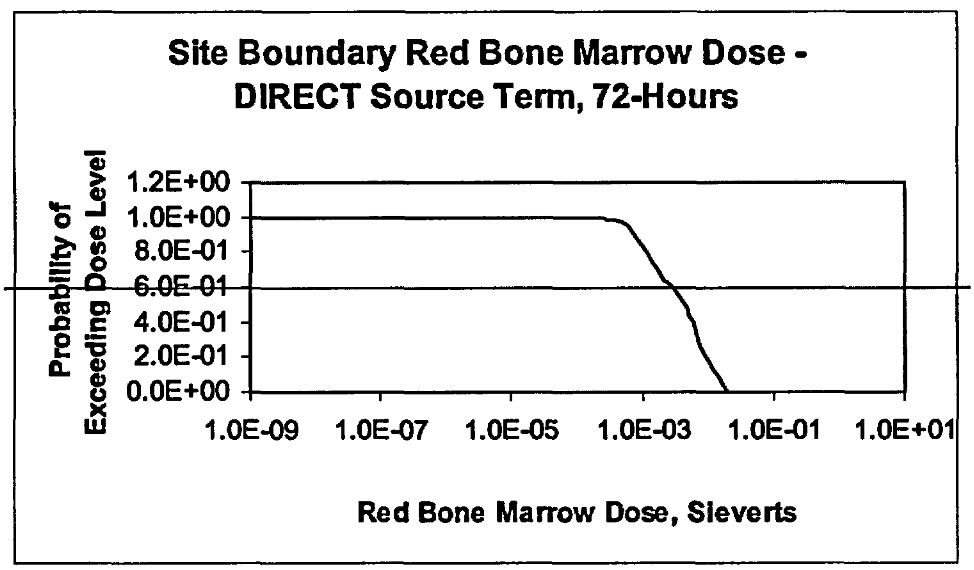
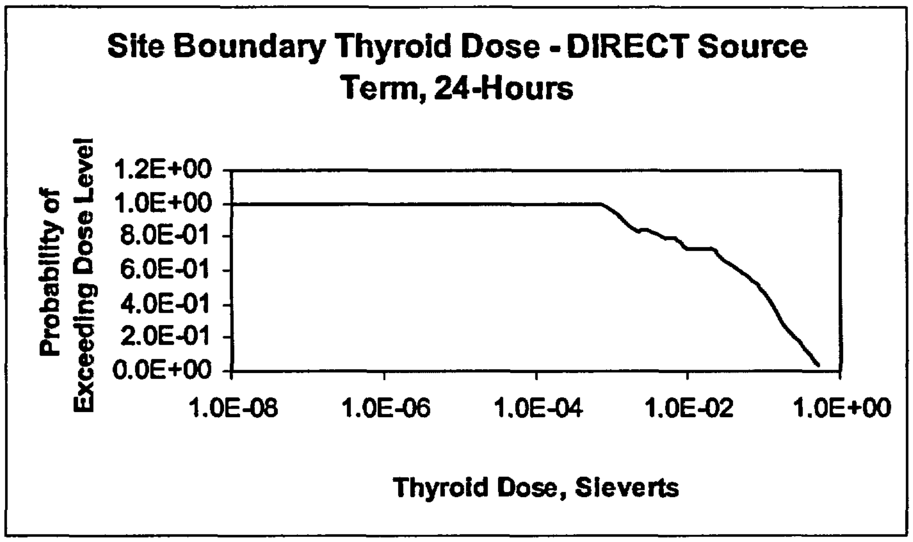


Figure 49-46

Site Boundary Red Bone Marrow Dose – DIRECT Source Term, 72 Hours



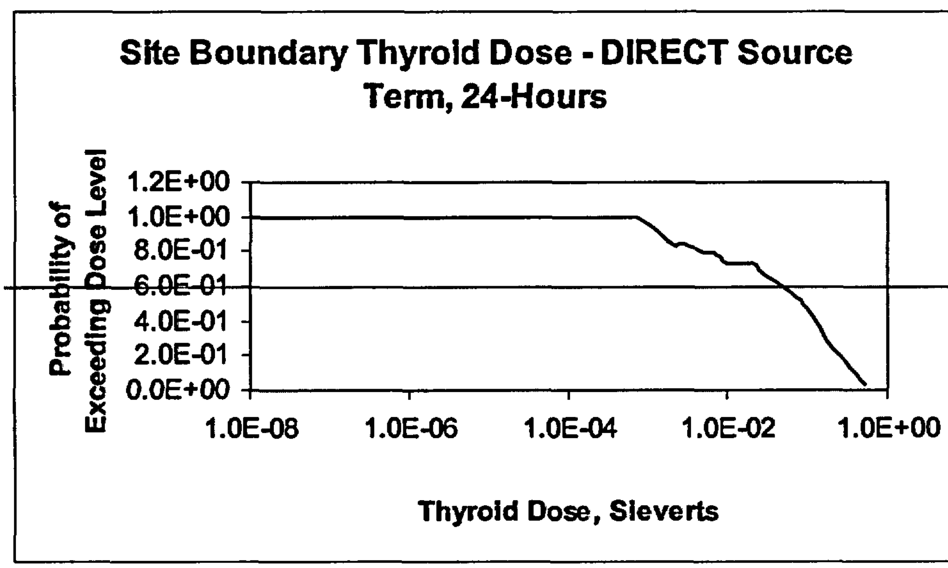
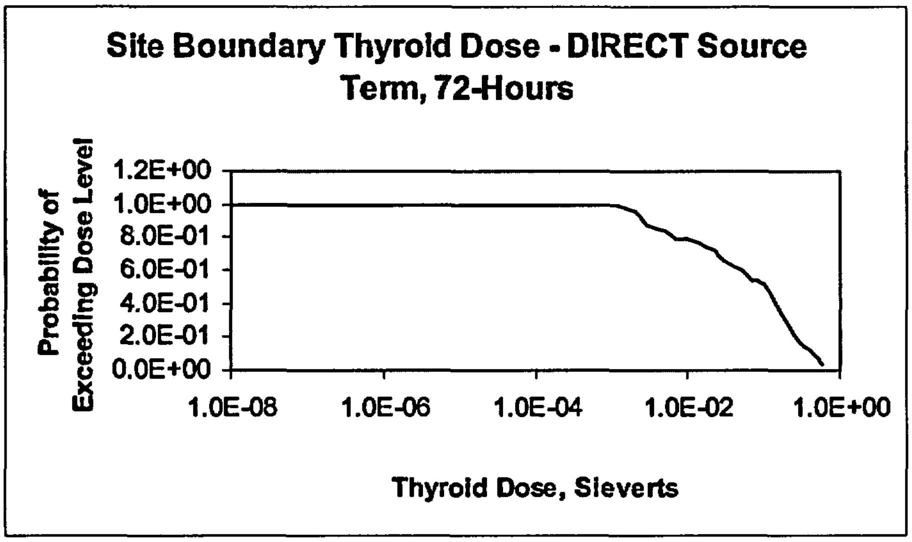


Figure 49-47

Site Boundary Thyroid Dose – DIRECT Source Term, 24 Hours



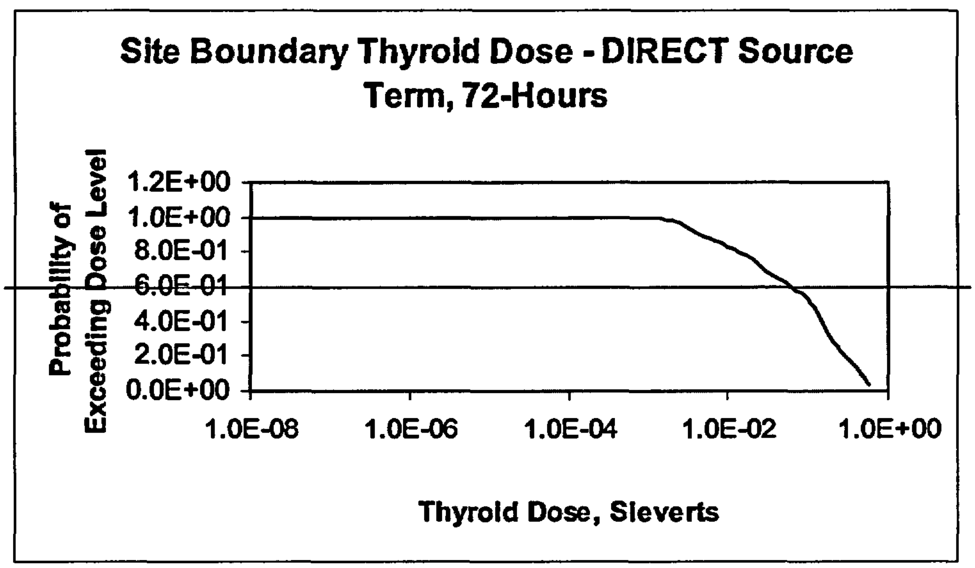
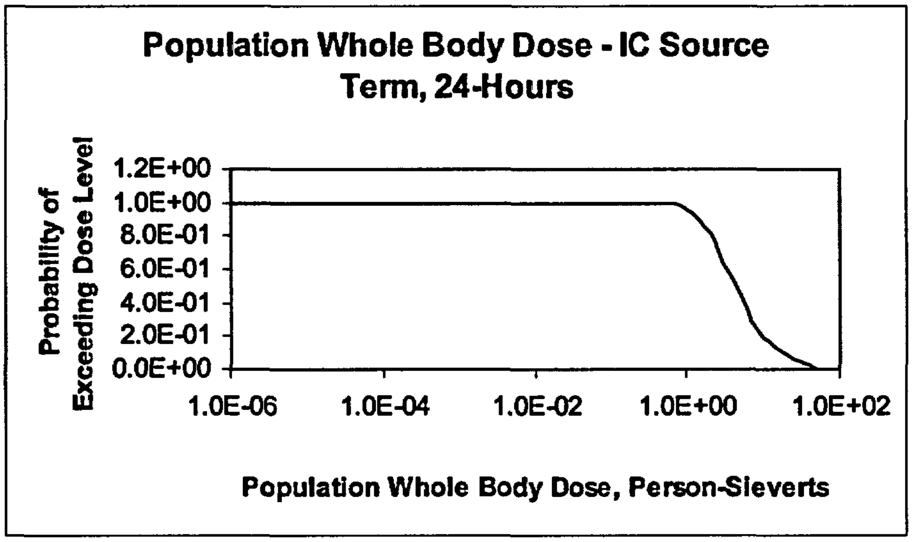


Figure 49-48

Site Boundary Thyroid Dose – DIRECT Source Term, 72 Hours



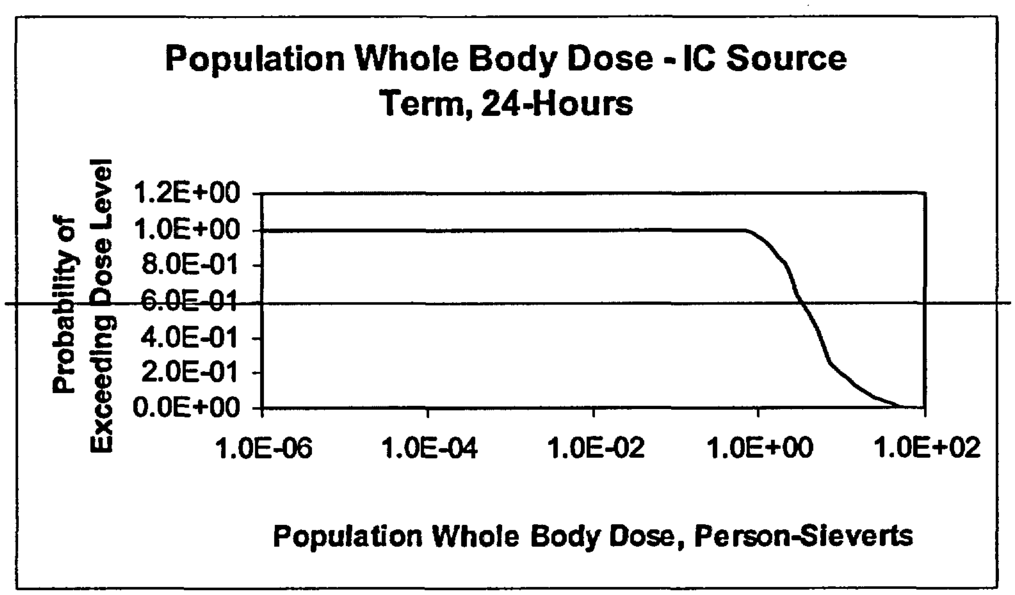
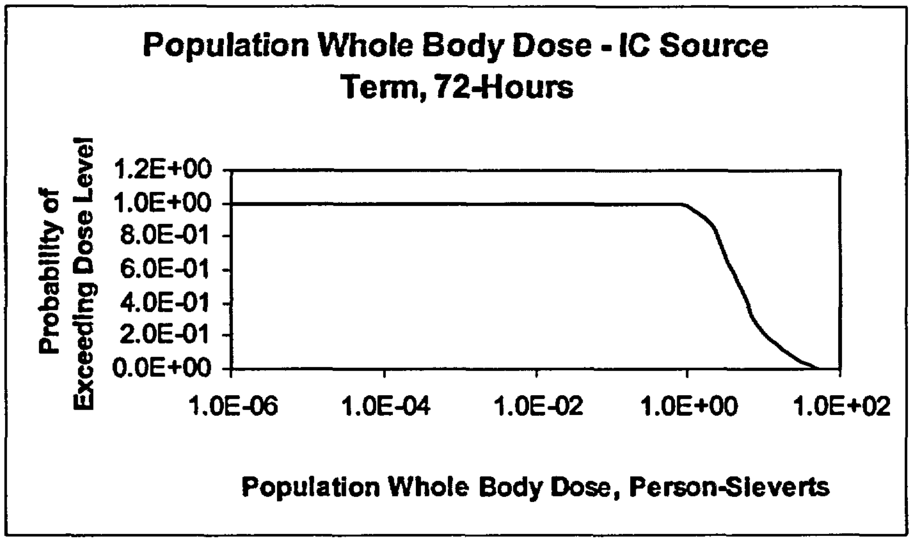


Figure 49-49

Population Whole Body Dose – IC Source Term, 24 Hours



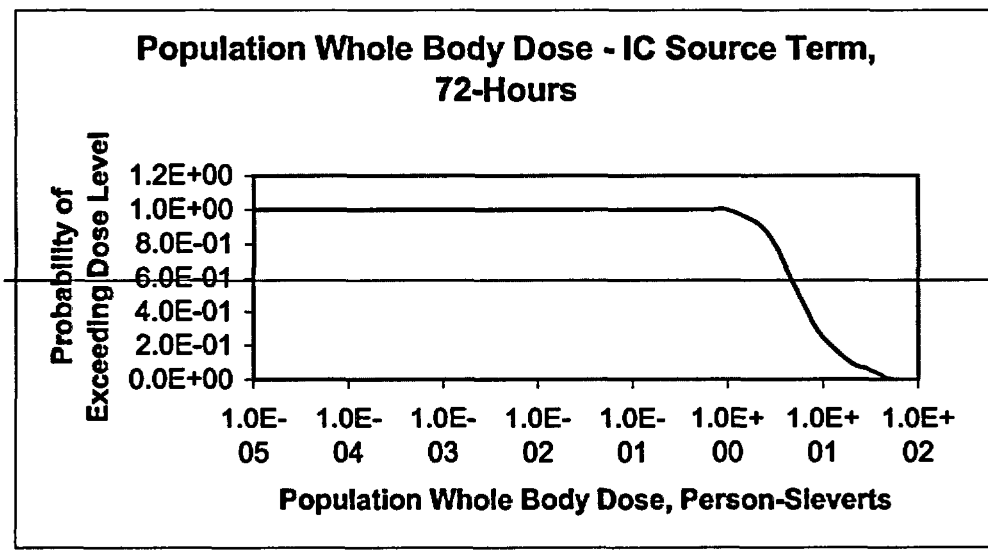
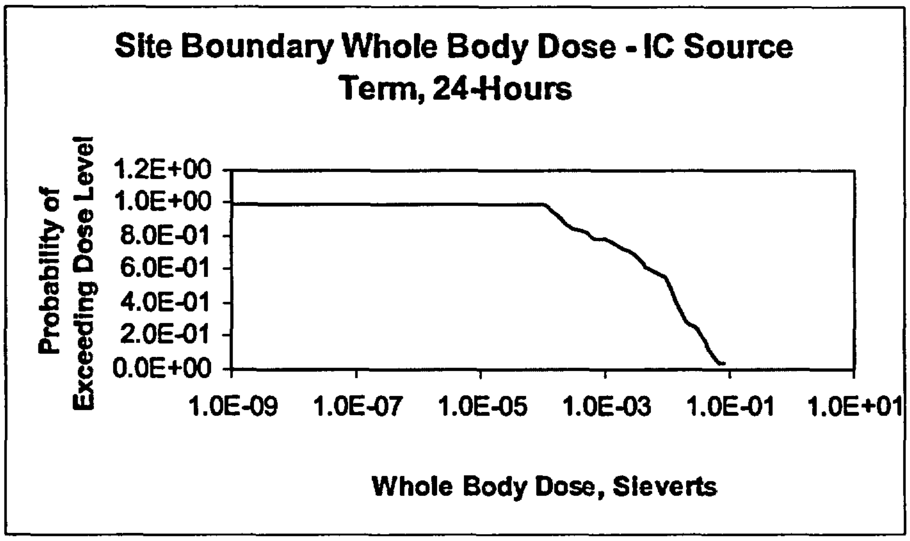


Figure 49-50

Population Whole Body Dose – IC Source Term, 72 Hours



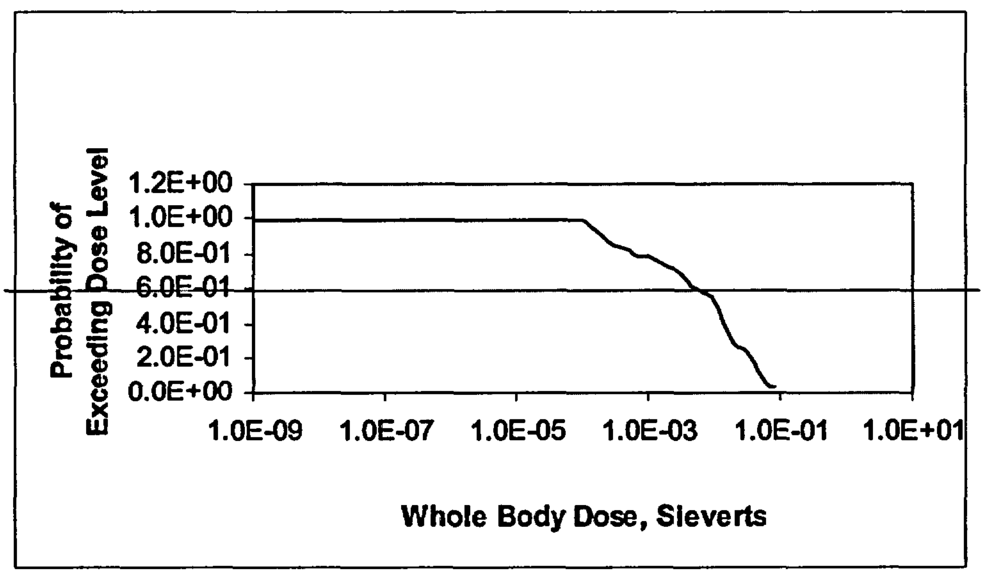
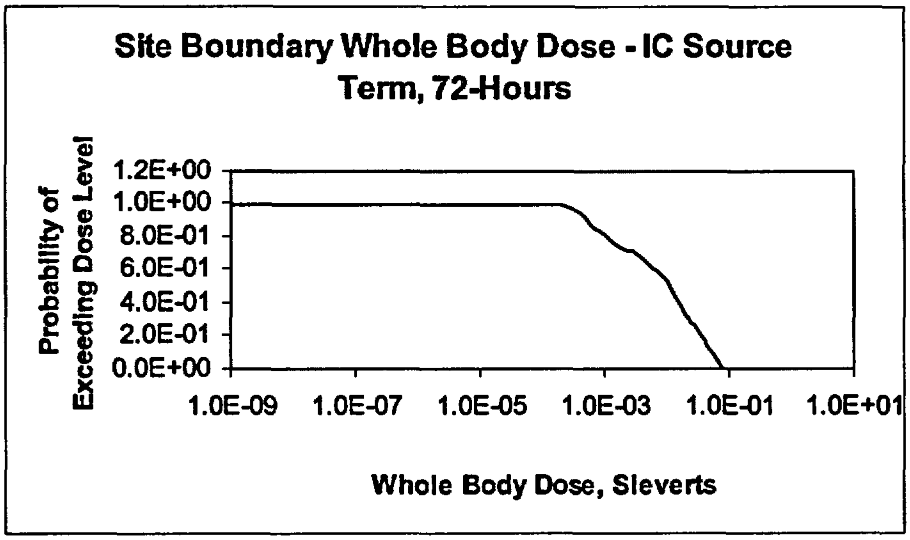


Figure 49-51

Site Boundary Whole Body Dose – IC Source Term, 24 Hours



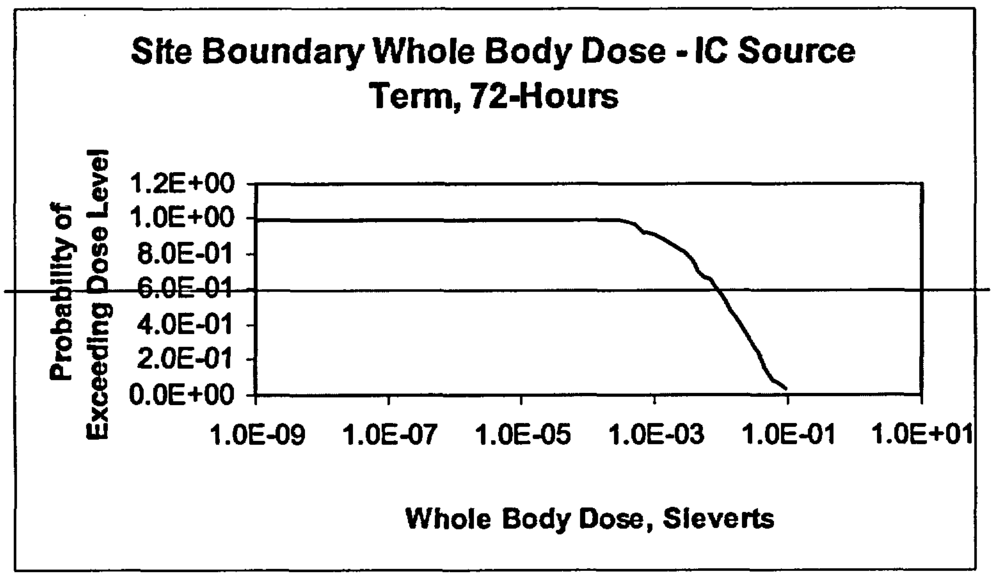
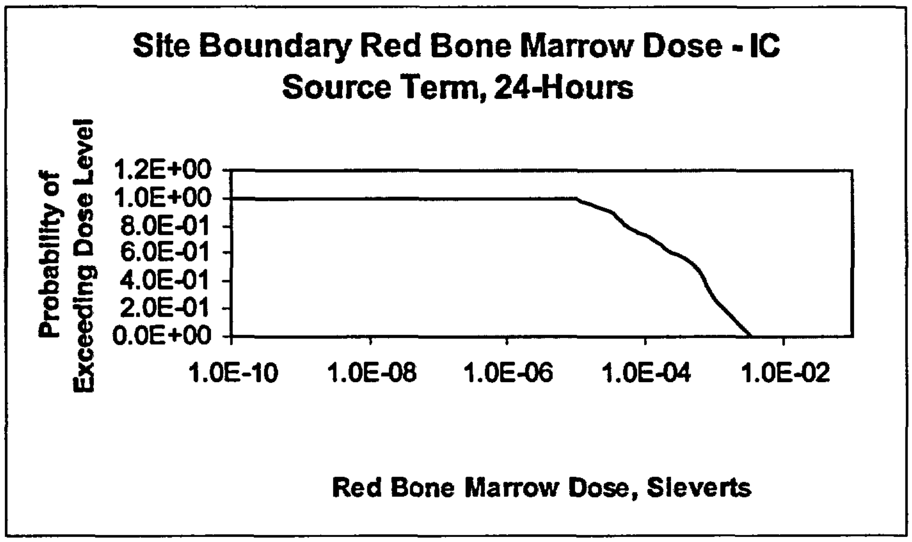


Figure 49-52

Site Boundary Whole Body Dose – IC Source Term, 72 Hours



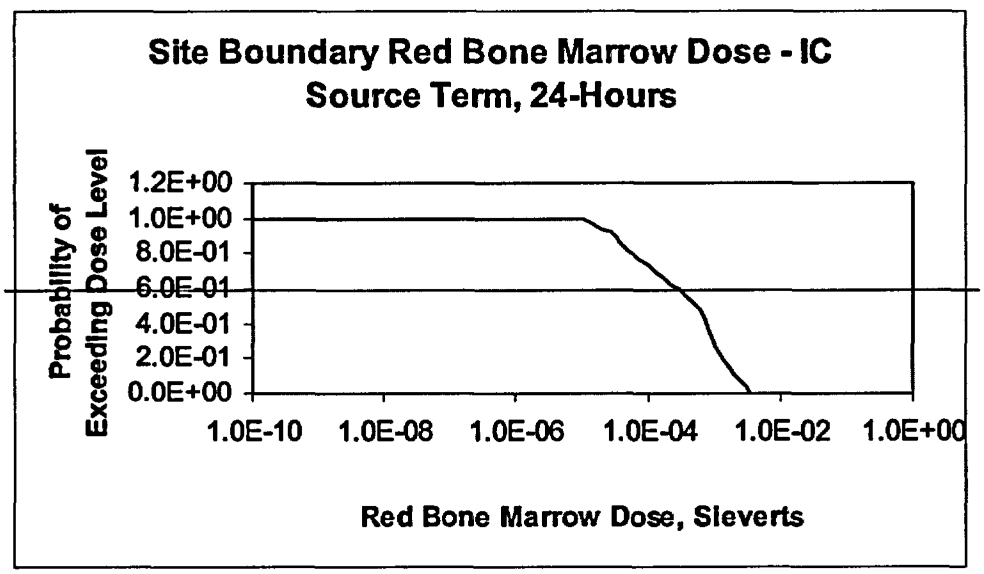
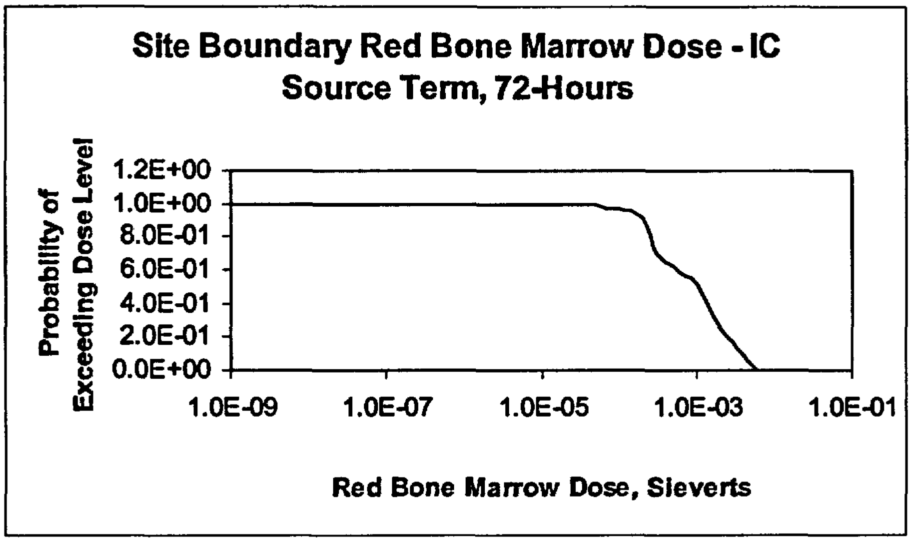


Figure 49-53

Site Boundary Red Bone Marrow Dose – IC Source Term, 24 Hours



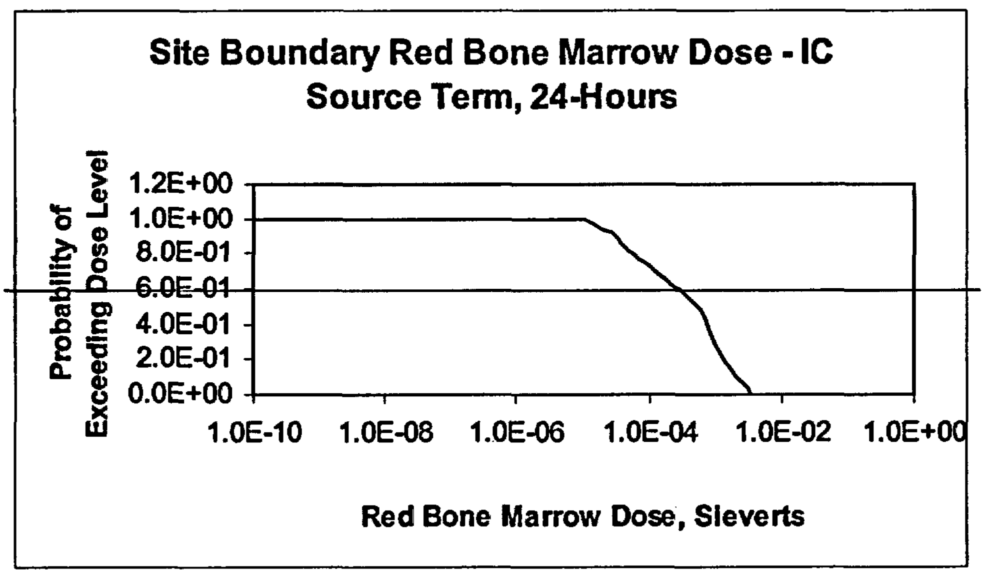
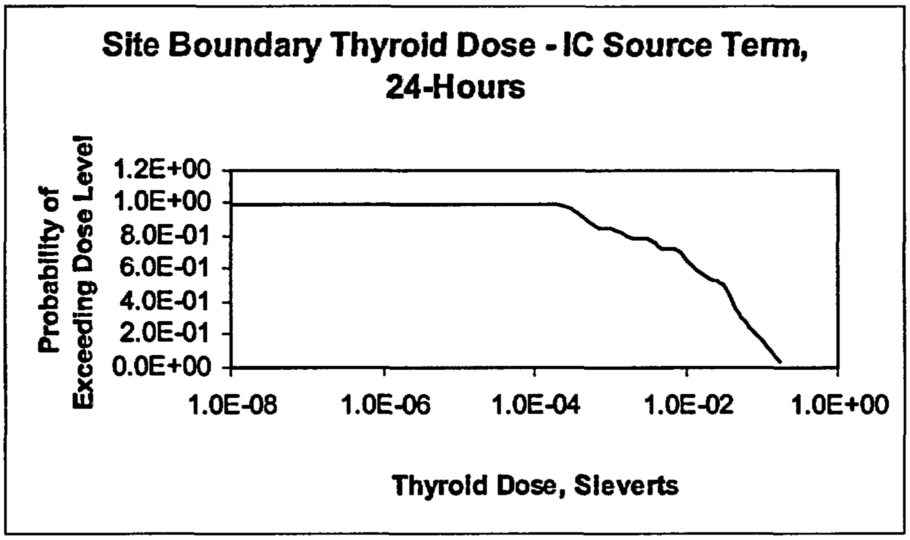


Figure 49-54

Site Boundary Red Bone Marrow Dose – IC Source Term, 72 Hours



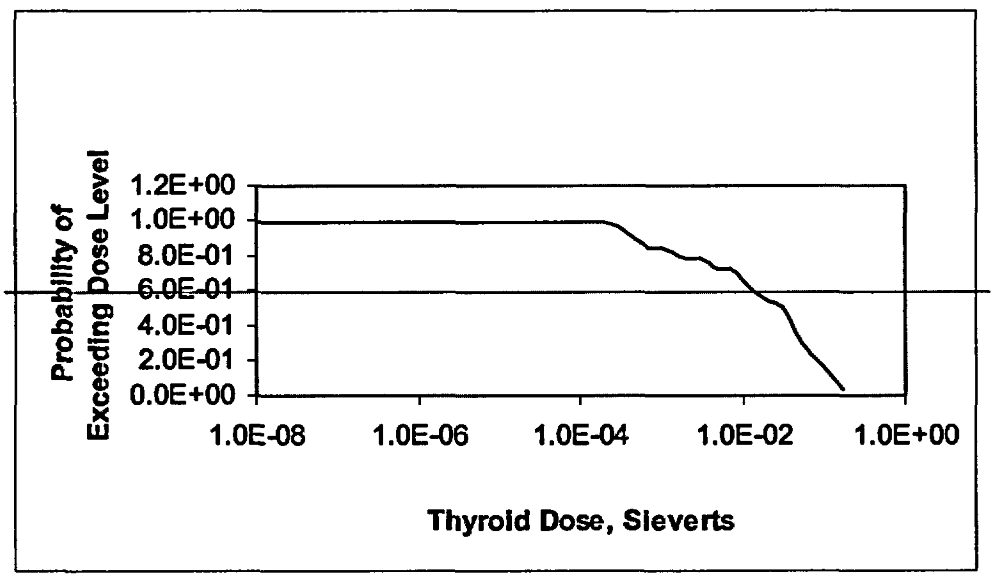
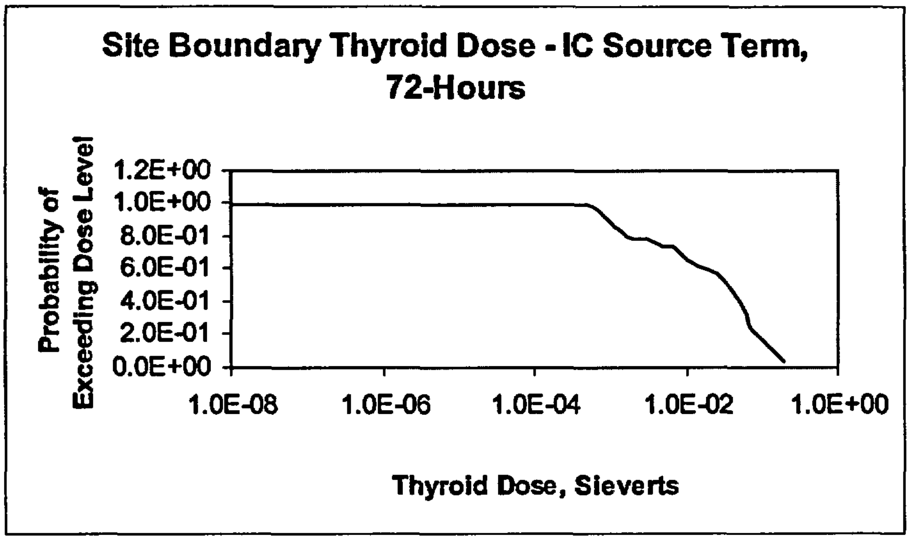


Figure 49-55

Site Boundary Thyroid Dose – IC Source Term, 24 Hours



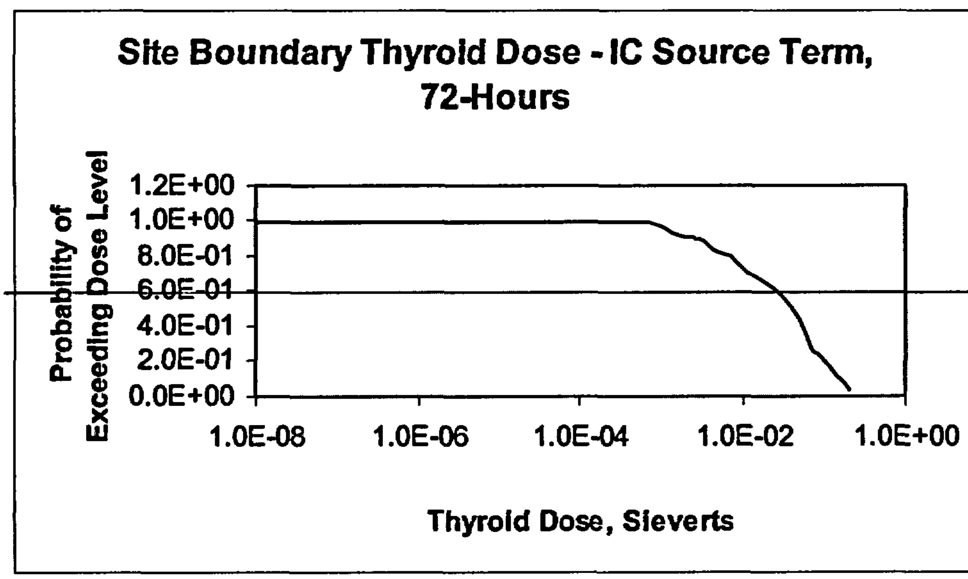
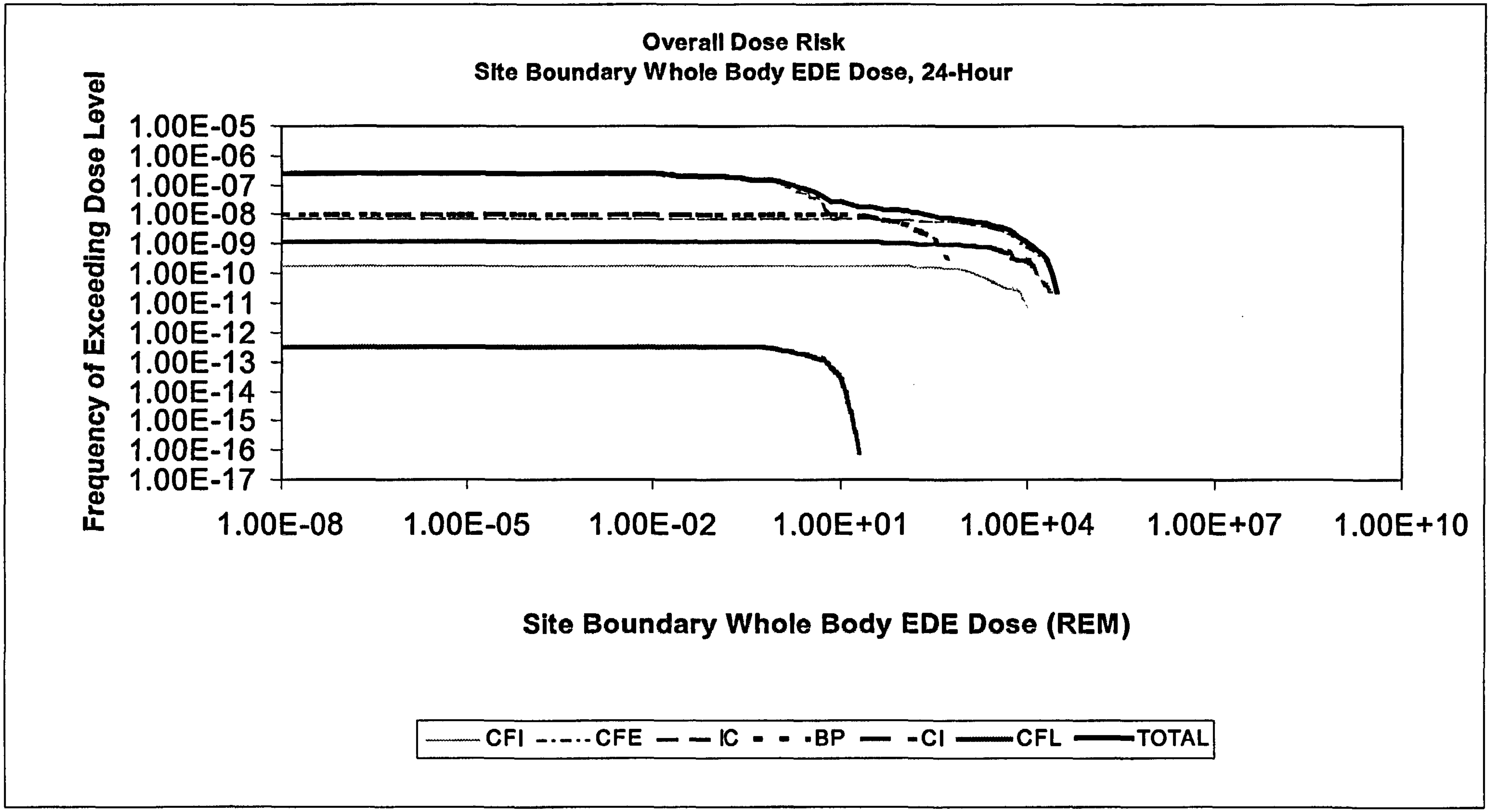


Figure 49-56

Site Boundary Thyroid Dose – IC Source Term, 72 Hours


Overall Dose Risk
Site Boundary Whole Body EDE Dose, 24-Hour
Frequency of Exceeding Dose Level
1.00E-05
1.00E-06
1.00E-07
1.00E-08
1.00E-09
1.00E-10
1.00E-11
1.00E-12
1.00E-13
1.00E-14
1.00E-15
1.00E-16
1.00E-17
1.00E-08
1.00E-05
1.00E-02
1.00E+01
1.00E+04
1.00E+07
1.00E+10
Site Boundary Whole Body EDE Dose (REM)
CFI
CFE
IC
BP
CI
CFL
TOTAL

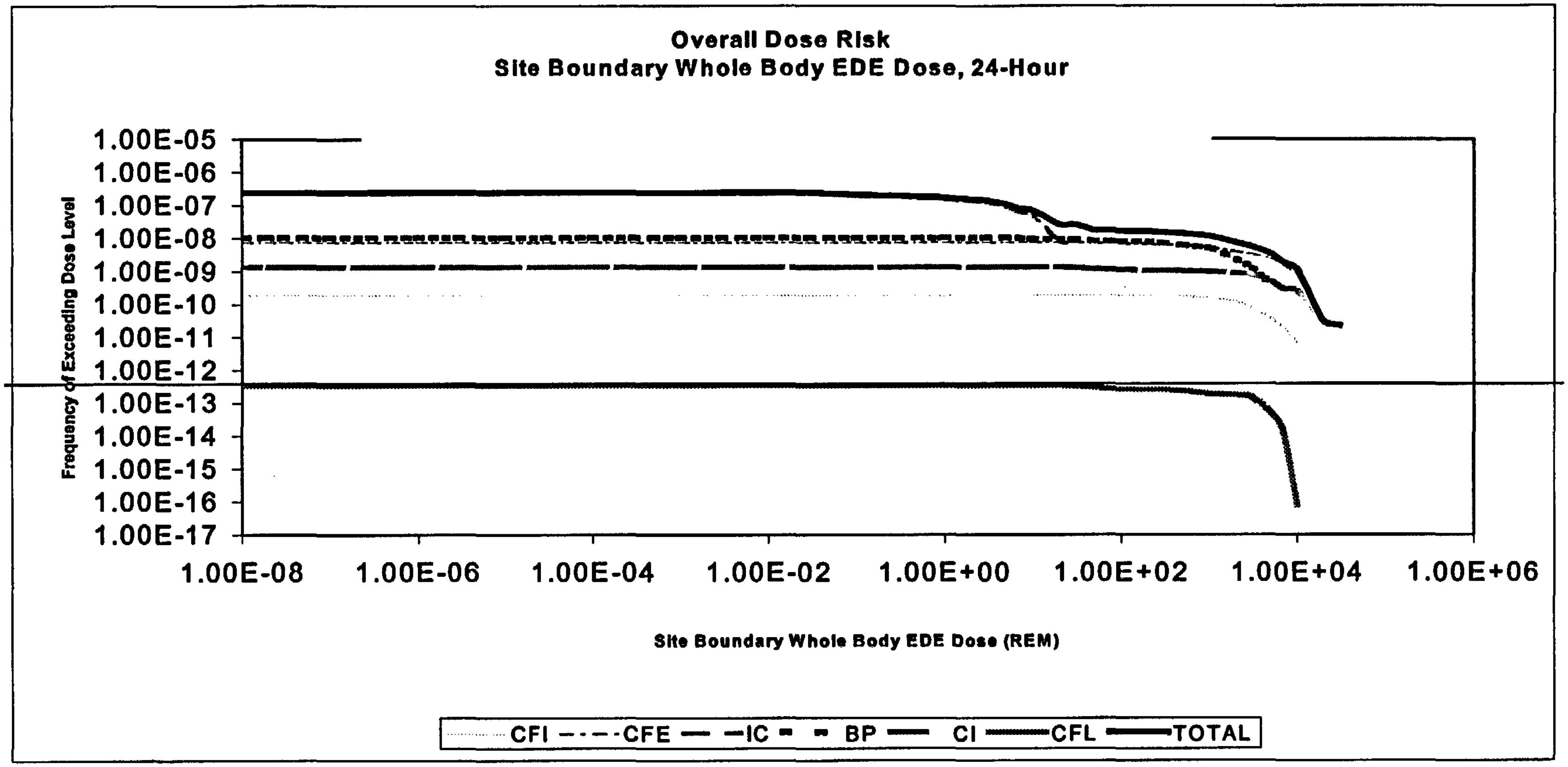
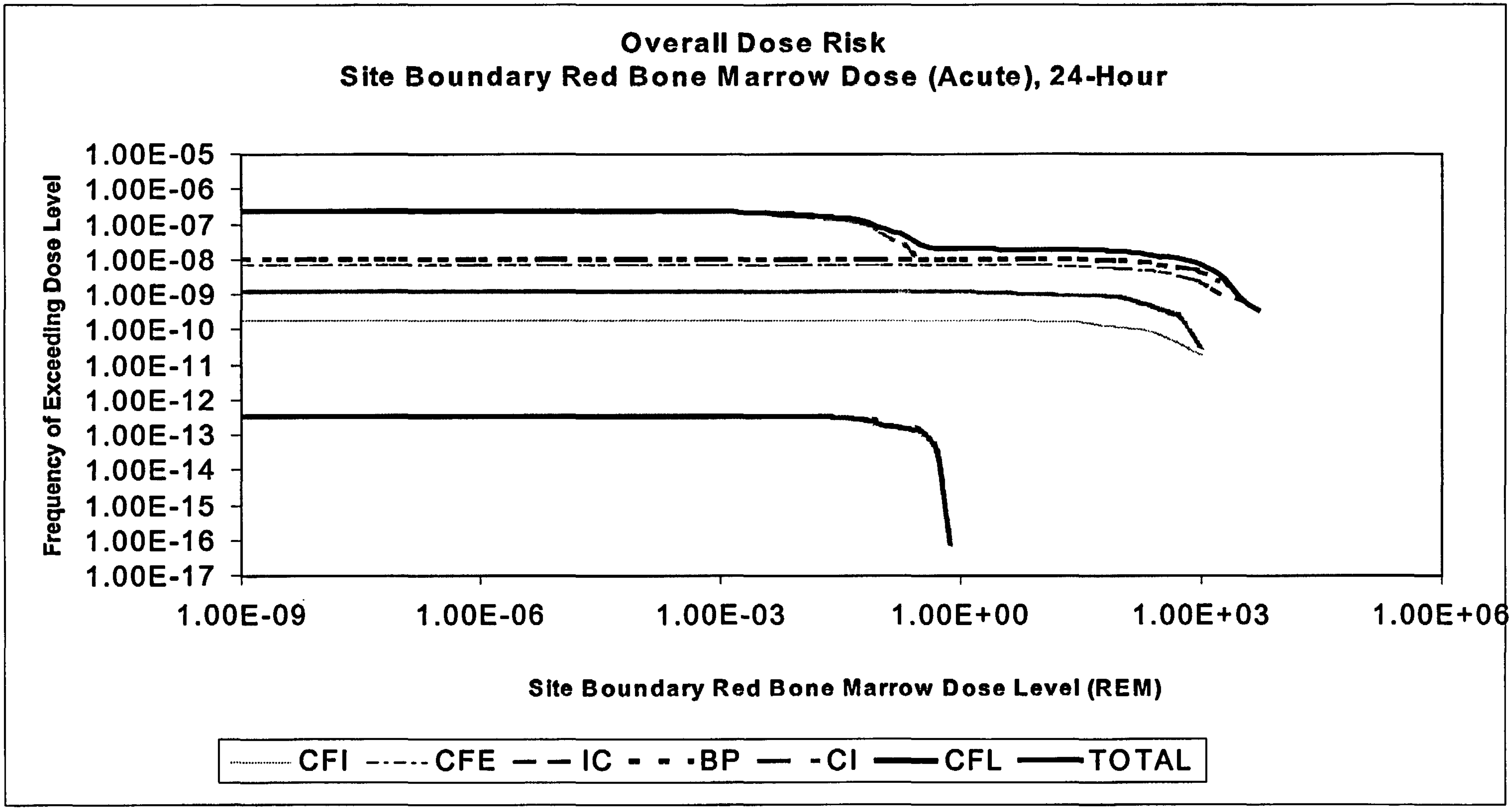


Figure 49-57

Overall Dose Risk – Site Boundary Whole Body EDE Dose, 24 Hours


Overall Dose Risk
Site Boundary Red Bone Marrow Dose (Acute), 24-Hour
Frequency of Exceeding Dose Level
1.00E-05
1.00E-06
1.00E-07
1.00E-08
1.00E-09
1.00E-10
1.00E-11
1.00E-12
1.00E-13
1.00E-14
1.00E-15
1.00E-16
1.00E-17
1.00E-09
1.00E-06
1.00E-03
1.00E+00
1.00E+03
1.00E+06
Site Boundary Red Bone Marrow Dose Level (REM)
CFI
CFE
IC
BP
CI
CFL
TOTAL

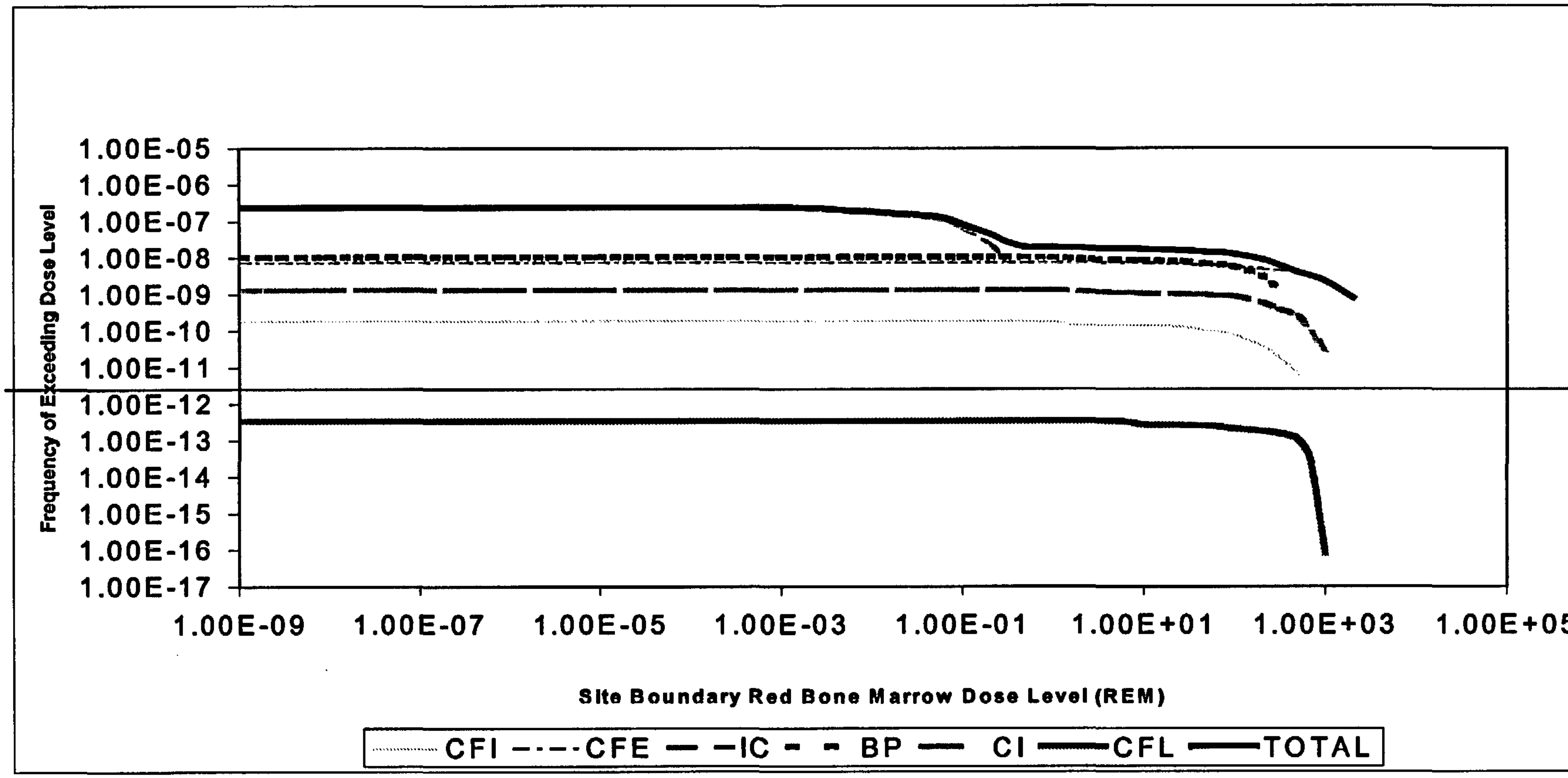


Figure 49-58

Overall Dose Risk – Site Boundary Red Bone Marrow Dose (Acute), 24 Hours

REVISIONS TO PRA REPORT REVISION 3 CHAPTER 59

PRA RESULTS AND INSIGHTS

From page 59-25:

59.7 Plant Dose Risk From Release of Fission-Products

Chapter 49 discusses the Level 3 results for at-power and shutdown internal events. The dose risks are quantified by multiplying the fission product release category frequency vector by the release category mean dose vectors. The goal is that a 24-hour, whole-body, site boundary dose greater than 25 rem has a frequency (large release frequency) of less than 1E-06 per year. The AP1000 large release frequency is 1.95E-08 per year, which is a factor of 50 times less than the goal.

The total at-power risk from a postulated release of fission products (the 24-hour, site boundary effective dose equivalent (EDE) is ~~5.15E-05~~**1.83E-04** rem per reactor-year. For shutdown, this risk was calculated to be 7.1E-05 rem per reactor-year for AP600. For AP1000, this shutdown risk could be estimated as 9.7E-05 rem per reactor-year (estimated the same way as shutdown LRF in Table 59-15). Table 59-16 and Figure 59-2 summarize the plant dose results.

~~Early containment~~**Containment** bypass failures account for ~~61~~**79** percent of the dose risk. These types of failures are usually assumed as a result of ~~sump flooding failure, vessel failure, or core reflooding failure~~**steam generator tube rupture**. A less conservative analysis of the ~~early~~ containment bypass failures may show a smaller frequency, and, as a result, a smaller dose risk.

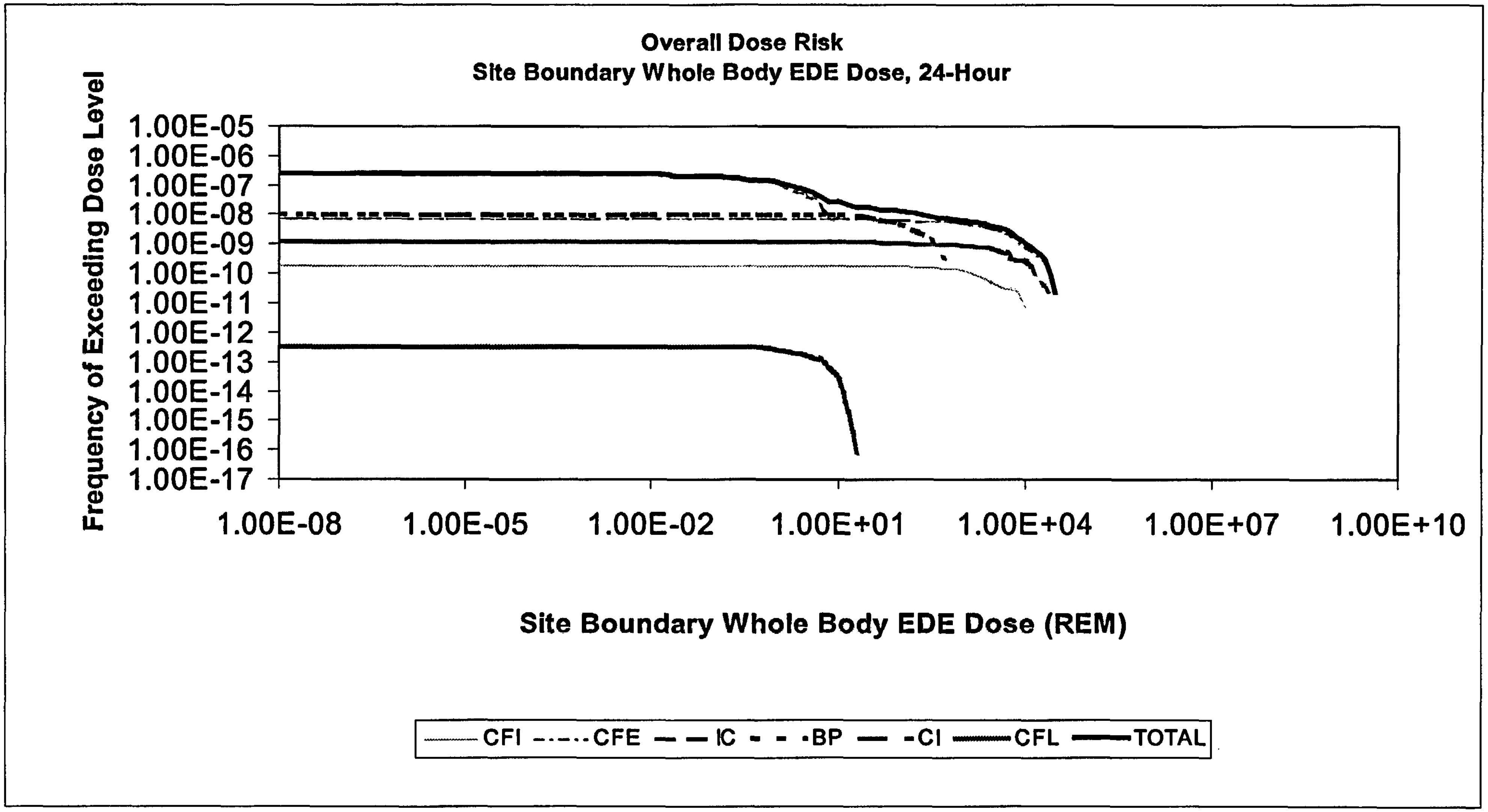
From page 59-73:

Table 59-16

SITE BOUNDARY WHOLE BODY EDE DOSE RISK – 24 HOURS

Release Category	Release Frequency (/reactor year)	Mean Dose (sieverts)	Dose (REM)	Risk (REM/reactor year)	Percent Contribution to Total Risk
CFI	1.89E-10	2.59E+01~~3.25E+01~~	2.59E+03~~3.25E+03~~	4.90E-07~~6.14E-07~~	0.3~~1.2~~
CFE	7.47E-09	4.23E+01~~4.23E+01~~	4.23E+03~~4.23E+03~~	3.16E-05~~3.16E-05~~	17.3~~61.4~~
IC	2.21E-07	1.82E-02~~1.82E-02~~	1.82E+00~~1.82E+00-01~~	4.02E-07~~4.02E-07~~	0.2~~0.8~~
BP	1.05E-08	1.37E+02~~1.15E+01~~	1.37E+04~~1.15E+03~~	1.44E-04~~1.21E-05~~	78.6~~23.5~~
CI	1.33E-09	5.10E+01~~5.10E+01~~	5.10E+03~~5.10E+03~~	6.78E-06~~6.78E-06~~	3.7~~13.2~~
CFL	3.45E-13	3.84E-02~~2.58E+01~~	3.84E+00~~2.58E+03~~	1.32E-12~~8.90E-10~~	0.0~~0.0~~
	2.4E-07		Total Risk =	1.83E-04~~5.15E-05~~	100.0

From page 59-100:


Overall Dose Risk
Site Boundary Whole Body EDE Dose, 24-Hour
Frequency of Exceeding Dose Level
1.00E-05
1.00E-06
1.00E-07
1.00E-08
1.00E-09
1.00E-10
1.00E-11
1.00E-12
1.00E-13
1.00E-14
1.00E-15
1.00E-16
1.00E-17
1.00E-08
1.00E-05
1.00E-02
1.00E+01
1.00E+04
1.00E+07
1.00E+10
Site Boundary Whole Body EDE Dose (REM)
CFI
CFE
IC
BP
CI
CFL
TOTAL

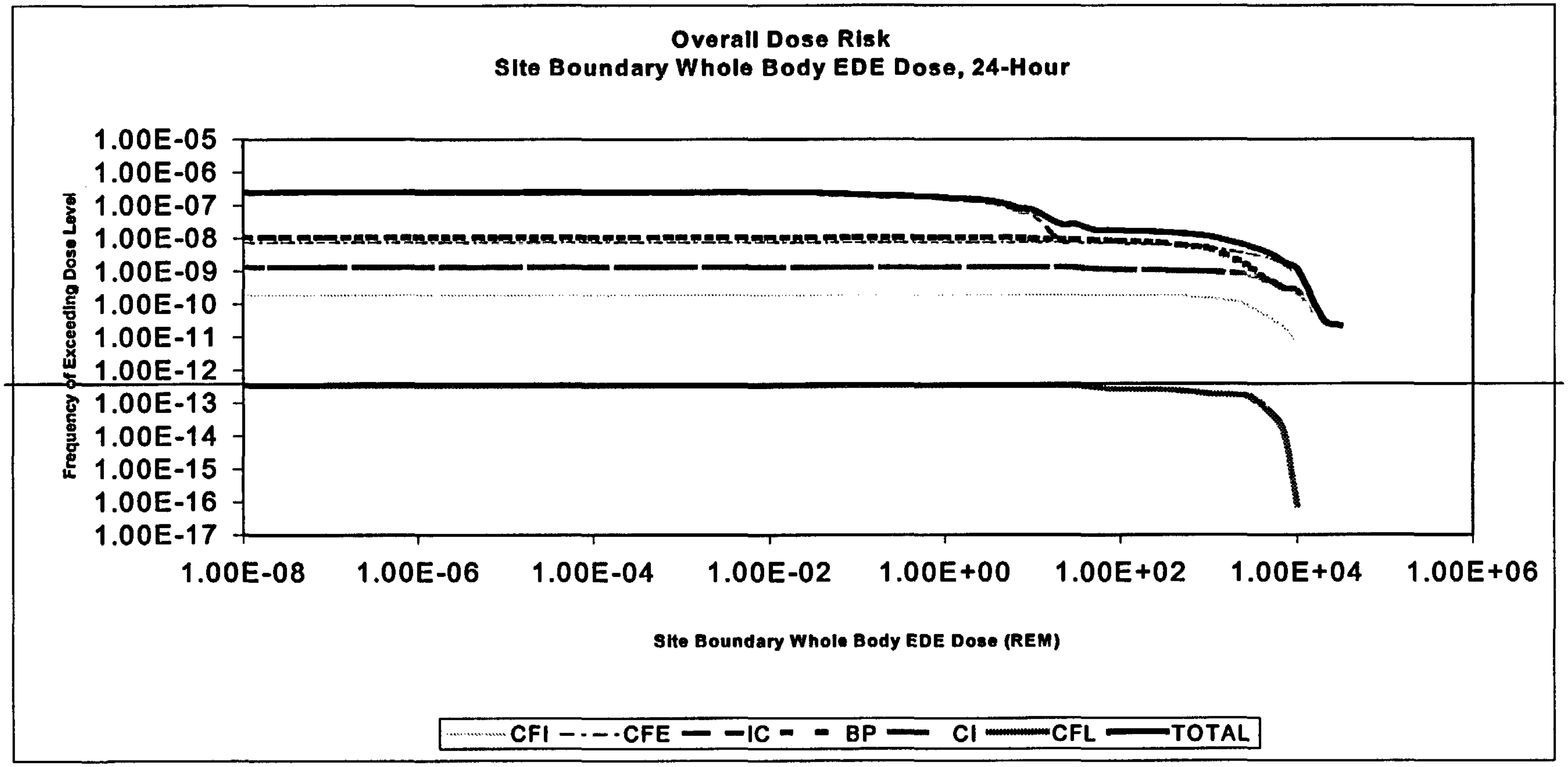


Figure 59-2

24-Hour Site Boundary Dose Cumulative Frequency Distribution

Table 15B-1

AEROSOL REMOVAL COEFFICIENTS IN THE AP1000 ~~AP600~~ CONTAINMENT FOLLOWING A DESIGN BASIS LOCA WITH CORE MELT(1)

(CONTENT TO BE REVISED)

Time Interval (hours)		Removal Coefficient (hr^{-1})
~~0.167~~	~~0.3~~	~~0.59~~
~~0.3~~	~~0.4~~	~~0.58~~
~~0.4~~	~~0.5~~	~~0.55~~
~~0.5~~	~~0.6~~	~~0.53~~
~~0.6~~	~~0.7~~	~~0.50~~
~~0.7~~	~~0.8~~	~~0.51~~
~~0.8~~	~~0.9~~	~~0.53~~
~~0.9~~	~~1.0~~	~~0.67~~
~~1.0~~	~~1.2~~	~~0.65~~
~~1.2~~	~~1.3~~	~~0.67~~
~~1.3~~	~~1.7~~	~~0.70~~
~~1.7~~	~~1.8~~	~~0.72~~
~~1.8~~	~~1.9~~	~~0.70~~
~~1.9~~	~~2.0~~	~~0.72~~
~~2.0~~	~~2.1~~	~~0.71~~
~~2.1~~	~~2.2~~	~~0.72~~
~~2.2~~	~~2.3~~	~~0.71~~
~~2.3~~	~~2.6~~	~~0.69~~
~~2.6~~	~~2.8~~	~~0.67~~
~~2.8~~	~~3.4~~	~~0.65~~
~~3.4~~	~~3.8~~	~~0.64~~
~~3.8~~	~~4.0~~	~~0.63~~
~~4.0~~	~~4.5~~	~~0.62~~
~~4.5~~	~~5.0~~	~~0.60~~
~~5.0~~	~~5.5~~	~~0.59~~
~~5.5~~	~~6.0~~	~~0.57~~
~~6.0~~	~~6.5~~	~~0.56~~
~~6.5~~	~~7.0~~	~~0.54~~
~~7.0~~	~~8.0~~	~~0.53~~
~~8.0~~	~~9.0~~	~~0.51~~
~~9.0~~	~~10.0~~	~~0.50~~
~~10.0~~	~~12.0~~	~~0.48~~
~~12.0~~	~~14.0~~	~~0.47~~
~~14.0~~	~~20.0~~	~~0.45~~
~~20.0~~	~~24.0~~	~~0.43~~

~~Note:~~

~~1. To provide additional conservatism, the aerosol removal coefficients provided in this table are 0.1 hr^{-1} lower than the values calculated using the models and assumptions of this appendix.~~

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.2-8 (Revision 1)

Original RAI Number(s): None

Summary of Issue:

Deterministic Approach

The applicant used the deterministic approach to estimate the HCLPF values of primary system component supports. The components included in the approach are: polar crane, baffle plate supports, heat exchanger for the passive residual heat removal system, core makeup tank and valves. The applicant used lower bound values, and it appears that there was no need for invoking factors of conservatism to arrive at the HCLPF values. It is noted that the core makeup tank has a HCLPF value of 0.54g; therefore, any increase in seismic response of the containment internal structure due to lift off of the internal structure or the nuclear island structure would necessitate a review of this HCLPF value. This is Open Item 19A.2-8.

Additional comments during meeting on July 10, 2003

This response is incomplete; it does not include the effect of CIS lift off.

Westinghouse Response (Revision 1):

Nuclear Island Basemat Uplift

The effects of basemat uplift have been evaluated for AP1000 using seismic time history analyses. Peak accelerations, floor response spectra, and member forces from seismic time history analyses that included basemat uplift were compared to seismic time history analyses that did not include lift off. The comparisons show that the basemat uplift effect is insignificant. Results and discussion are given in DSER Open Item Number 3.7.2.3-1.

In order to make conclusions for higher seismic events (> 0.3g), seismic response spectra (@5% equipment damping) were developed for the 0.5g case. Response spectra that include nonlinear liftoff effects were developed at different elevations using the auxiliary shield building (ASB) stick model described in DSER Open Item Number 3.7.2.3-1. The maximum uplift of the basemat is 0.29". In Figures 19A.2-8-1 to 19A.2-8-5 comparisons are made of these response spectra to the seismic response spectra using the linear ASB stick model. As seen from this comparison liftoff is not significant for horizontal response since the horizontal response spectra are similar. In the region of the shield building roof the vertical response spectra are comparable. There are differences in the vertical response spectra in the higher frequency region for the Shield Building cylinder up to elevation 265'. However, the differences shown in these curves will not affect the HCLPF values because:

- High frequency content due to impact is not damaging

- High frequency seismic response is limited to the cylinder portion of the Shield Building
- Side soil effects, not considered in overturning study, will have significant effect on liftoff

These items are discussed in more detail below.

High Frequency Content

High frequency content caused by liftoff is intermittent during the seismic response due to the impact of the NI basemat on the foundation media. This is not a damaging excitation since response is limited and resonance effects are greatly reduced.

High frequency seismic response of the Shield Building cylinder

The cylinder portion of the Shield Building high frequency content is amplified. Outside of the Shield Building, this high frequency response will not be as pronounced since the height of the other buildings are significantly reduced, and the mass effect is not as prominent because of the height reduction and there is no water mass at high elevations as there is for the Shield Building because of the PCS tank. The high frequency response will be filtered in the other buildings reducing rigid body motion, similar to the Shield Building roof response, due to such items as lower response frequencies and construction joints that introduce "gaps".

Side Soil Effects

It is noted that these analyses did not include the effect of side soils that can be significant. For the AP600 plant a study of seismic soil pressure distribution on the Nuclear Island structure was made. The effect of side soil on the basemat is a measure of its effect on the seismic response with basemat liftoff. The analyses performed in this study used a uniform subgrade modulus and included liftoff when the soil springs would be in tension. Soil reactions on the underside of the mat and the sidewalls were investigated from a two-dimensional (2D) SASSI analysis of the AP600. The SASSI results indicate that bearing pressures due to seismic loads may be only 54 percent of those calculated neglecting the benefit of the side soil. From the AP600 study of seismic soil pressure distribution it can be concluded that side soil effects, if included in the nonlinear liftoff seismic response study, can reduce (potentially significantly) the vertical seismic response. Using 50%, similar to the reduction in bearing pressure documented from the AP600 study, the vertical seismic response spectra with liftoff will be similar or below the seismic response spectra with no liftoff.

Containment Internal Structure Uplift under SSE loads

The containment internal structure uplift was addressed for the AP1000 in the response to RAI 220.022. The RAI response is revised and supplemented below.

The bottom head of the containment vessel rests on the nuclear island basemat. The containment internal structures basemat rests within the bottom head. There are no shear studs or anchors designed to transfer loads tangential to the vessel surface. The

interface is designed to transfer load in compression and friction. The configuration is identical to the AP600.

The provisions in the nuclear island basemat model are included for use in the equivalent static analyses to develop design loads for basemat design. The uplift capability assures that the reaction between the two basemats is correctly transferred as compression loads only. The stability evaluation follows the AP600 methodology described in the AP600 response to RAI 230.47. Two cases were evaluated:

- The evaluation was performed for seismic loads obtained from the time history analyses of the nuclear island stick model as presented in the tables in DCD subsection 3.7.2. Maximum member forces from the containment internal structure stick model at grade include the phasing of the response of the containment internal structures and the reactor coolant loop. These member forces were combined absolutely with the maximum member forces from the containment vessel stick and the mass times acceleration of the concrete inside containment below grade. The stability evaluation shows a factor of safety against overturning of 2.10 for the safe shutdown earthquake.

- The evaluation was performed for seismic loads obtained from the equivalent static analyses of the containment internal structures. These analyses apply the maximum response acceleration at each mass node of the building stick models and also assume an absolute summation of the reactor coolant loop support loads. The stability evaluation shows a factor of safety against overturning of 1.95 for the safe shutdown earthquake.

Since the deadweight has not been overcome no "liftoff or slapping" is expected to occur. However, allowing for a small separation of the containment internal structures from the basemat, there would be no significant effect to the seismic design loads or the in-structure response spectra. A small separation (slapping) might cause small localized changes in seismic response loads similar to those for the lift off observed between the nuclear island basemat and the rock addressed previously in the response to RAI 230.021. Any change in high frequency response due to "slapping back" would not propagate through the building structures to affect the seismic response. This is because of energy loss, damping, and filtering effects due to gaps and cracking. Therefore, it is not necessary to modify the analysis methods for the safe shutdown earthquake from those that were accepted by the NRC for the AP600 plant.

The lift off of the containment internal structures and the containment vessel relative to the nuclear island basemat has also been confirmed using nonlinear analyses with equivalent static loads applied to the ANSYS model of the containment and internal structures. The loads applied are:

- Dead Weight
- Seismic Loads

- These loads are applied as equivalent static loads and include loads from the steel containment vessel and the reactor coolant loop.
- The inertia loads on the containment internal structures concrete are defined from the plant equivalent static seismic acceleration values.
- The loads from the steel containment vessel are defined as equivalent static forces
- The vertical seismic accelerations act opposite to the dead weight.
- The horizontal accelerations are taken in the direction from the center of containment to the center of mass of the CIS. This minimizes the dead load resistance.
- The analyses are performed applying a factor α to all of the seismic loads (a value of 1.0 for α is equivalent to the plant SSE level). α is increased until instability occurs. Torsional loads are not considered.
- The seismic loads in the three directions are combined by the (1.0, 0.4, 0.4) method.

The analyses were performed using the 3D CIS super element created from the detail finite element model shown in DCD Figure 3.7.2-3. ANSYS Contact 52 elements were added at each of the interface nodes within a radius of 60.6' at the surface of the containment vessel bottom head. These elements have springs and gaps normal to the surface of the containment vessel and carry compression only. They have sliding friction tangential to the vessel. The bottom end of the spring was fixed.
Figure 19A.2-8-6 shows the vertical displacement at the edge of the containment internal structures at elevation 100' as a function of the value of α for seismic loads in the EW and NS directions. The analyses became unstable at a value of α greater than 1.75 for primary loads in the east west direction and at a value of 1.80 for primary loads in the north south direction. These values are lower than those in the hand calculations described above using the conservative equivalent static loads. These differences are due to the elevation of the highest interface node in the ANSYS model, consideration of three directions of seismic input instead of 2 in the hand calculations, and the fact that ANSYS is unable to analyse the actual point of instability. At a value of 1.0, corresponding to input ground motion of 0.3g, the vertical deflection at the edge is 0.098 inches. The lift off at other locations of the interface is smaller than those at the edge. This lift off is conservative as demonstrated by the more realistic time history loads used in the hand calculation described above. The actual lift off would be smaller than that evaluated for the nuclear island basemat lift off and there would be no significant effect on the seismic design loads or on the in-structure response spectra.

Containment Internal Structure Uplift under Review Level Earthquake

As noted above, the SSE stability analyses that have been performed are very conservative because:

- The seismic loads are applied statically without consideration of their variation due to the dynamic response of the structures.

- **The equivalent static loads are based on the maximum loads associated with each component for the complete seismic time history without consideration of the time of occurrence.**
- **The loads associated with the CIS, SCV, and RCL are added absolutely.**

The overturning moment due to the steel containment vessel is about equal to that due to the concrete internal structures. Combination of these loads by the square root of the sum of the squares instead of by absolute sum would reduce the overturning moment by about 30%. Further, during earthquakes higher than the site SSE level (0.3g), the damping will increase because of concrete cracking. Considering the increase in damping from 5 to 7 percent there is a potential reduction in response of 13% (based on Reg. Guide 1.60 at 10 hertz). There will also be nonlinear behavior due to the increase in stress and strain levels within the structures that will further reduce the overturning loads since structures will tend to respond away from resonance. Recognizing these conservatisms the uplift response is expected to be similar to that described for the SSE with any separation less than 0.1".

Conclusion

There is no increase in seismic response of the containment internal structure due to lift off **of either the internal structure or the nuclear island structure** that will change the HCLPF values.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

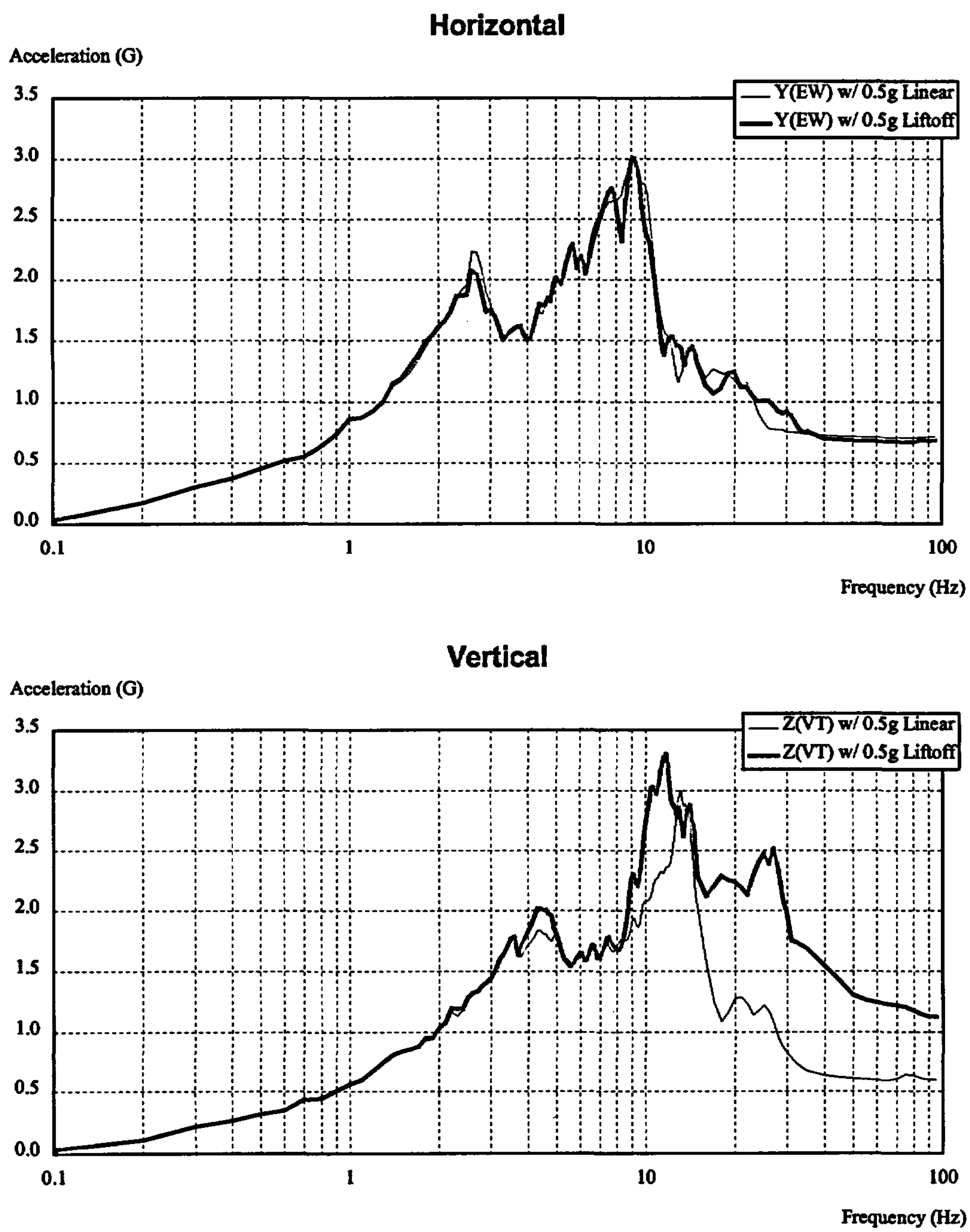
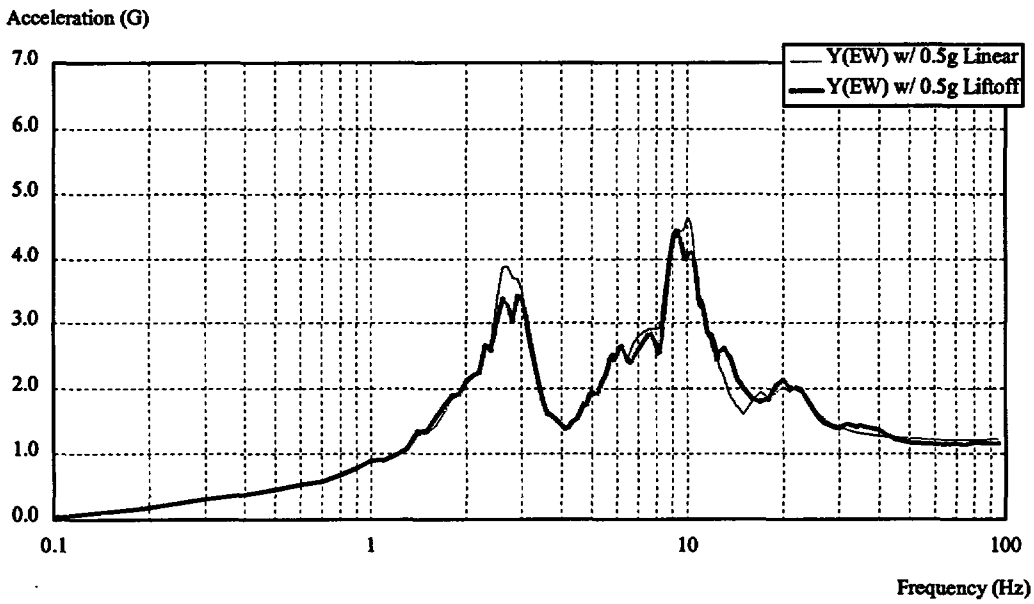


Figure 19A.2-8-1: Floor Response Spectra of ASB Node at EL. 116.50'

Horizontal



Vertical

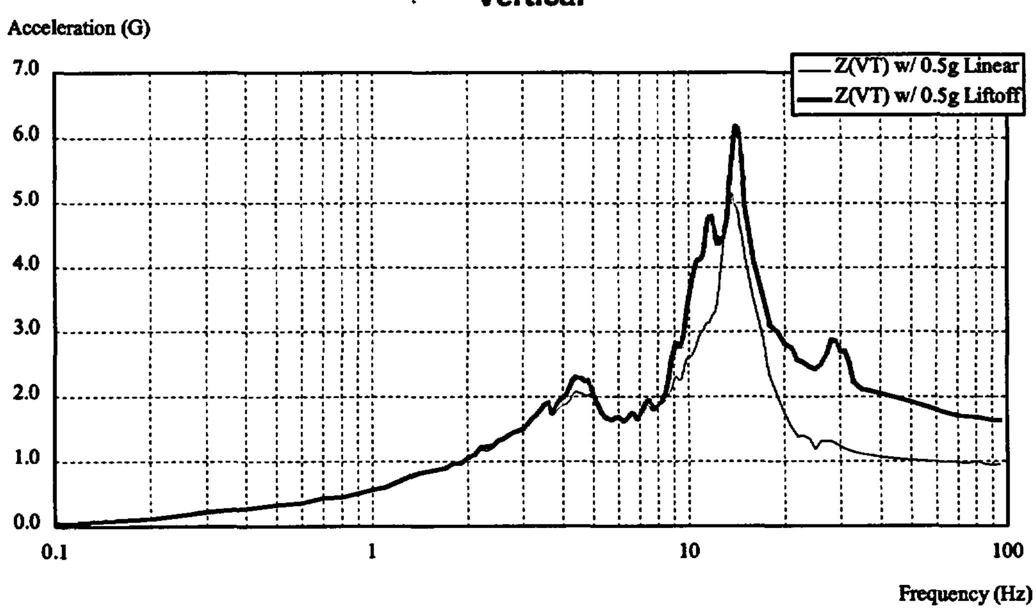
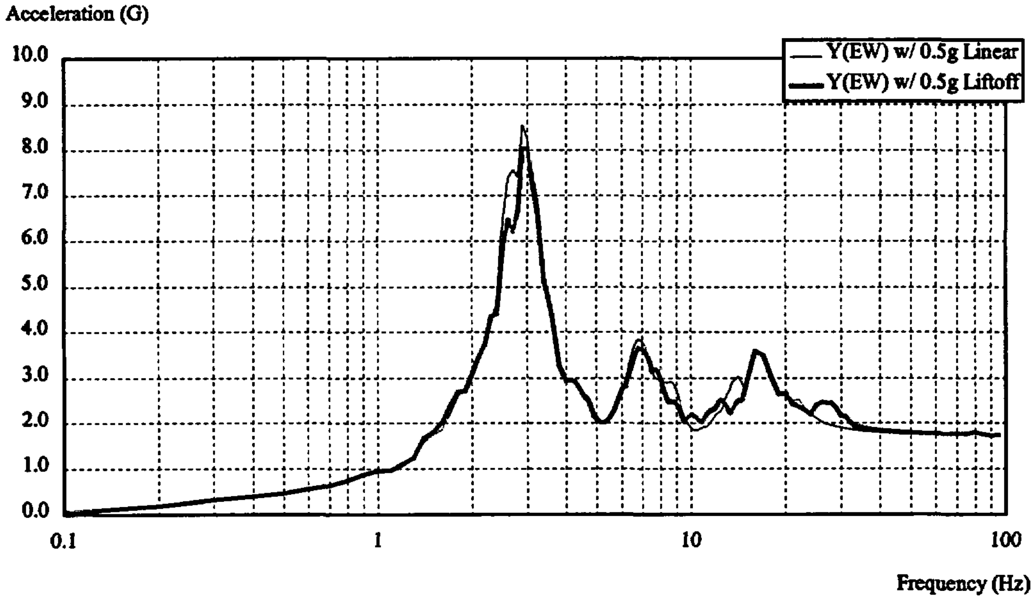


Figure 19A.2-8-2: Floor Response Spectra of ASB Node at EL. 179.56'

Horizontal



Vertical

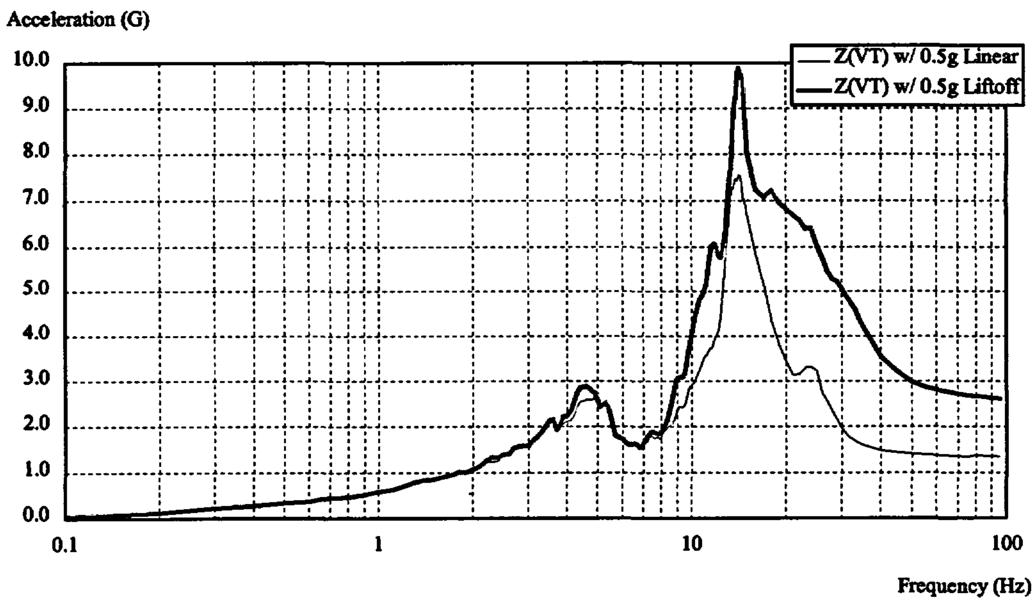
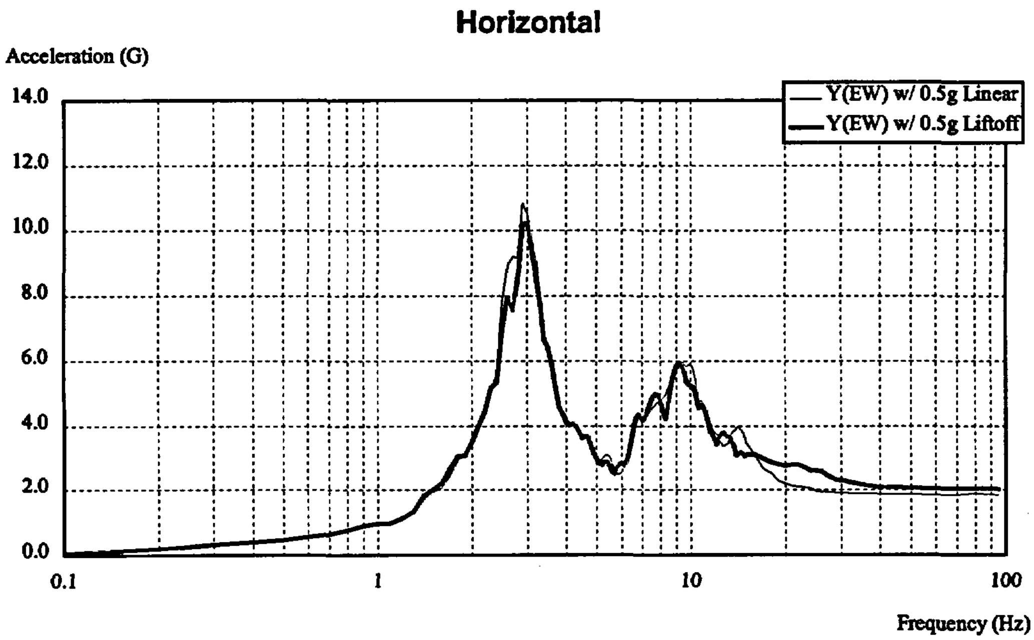


Figure 19A.2-8-3: Floor Response Spectra of ASB Node at EL. 265.00'



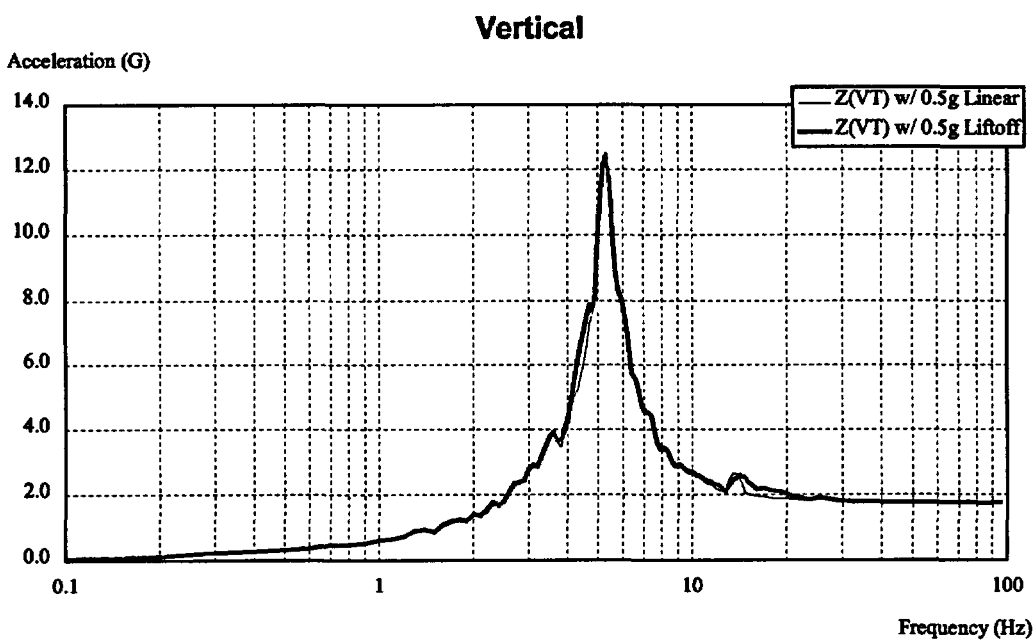
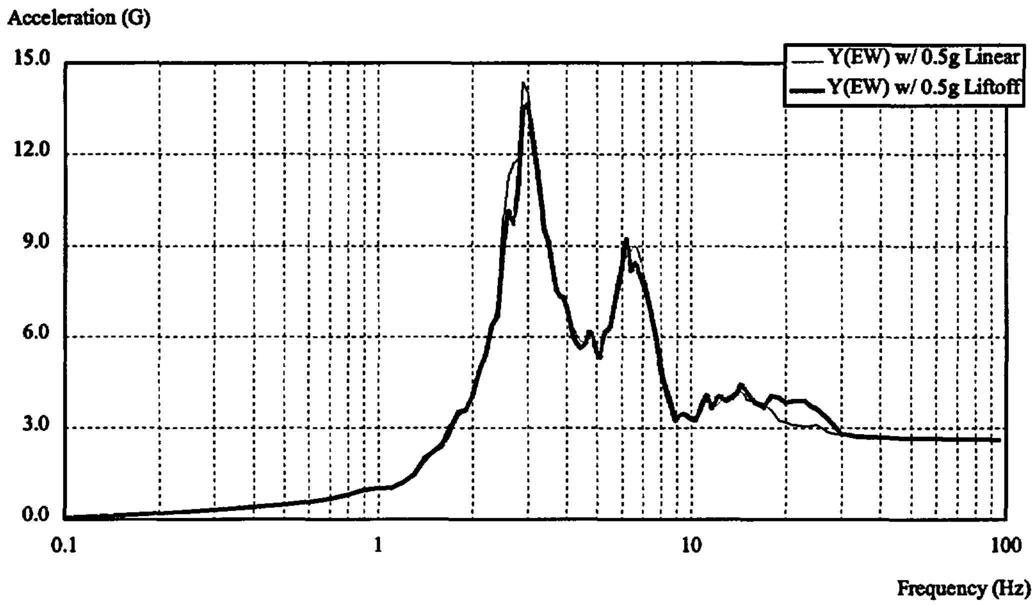


Figure 19A.2-8-4: Floor Response Spectra of ASB Node at EL. 295.23'

Horizontal



Vertical

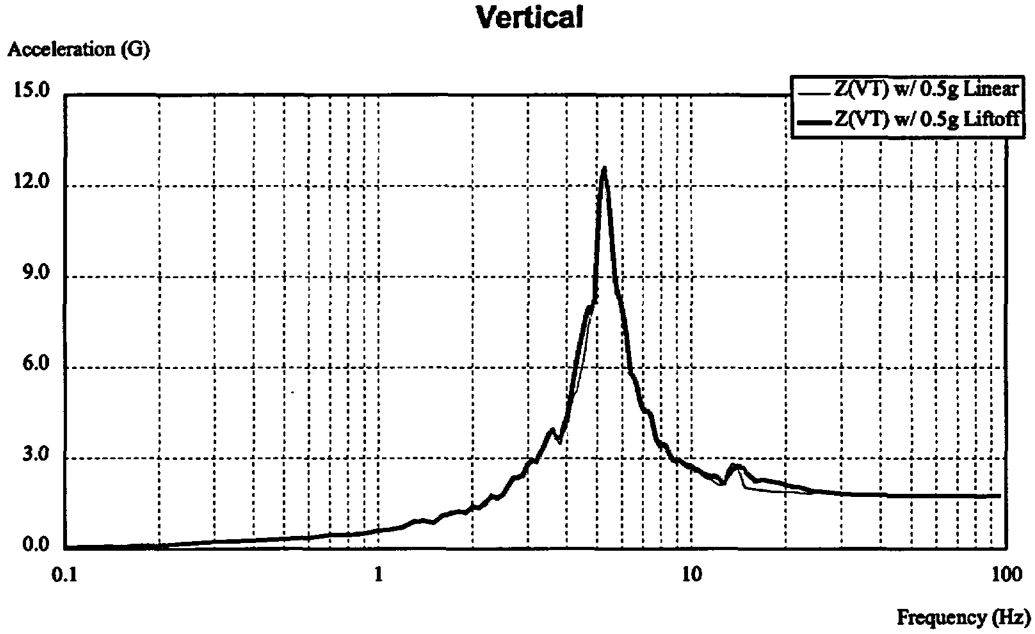
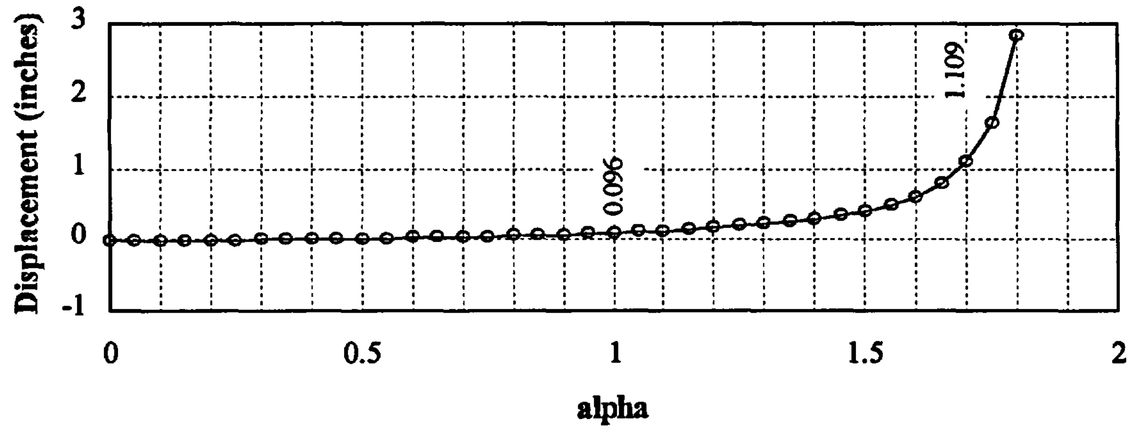


Figure 19A.2-8-5: Floor Response Spectra of ASB Node at EL. 333.13'

NS Primary:

Vertical displacement of North Edge (N100227)
Load case: DL+Alpha x (-1.0NS+0.4EW+0.4VT), Mu=0.4



EW Primary:

Vertical displacement of East Edge (N100251)
Load Case: DL+Alpha x (-0.4NS+1.0EW+0.4VT), Mu=0.4

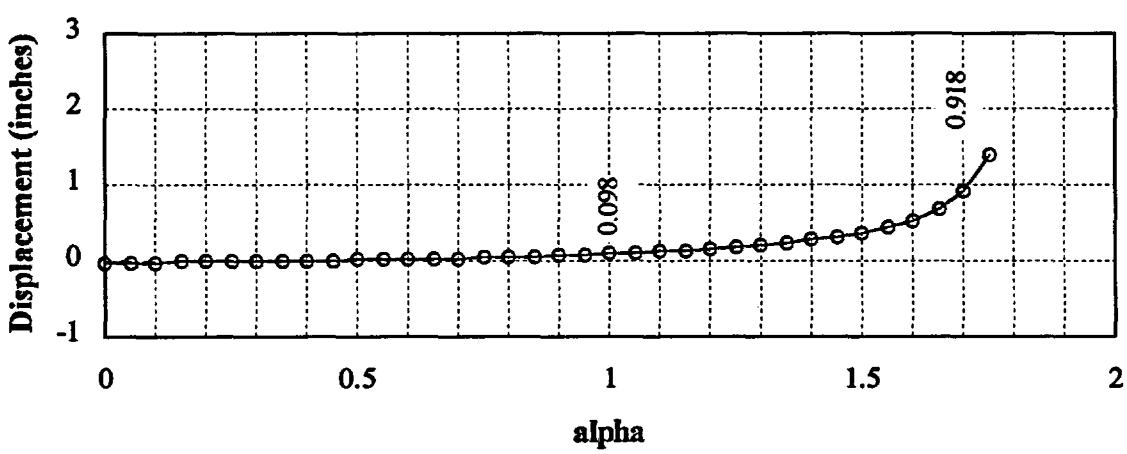


Figure 19A.2-8-6
Containment Internal Structures Overturning Evaluation