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ADDENDUM TO THE COMPOSITE ANALYSIS FOR THE E-AREA VAULTS AND SALTSTONE DISPOSAL FACILITIES

Authors

JAMES R. COOK

ELMER L. WILHITE

LEONARD B. COLLARD

Westinghouse Savannah River Company
Savannah River Technology Center
Aiken, SC 29808

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Westinghouse Savannah River Company
Savannah River Site
Aiken, SC 29808

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REVIEWS AND APPROVALS

Authors

James R. Cook, Author 4/16/02
Waste Disposal and Environmental Development Date

Elmer L. Wilhite, Author 4/16/2002
Waste Disposal and Environmental Development Date

B.T. Butcher for L.B. Collard
Leonard B. Collard, Author 4/16/02
Waste Disposal and Environmental Development Date

Approvals/Review

W.E. Stevens, Level 3 4/17/02
Waste Processing Technology Date

B.T. Butcher, Level 4 4/16/02
Waste Disposal and Environmental Development Date

W.T. Goldston 4/16/02
Solid Waste Division Date

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INTRODUCTION

Revision 1 of the Composite Analysis (CA) Addendum has been prepared to respond to the U.S. Department of Energy (USDOE) Low-Level Waste Disposal Facilities Federal Review Group (LFRG) review of the CA. The following section describes the revision history of this document.

Revision History

In September 1997, the Composite Analysis (CA) for the E-Area Vaults and Saltstone Disposal Facilities (WSRC-RP-97-311, Rev. 0) was issued per USDOE 5820.2A and associated guidance. Subsequently, the U.S. Department of Energy (USDOE) Low-Level Waste Disposal Facilities Federal Review Group (LFRG) conducted a review of the CA. On January 21, 1999, USDOE approved the CA with several conditions (J. Fiore and M. Frei Memorandum to Assistant Manager for Environmental Management, Savannah River Operations Office, *Review of the Savannah River Site Composite Analysis*, 1/21/99). This approval memorandum follows this discussion.

Revision 0 (September 23, 1999) of this addendum to the CA was prepared to respond to each of the conditions of approval. In October 2001, the LFRG provided comments on the Rev. 0 CA Addendum (M. Frei Memorandum to G. Rudy, Manager, Savannah River Operations Office, *Disposal Authorization for the Savannah River Site E-Area Vaults and Saltstone Disposal Facilities Composite Analysis Addendum*, 10/11/01). This memorandum indicated LFRG concurrence that many approval conditions had been adequately addressed but requested additional information or clarification for certain conditions. The memorandum and its Attachment 1, *LFRG Response to Outstanding Disposal authorization Statement Conditions for the Savannah River Site E-Area Vaults and Saltstone Disposal Facilities* are provided at the end of this discussion for reference.

The USDOE Savannah River (SR) Operations Office, Waste and Operations Division Director responded to the LFRG in November 2001 (V. Sauls Memorandum to J. Rhoderick et al., LFRG, *Disposal Authorization for the Savannah River E-area Vaults and Saltstone Disposal Facilities Composite Analysis Addendum [Memo, Frei to Rudy dated 11/21/01]*). This response contained information to address the outstanding information for Condition 1 and also indicated that the information or clarification requested to address Condition 3 and the Performance Assessment and Composite Analysis Maintenance Conditions would be included in the Rev. 1 CA Addendum. This response memorandum is also provided following this discussion for reference. In addition, a CA Condition 3 Response table is provided in this section to summarize the actions taken in response to each of the LFRG comments on Condition 3 to aid readers of the CA Addendum (Rev. 1) in finding new or modified text.

In this addendum, each of the conditions is stated in italicized text with the response following. The first four conditions are numbered as in the approval memorandum, the last three were unnumbered in the approval memorandum but have been numbered here for ease of reference. Documents attached to the CA Addendum have been reviewed and updated as follows during development of Rev. 1: Attachment 1, the maintenance program has been updated with the most current document available, Attachment 2, the land use plan has been updated with the recently released Long Range Comprehensive Plan. Review of this new document indicates that the land use items critical to the CA (i.e., existing site boundaries and permanent restriction on residential use of SRS property) remain unchanged.

Per the SRS PA/CA maintenance plan, *Performance Assessment and Composite Analysis Maintenance Program,* (SWD-SWE-2002-00002), Attachment 1, the information contained in this addendum will be incorporated into the next revision of the Composite Analysis.

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Composite Analysis Condition 3 Response Table

Condition	LFRG Recommendations	Response
Perform a sensitivity analysis on the radionuclides important to the composite analysis and flux rates on the hydrogeologic model including the groundwater divide and the model boundary conditions. Perform an uncertainty analysis on the inventory, flux rates, and resultant dose calculations for the radionuclides important to the composite analysis.	This condition has not been met and remains open. The addendum to the composite analysis considers the sensitivity and uncertainty of the results presented in the Composite Analysis. The sensitivity of the results to the radionuclides considered and the hydrologic model were addressed. The sensitivity of the hydrologic model was examined with regards to the groundwater divide between Upper Three Runs and Fourmile Branch, and boundary conditions used in the modeling. An uncertainty analysis was presented that considered the inventory, flux rates, and resultant dose for radionuclides important to the Composite Analysis. The review of the sensitivity and uncertainty analysis in the Composite Analysis addendum raised the following items that should be addressed with the attendant revisions to the Composite Analysis.	
	• On page 2-6, there is a discussion of upward flow from the aquifer into the creek. This is physically questionable, because there is no mechanism for producing an artesian gradient in the vicinity of the creek. Altering the second sentence to read, "As the groundwater flow approaches the groundwater divide created by a stream, the flow discharges to the ground surface at seepage faces comprising the stream bed and/or adjoining wetland areas." remedies this misconception.	No change has been made to the document based on this comment. SRS supports the hydrogeologic description of the conditions in the field as currently described. These conditions are similar to those described in Freeze, R.A. and J.A. Cherry, 1979, *Groundwater*, which provides additional credence for the description.

Condition	LFRG Recommendations	Response
	• Figure 2.2-6 makes the graphical presentation of velocity vectors pointing upwards. This figure should be revised to suggest water flows horizontally in the vicinity of the stream and not vertically.	No change has been made to the document based on this comment. SRS supports the hydrogeologic description of the conditions in the field as currently described. These conditions are similar to those described in Freeze, R.A. and J.A. Cherry, 1979, *Groundwater*, which provides additional credence for the description.
	• On page 2-13 screening levels were set at 1 E-03, 1 E-04, and 1 E-05 below the maximum recorded inventories or contaminant fluxes. The justification for these differing screening levels is not provided. A single screening level would seem to be more reasonable.	Clarifying text has been added to page 2-10.
	• Figure 6.6-8 screens out F-sewer, H-sewer, and H-seep without explanation or justification.	The figure presented in the Rev. 0 addendum had plotting errors for the facilities in question. A revised figure has been provided in the Rev. 1 addendum. In this revised figure, the F Process Sewer does not screen out. The H Process Sewer and the H Seepage Basin GW OP do screen out.
	• In Table 6.6-4, footnote 14 screens out facilities that were added to the Composite analysis a few pages earlier. The apparent contradiction is not explained.	The actual peak flux value has been added to the table in place of the comment "Other Screen" and the unnecessary footnote has been deleted.

Condition	LFRG Recommendations	Response
	• In Table 6.6-5, F-sewer, H-sewer, and H-seep are excluded without explanation.	As noted above, the H Seep and H Sewer are properly screened out. The F Sewer should have been retained, with a peak flux to the water table of 2.91×10^{-4} Ci/yr. This flux value is about 30 times larger than the flux value from HT 9 and would have produced a dose approximately 30 times less than the 1.14×10^{-1} mrem/yr given for HT 9 in Table 6.6-9 of the addendum. This is an insignificant addition that would not have affected the conclusions from the CA. The next revision of the CA will carry out this calculation.
	• On page 2-26, the erroneous statement is made that the total dose at the stream is the sum of the doses from each source. This assumption is only correct if the mass flux for each source is the same. Since this is apparently not the case for the Composite Analysis, the assumption should be recast correctly and the results revised accordingly. Similar changes would be appropriate for Table 6.6-7 and Figure 6.6-9.	The referenced text, table, and figure have been modified for clarification.
	• Figures 6.6-19 to 6.6-21 show differing numbers of realizations exceeding the dose limit for different isotopes considered in the analysis. The significance of these results is not discussed or interpreted in the text on page 2-45.	The referenced text has been modified for clarification.

Condition	LFRG Recommendations	Response
	• The results of the uncertainty analysis would seem to be relevant for the development of the monitoring program in Section 6 of the Composite Analysis, but the relationship between the monitoring program and the analyses presented in the Composite Analysis Addendum is not apparent.	The uncertainty analysis requested by the LFRG was limited to the source terms used in the CA. As such, the end point of the study was the flux of contaminants to the water table. These results do not provide insight into potential monitoring program improvements. Text has been added to the monitoring section (Section 6) about the larger probabilistic uncertainty program underway at SRS and how information from this program might be used to improve the CA monitoring program.

1.0 Condition 1

Point of Assessment/Pathways – Based on the approved Land Use Plan and as a first step in a more comprehensive analysis, issue an addendum to the composite analysis to reflect a single point of compliance at the confluence of Upper Three Runs with the Savannah River using the recreational scenario currently in the composite analysis.

Following are pertinent sections of the Savannah River Site (SRS) CA, which have been revised in response to the condition stated above. Section numbering, headings, table and figure numbers, and references refer to the original CA document (WSRC-RP-97-311, Rev. 0). The complete source term for the Tims Branch watershed, developed in response to Condition 3, has been incorporated in these revised sections.

1.0 SUMMARY AND CONCLUSIONS

This report documents the CA performed on the two active SRS low-level radioactive waste (LLW) disposal facilities. The facilities are the Z-Area Saltstone Disposal Facility and the E-Area Vaults (EAV) Disposal Facility. The analysis calculated potential releases to the environment from all sources of residual radioactive material expected to remain in the General Separations Area (GSA). The GSA is the central part of the SRS and contains all of the waste disposal facilities, the chemical separation facilities and associated high-level waste storage facilities as well as numerous other sources of radioactive material. The analysis considered 114 potential sources of radioactive material containing 115 radionuclides.

As shown in Table 1-1, the calculated maximum dose to a hypothetical future member of the public is 1.8 mrem/year at the mouth of UTR, the point of maximum exposure to which the public may have access, based on the approved Future Use Plan (Attachment 2). This dose is well below the U.S. Department of Energy (USDOE) primary dose limit of 100 mrem/year and the dose constraint of 30 mrem/year. The calculated maximum collective dose to a hypothetical future population is 0.045 person-rem/year. The radionuclides contributing the majority of the dose are ^{3}H, ^{14}C, ^{237}Np, and isotopes of uranium. A former LLW disposal facility, the Mixed Waste Management Facility (MWMF) is the major source of these isotopes. Based on the low calculated doses, a quantitative As Low As Reasonably Achievable (ALARA) analysis of disposal options was not deemed necessary in this iteration of the CA.

The results of the CA clearly indicate that continued disposal of low-level waste in the Saltstone and EAV facilities, consistent with their respective radiological performance assessments, will have no adverse impact on future members of the public.

2.4.1 Points of Assessment

The point of assessment for the CA is the geographic location that hypothetical future members of the public (both individuals and populations) can reasonably be expected to access, taking into consideration any natural barriers and land use planning for the SRS and vicinity. Two media could be contaminated by radionuclides contained in facilities located in the GSA: groundwater and surface water that is recharged by groundwater. Contamination of the ground surface is not expected and thus air and soil are not routes of potential contaminant transport. A more in-depth discussion of transport pathways is provided in Section 4.3.

Table 1-1 Composite Analysis Results – Upper Three Runs at Savannah River

Radionuclide	Time of Peak Dose (years)	Fish Ingestion Dose (mrem/yr)	Shore-line Dose (mrem/yr)	Swimming Dose (mrem/yr)	Boating Dose (mrem/yr)	Total Recreational Fishing Dose[1] (mrem/yr)	Water Ingestion Dose[2] (mrem/yr)	All Pathways Dose[3] (mrem/yr)	Collective Dose[4] (person-rem/yr)
^{3}H	62	5.9×10^{-02}	0.0	1.1×10^{-3}	0.0	6.0×10^{-2}	1.2	1.3	3.2×10^{-02}
^{14}C	728	1.8	0.0	0.0	0.0	1.8	7.4×10^{-03}	1.8	4.5×10^{-02}
^{237}Np	685	2.2×10^{-02}	5.2×10^{-05}	8.2×10^{-09}	9.6×10^{-09}	2.2×10^{-02}	4.3×10^{-02}	6.5×10^{-02}	1.6×10^{-03}
^{233}U	545	3.0×10^{-04}	7.7×10^{-07}	8.1×10^{-11}	9.6×10^{-11}	3.0×10^{-04}	2.9×10^{-03}	3.2×10^{-03}	8.0×10^{-05}
^{234}U	383	6.9×10^{-03}	3.1×10^{-05}	1.3×10^{-09}	1.6×10^{-09}	6.9×10^{-03}	6.8×10^{-02}	7.5×10^{-02}	1.9×10^{-03}
^{235}U	548	2.4×10^{-04}	2.4×10^{-04}	4.5×10^{-08}	5.3×10^{-08}	4.8×10^{-04}	2.4×10^{-03}	2.9×10^{-03}	7.2×10^{-05}
^{236}U	549	9.6×10^{-04}	3.9×10^{-06}	1.5×10^{-10}	1.8×10^{-10}	9.6×10^{-04}	9.4×10^{-03}	1.0×10^{-02}	2.5×10^{-04}
^{238}U	551	6.5×10^{-03}	2.6×10^{-05}	9.7×10^{-10}	1.1×10^{-09}	6.5×10^{-03}	6.3×10^{-02}	7.0×10^{-02}	1.8×10^{-03}

Notes:

1. The recreational fishing scenario, which includes fish ingestion, shoreline exposure, and boating, is used to estimate the maximum dose to a hypothetical individual.
2. The water ingestion dose, assuming consumption of one liter of untreated Upper Three Runs water per day, was computed to estimate collective dose to a hypothetical population.
3. To estimate population dose, it was assumed that each person in the hypothetical population would be exposed per the recreational fishing scenario and the drinking water scenario.
4. The hypothetical population is assumed to consist of 25 adult persons.

UTR and Fourmile Branch (FMB) form the northern and southern boundaries of the GSA (Figure 2.3-2). Both of these streams remain on site until they reach the Savannah River. Both of the streams cut into the uppermost aquifer subject to contamination from the GSA (Section 2.3.5). UTR also cuts into the Gordon aquifer, which is the lowermost of the two aquifers subject to contamination from the GSA. FMB is upgradient with respect to the GSA for the Gordon aquifer. The Gordon aquifer flows northwestward under FMB towards UTR. Thus, these streams will intercept all plumes of groundwater contamination emanating from the GSA. The SRS Future Use Plan (Attachment 2) indicates that release of the site to the public for unrestricted use will not occur over the time period of this analysis; therefore, on-site use by the public of potentially-contaminated groundwater is not a reasonable expectation.

Contaminated surface water is considered a potential source of exposure to a hypothetical future member of the public in this analysis. All contaminated groundwater will discharge to streams that bound the GSA. Water infiltrating the disposal facilities under consideration, Saltstone and the EAV, will discharge to UTR. While land-use plans are expected to restrict use of the SRS during the time period of the analysis, the confluence of on-site streams with the Savannah River poses a potential means of public access to contaminated environmental media. Thus, the point of assessment for this analysis is the mouth of UTR at the Savannah River.

Even though land-use planning envisions the continual control of the SRS, consistent with current boundaries, it is conceivable that a member of the public could gain access to the mouth of UTR by boat from the Savannah River. Thus, the mouth of UTR, at the furthest downstream point where stream water remains undiluted with Savannah River water, is the point for the assessment of potential dose to a hypothetical future member of the public.

For the assessment of potential collective dose to future populations, this analysis conservatively assumed that a population of 25 individuals received their drinking water (1 L per day per person) from the mouth of UTR. This population was also assumed to take part in activities defined for the maximally exposed individual (i.e., recreational fishing).

7.1 Comparison With Dose Limits and Constraints

The peak dose to a maximally exposed individual within the performance time period of 1000 years is estimated to be approximately 1.8 mrem/yr at the mouth of UTR. This estimated dose is well below the primary dose limit of 100 mrem/year established by USDOE Order 5400.5 (Section 2.4.3).

In the CA Guidance document, an additional dose constraint of 30 mrem/year is used "to ensure that no single source, practice, or pathway uses an extraordinary portion of the primary dose limit." The estimated dose in this CA is also below this constraint. Thus an options analysis is not required.

7.2 Principal Sources Contributing to Dose

The major radionuclides contributing to dose in the Composite Analysis are ^{14}C, ^{3}H, ^{237}Np, and isotopes of uranium (Section 5.5). The predominant source of these radionuclides is the MWMF, as indicated in Table 4.4-5.

The active low-level waste disposal facilities addressed in the CA, the EAV and the Saltstone facilities, are relatively insignificant sources of these radionuclides. The saltstone wasteform and the naval reactor components disposed in the EAV resist leaching and the vaults control infiltration of water into the wastes. These barriers to leaching reduce and delay the release of radionuclides to the subsurface environment. Predicted releases from these facilities during the first 1000 years after disposal are therefore negligible and the doses attributable to these facilities during this time period are insignificant relative to the total dose calculated for the CA.

7.3 Effects of Sensitivities

The sensitivity analysis (Section 6) shows that the results of the CA are most sensitive to the selection of the point of assessment. The point of assessment was derived from the SRS Future Use Plan (Attachment 2) which projects no unrestricted use of any of the current SRS lands. Near the GSA, the dose to the hypothetical maximally exposed member of the public would only be 2.4 mrem/year. Given the conservatism of the current analysis, potential doses to members of the public, even on UTR, are unlikely to exceed the dose constraint.

7.4 ALARA Considerations

The maximum peak dose of 1.8 mrem/yr calculated for the GSA in this analysis is considerably lower than the dose limit (100 mrem/yr) and dose constraint (30 mrem/yr). Thus, a quantitative ALARA analysis of options for reducing future doses may not be warranted. Such an assessment analyzes the cost-benefit of dose reduction; however, if the estimated cost of the analysis alone is likely to exceed the monetary equivalent of reducing the dose to zero, then the assessment is not warranted.

To determine whether a quantitative ALARA analysis is warranted, a monetary equivalence of potential dose reduction must be assigned. The USDOE recommends an equivalence in the range from $1,000 to $10,000 per person-rem reduced. Thus, calculation of population doses associated with the GSA was required to make this determination.

7.4.1 Population Doses

The population dose calculated for the ALARA process in this CA conservatively assumes that a hypothetical population of 25 adult individuals is exposed to water at the mouth of UTR. These persons are assumed to obtain their drinking water (1 L per day) from UTR. They are also assumed to carry out the activities in the recreational fishing scenario used for the maximally exposed individual.

Population doses were calculated using the LADTAP XL spreadsheet model (Hamby 1991a), described in Sections 5.4 and 5.5. The peak dose to the hypothetical population was 0.045 person-rem/yr.

7.4.2 ALARA Analysis

An ALARA analysis calculates the cost of actions that could be taken to reduce population dose versus the benefit of the dose reduction. However, when maximum individual doses are calculated to be below the 30 mrem/yr dose constraint in a CA, the question becomes whether the cost of a quantitative ALARA analysis is justified.

In this CA of the GSA, the maximum individual dose was calculated to be 1.8 mrem/yr for all radionuclides: well below the 30 mrem/yr dose constraint. To evaluate whether an ALARA analysis is warranted, population doses were also calculated. The maximum population dose was calculated to be approximately 0.045 person-rem/yr. Using the USDOE's estimate of monetary equivalence for dose reduction of between $1,000 to $10,000 per person-rem potentially avoided, a maximum cost of dose reduction of $450 is calculated. This maximum cost is calculated assuming dose is reduced to zero, at an upper-end cost of $10,000 per person-rem and assuming a dose integration time of one year. The many conservative assumptions that went into estimation of population dose further maximizes this cost. The cost of the present analysis of the base case exceeds this maximum cost, and thus the cost of evaluating the impact of more than one option for the GSA is expected to greatly exceed the maximum cost. Based on this information, an ALARA analysis is not warranted because of the low population dose potentially associated with the presence of subsurface radionuclides in the GSA.

The conclusion that an ALARA analysis is not warranted is strongly influenced by the selection of the time over which population dose is integrated. USDOE guidance on the dose integration time has not been issued. Due to the conservative assumptions used in this CA, a one-year integration time was selected.

7.5 Options Analysis

The calculated dose to the hypothetical maximally exposed member of the public of 1.8 mrem/yr is below the dose constraint of 30 mrem/yr. Thus, per USDOE guidance, an options analysis is not required.

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2.0 Condition 2

Uncertainty and Sensitivity Analysis – Perform a sensitivity analysis on the radionuclides important to the composite analysis and flux rates and on the hydrologic model including the groundwater divide and the model boundary conditions. Perform an uncertainty analysis on the inventory, flux rates, and resultant dose calculations for the radionuclides important to the composite analysis.

2.1 Sensitivity Analysis - Radionuclides

The sensitivity analysis on the radionuclides important to the CA and flux rates is integral to the uncertainty analysis, which is presented at the end of this section, and is not reproduced here.

2.2 Sensitivity Analysis – Hydrologic Model

The additional sensitivity analysis on the hydrologic model focused on the groundwater divide (i.e., impact of remediation activities, bounding estimates of dose resulting from all radionuclides migrating to either stream) and the model boundary conditions. Each of the investigations is presented below.

2.2.1 Impact of Remediation Activities on the Groundwater Divide Between Fourmile Branch and Upper Three Runs within the General Separations Area

The groundwater divide between FMB and UTR within the "upper" aquifer zone (water table) based on groundwater flow simulations (Flach and Harris, 1997) is depicted in Figure 2.2-1. The shaded arrows in Figure 2.2-1 are constant in length, and therefore only show groundwater flow direction in the horizontal plane. The divide can be affected by large-scale remediation activities that alter surface recharge or involve groundwater pumping. Candidates include the interim surface cap for the Old Burial Ground (OBG) applied in 1997, and pump-treat-reinject (PTR) operations for the F- and H-Area seepage basins scheduled for 1998. Changes to groundwater flow following the OBG cap and long-term F- and H-Area PTR operation were simulated by Flach (1998). The modeling results described in Flach (1998) can be used to investigate impacts to the groundwater divide. Figure 2.2-2 shows predicted steady-state groundwater flow directions after the three large-scale remediation operations have been in place for several years. The heavy solid line shows the groundwater divide before remediation activities, and the heavy dotted line depicts the divide after long-term remediation. Groundwater injection in F- and H-Area is seen to move the divide toward FMB, whereas the decreased surface recharge over the OBG moves the divide away from FMB towards UTR. Figures 2.2-3 and 2.2-4 are the same as Figures 2.2-1 and 2.2-2 respectively, except that vectors proportional to the rate of groundwater flow are shown. These figures better illustrate three-dimensional aspects of the overall groundwater flow field. Near the groundwater divide, there is a strong downward flow component. Near groundwater discharge areas, the lateral flow components dominate.

2.2.2 Bounding Estimate of All General Separations Area Contaminants Migrating to Either of the Streams

The sensitivity of results calculated in the SRS CA to the location of the groundwater divide was discussed qualitatively in Section 6.4 of the CA. Following is a more quantitative analysis.

Groundwater flow directions in Upper Three Runs aquifer unit, "upper" zone

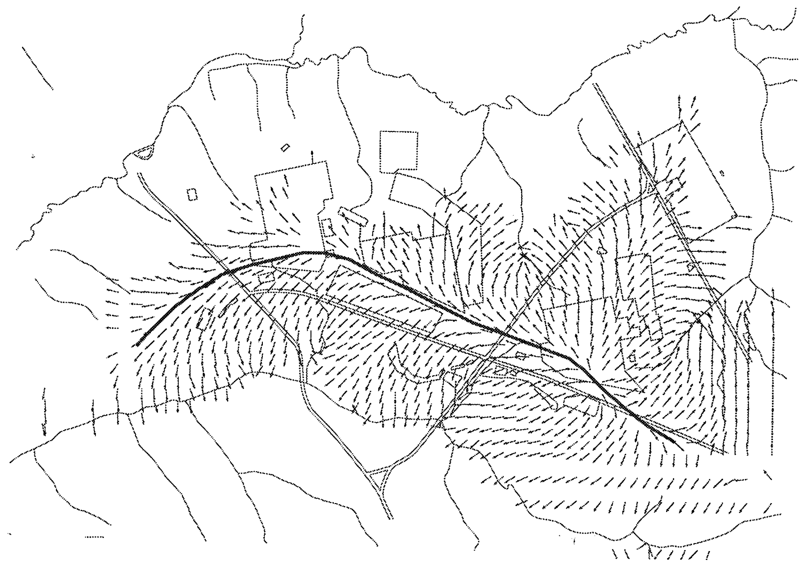

Figure 2.2-1 Simulated Groundwater Divide *Before* OBG Cap and F- and H-Area PTR Systems; Flow Direction Illustrated by *Constant* Length Vectors

Groundwater flow directions in Upper Three Runs aquifer unit, "upper" zone

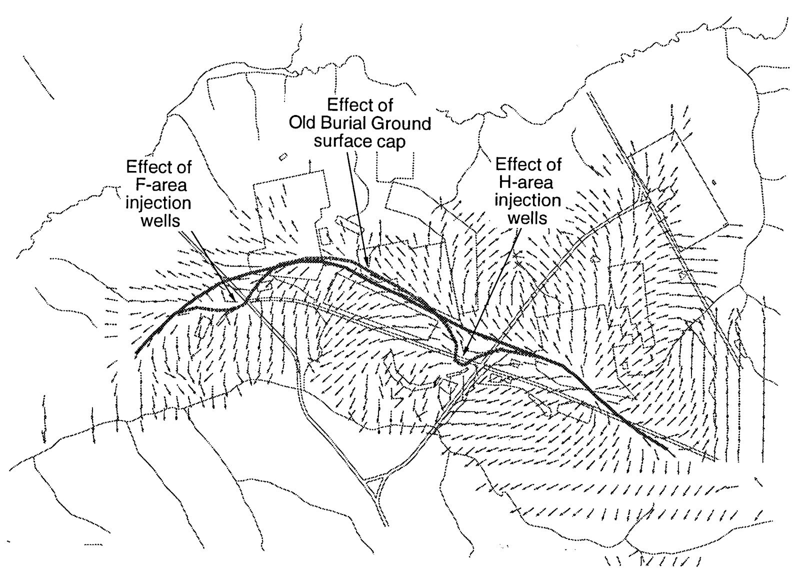

Figure 2.2-2 Simulated Groundwater Divide *After* OBG Cap and F- and H-Area PTR Systems; Flow Direction Illustrated by *Constant* Length Vectors

Groundwater flow directions in Upper Three Runs aquifer unit, "upper" zone

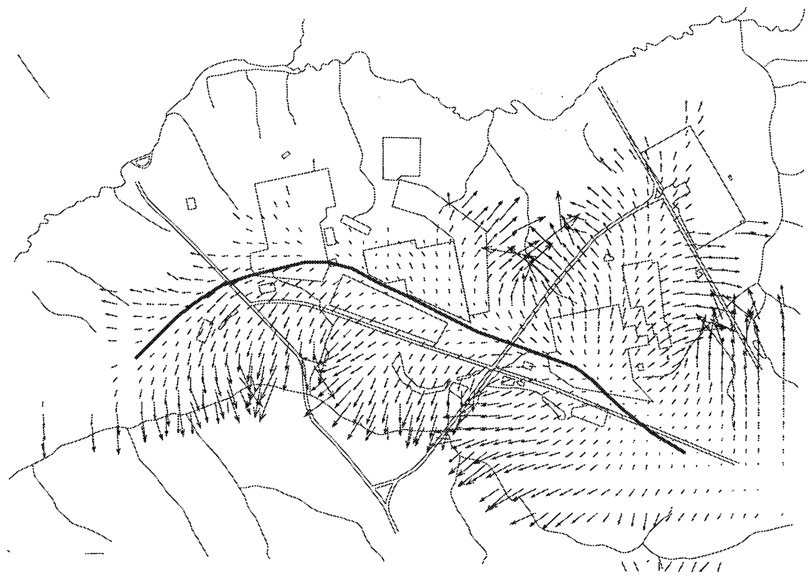

Figure 2.2-3 Simulated Groundwater Divide *Before* OBG Cap and F- and H-Area PTR Systems; Flow Direction Illustrated by *Proportional* Length Vectors

Groundwater flow directions in Upper Three Runs aquifer unit, "upper" zone

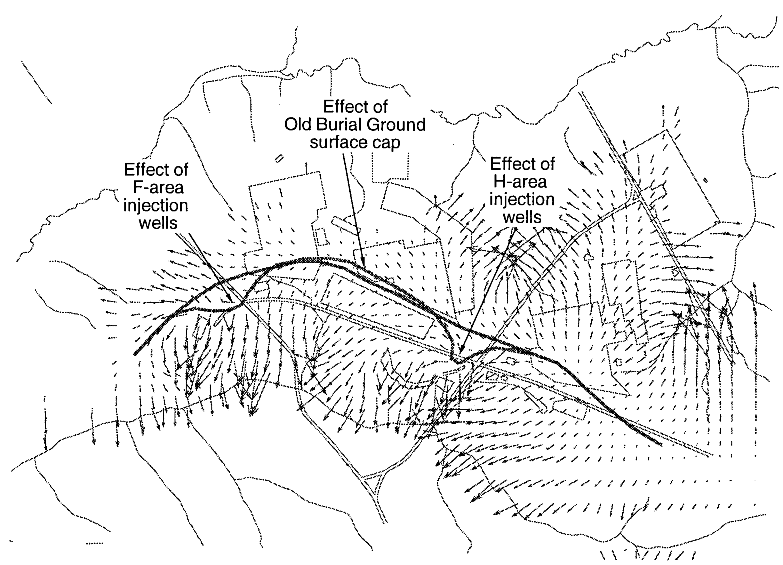

Figure 2.2-4 Simulated Groundwater Divide *After* OBG Cap and F- and H-Area PTR Systems; Flow Direction Illustrated by *Proportional* Length Vectors

In the CA, the present location of the groundwater divide, which lies between the MWMF and the OBG, was assumed to be constant for the entire period of analysis. To illustrate the sensitivity of the analysis results to the location of the divide, doses were estimated assuming that all contaminants released within the GSA would migrate to either of the two surface streams, UTR and FMB.

Doses in the CA are calculated from the concentration of radionuclides in the streams. Radionuclide concentrations are calculated from the flux of radionuclides to one of the streams and the average volumetric flow of the streams. The calculated peak fluxes to the streams are presented in Table 5.3-1 of the CA. Calculated doses at the stream mouths are presented in Table 5.5-2; doses calculated at the GSA are presented in Table 6.1-1.

The doses resulting from the assumption that all radionuclides would migrate to only one of the streams were calculated by ratio of the CA dose to the CA flux to one stream multiplied by the sum of the CA fluxes to each stream. This method over estimates the total flux to a given stream because it does not take into account the longer flow path from the disposal area to one of the streams that was used in the original CA calculation (e.g., tritium flux calculated in the CA to FMB will be attributed to UTR). This effect will be most pronounced for tritium because of its short half-life.

For example, the tritium dose due to drinking water from UTR at the GSA, assuming all of the sources migrate to UTR, was calculated according to the following equation:

$$\text{Dose}_{\text{UTR+FMB}} = \text{Flux}_{\text{UTR+FMB}} * \text{Dose}_{\text{UTR}} \; / \; \text{Flux}_{\text{UTR}}$$

where $\text{Dose}_{\text{UTR+FMB}}$ is the dose calculated from all sources,
$\text{Flux}_{\text{UTR+FMB}}$ is the sum of the fluxes to each of the streams from Table 5.3-1,
Dose_{UTR} is the dose due to tritium from only those sources that drain to UTR from Table 6.1-1, and
Flux_{UTR} is the flux of tritium to UTR from only those sources that drain to UTR from Table 5.3-1

$$\text{Dose}_{\text{UTR+FMB}} = (1.05 \times 10^4 + 6.34 \times 10^3) * 2.4 \; / \; 1.05 \times 10^4$$

$$\text{Dose}_{\text{UTR+FMB}} = 3.85 \text{ mrem/year}$$

Estimated doses from the significant radionuclides are presented in Table 2.2-1.

Dose calculated from drinking water at the GSA should be compared with values presented in Table 6.1-1. Doses calculated from the recreation scenario at the stream mouths should be compared with values presented for all pathways in Table 5.5-2. The increase in calculated dose is greatest for FMB due to the lower flow rate (24 cfs) compared with that in UTR (217 cfs).

Although the dose calculated for drinking water from FMB in this sensitivity analysis is large, 64 mrem/year, it is incredible that this dose would ever be realized. First, as discussed in the accompanying analysis of the factors affecting the location of the groundwater divide, the migration of all contaminants to only one stream is not credible. Second, the large dose calculated is due to tritium. As stated above, no correction was made for the decay that would take place due to the longer flow path if this scenario were to happen. Third, the dose due to tritium occurs very quickly (in Table 5.5-2 of the CA, the peak dose from tritium occurs at 62 years in UTR and 61 years in FMB). For the dose to be realized, the scenario of someone

Table 2.2-1 Estimated Doses from Significant Radionuclides

Radionuclide	Estimated Dose From Drinking UTR Water at GSA (mrem/year)	Estimated Dose From Drinking FMB Water at GSA (mrem/year)	Estimated Dose from Recreation Scenario at UTR Mouth, (mrem/year)	Estimated Dose from Recreation Scenario at FMB Mouth (mrem/year)
^{3}H	3.85	6.37×10^{1}	9.62×10^{-02}	8.50×10^{-01}
^{14}C	2.73×10^{-02}	3.99×10^{-01}	3.28	2.88×10^{1}
^{237}Np	3.84×10^{-01}	5.95	9.71×10^{-02}	9.05×10^{-01}
^{234}U	2.05×10^{-01}	3.27	1.09×10^{-02}	9.81×10^{-02}
^{235}U	9.26×10^{-03}	1.51×10^{-01}	9.67×10^{-04}	8.74×10^{-03}
^{236}U	3.82×10^{-02}	6.24×10^{-01}	2.04×10^{-03}	1.87×10^{-02}
^{238}U	2.33×10^{-01}	3.71	1.26×10^{-02}	1.15×10^{-01}

obtaining drinking water from FMB within 62 years would have to occur. This is incredible because of the land use planning discussed in the CA and because waste management and environmental remediation activities at SRS will continue for several more decades.

2.2.3 Model Boundary Conditions

Figure 2.2-5 is a hand-drawn (not produced by computer), large-scale, potentiometric map of the Gordon aquifer that incorporates well and stream water level data with a conceptual understanding of groundwater flow (Hiergesell, 1999). The map includes the updip continuation of the Gordon aquifer as the Steed Pond aquifer north of UTR. The Gordon aquifer is recharged from the overlying UTR aquifer, and by lateral flow into the domain across the east and south boundaries of GSA. The Gordon aquifer is discharged by UTR along the north boundary of the GSA and lateral outflow along the west boundary. Relative to recharge and lateral flows, net groundwater flow through the underlying Meyers Branch confining system is small. Simulated groundwater flow in the Gordon aquifer (CA Figure 5.1-20) agrees with Figure 2.2-5 and (CA Figure 5.1-14) which are based on measured water levels.

The no-flow boundary terminology used in discussions with the Review Team is confusing and has been subsequently clarified in WSRC-TR-96-0399, Rev. 1. The Gordon aquifer is assumed to completely discharge to UTR from both sides of the stream, because the stream bed and recent alluvium deeply incise the aquifer. Therefore, groundwater does not flow beneath UTR from one side to the other. UTR functions as a groundwater flow divide for the Gordon aquifer, and is a no-flow boundary in this sense.

Figure 2.2-6 schematically illustrates how model boundary conditions are defined along no-flow boundaries, such as UTR. As groundwater flow approaches the groundwater divide created by a stream, the flow turns upward and discharges to ground surface at seepage faces comprising the stream bed and/or adjoining wetland areas. This physical situation is reproduced in the model by assigning a drain boundary condition to the uppermost nodal layer and a no-flow boundary condition to underlying nodes, as shown in Figure 2.2-6. Therefore, no-flow boundaries actually consist of both drain and no-flow boundary conditions.

Figure 2.2-7 is a hand-drawn (not produced by computer), large-scale, potentiometric map of the water table that incorporates well and stream water level data with a conceptual understanding of groundwater flow (Hiergesell, 1998). In the GSA, the water table resides in the UTR aquifer. Alluvial deposits along FMB deeply incise the "lower" aquifer zone of UTR aquifer. FMB is assumed to completely drain the UTR aquifer from each side, such that FMB functions as a groundwater divide as shown in Figure 2.2-7. Drain boundary conditions are specified along FMB for surface nodes while no-flow conditions are prescribed for underlying nodes. Simulated flow agrees with Figure 2.2-7 and CA Figure 5.1-13, which are based on measured water and stream levels.

The no-flow boundary between McQueen Branch and FMB can be better justified by referring to Figure 2.2-7, which shows the water table over a larger area than Figure 5.1-13. As shown in Figure 2.2-7, the eastern, no-flow, boundary of the flow model crosses potentiometric lines at nearly right angles. Although there is probably some inward flux across this boundary, the head gradients are very small and can be neglected. Note that the simulated water table (CA Figure 5.1-11) agrees well with Figure 2.2-7, including along the eastern boundary.

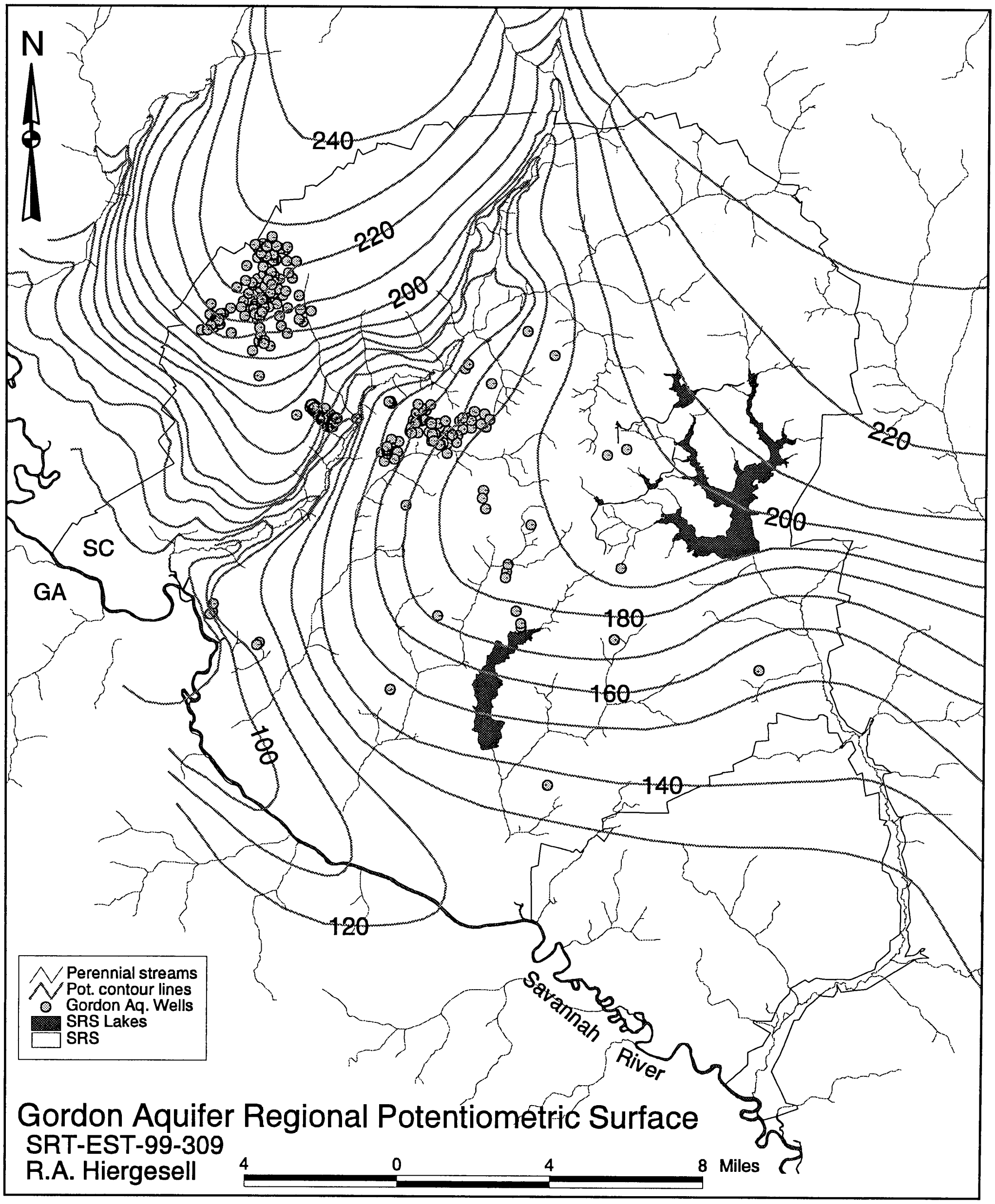

Figure 2.2-5 Gordon Aquifer Regional Potentiometric Surface

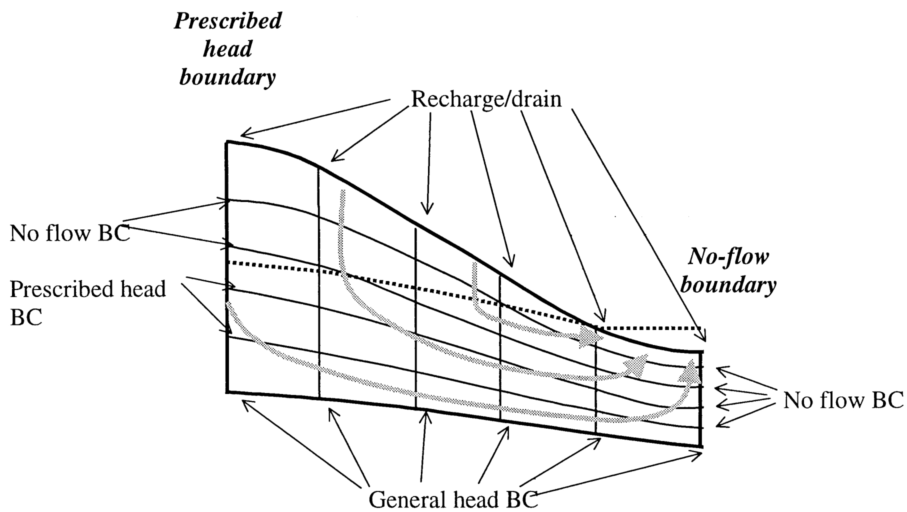

Figure 2.2-6 Schematic Diagram of No-flow and Prescribed Head Boundary Condition Specification

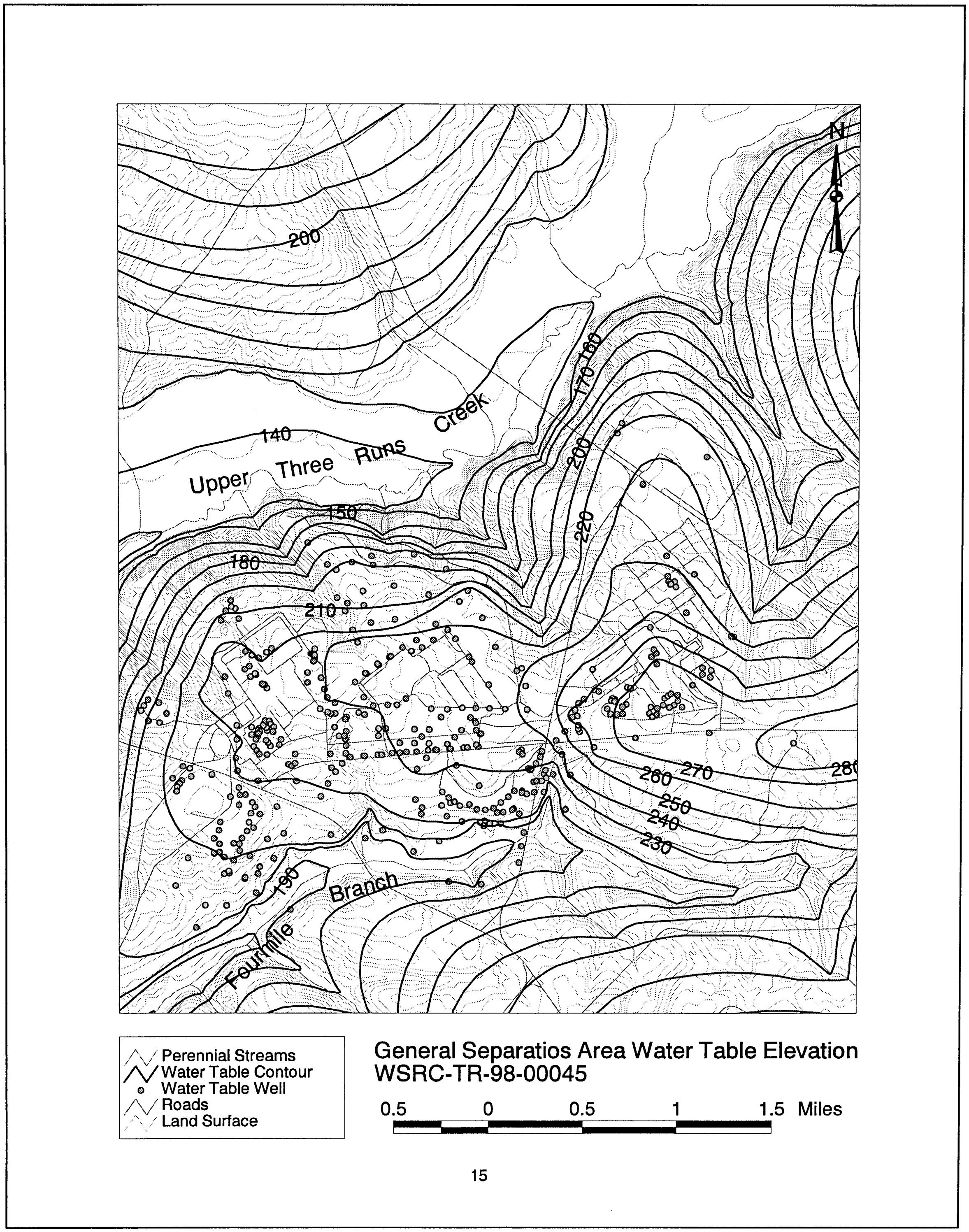

Figure 2.2-7 General Separations Area Water Table Elevation

2.2.4 Uncertainty Analysis

As part of the response to Condition 2, the following uncertainty analysis on the inventory, flux rates, and resultant dose calculations for the radionuclides important to the CA was performed. It is presented as Section 6.6 of the CA.

6.6 Uncertainty Analysis on Inventory

An uncertainty analysis on inventory was conducted for radionuclides important to the CA. Two general screening processes were employed to determine the most important radionuclides and their significant sources. First, dose results were screened to determine the most important radionuclides at each stream. Second, inventories and contaminant fluxes to the water table were screened to identify the significant sources of the most important radionuclides.

After screening was completed, sampling from probability density functions (PDFs) resulted in inventory variations at significant sources. The first realization set of inventory variations was generated by combining the first sample inventory from each source. Repeating this process of combining the nth sample inventory from each source generated one thousand realization sets. Each set of inventory variations was used to generate variations in contaminant fluxes to the water table, fluxes to streams, and hypothetical doses at the streams. Peak doses from each inventory variation were plotted and compared with the base case peak dose.

6.6.1 Dose Screening to Determine Important Radionuclides and Associated Streams

The radionuclides most important to the CA were determined by comparing doses (from Table 5.5-2) with a threshold value of one percent of the 30 mrem/yr dose constraint (i.e., 0.30 mrem/yr) established for SRS (see Section 2.4.3) . This step indicated that three radionuclides, as shown in Table 6.6-1, are important. All three contaminants are important at FMB, but only ^{14}C is important at UTR.

6.6.2 Inventory and Water Table Flux Screening to Determine Significant Sources

A two-step screening process determined the significant radionuclide sources of the important radionuclides. First, facilities with relatively low inventories were eliminated from further consideration. Second, facilities with relatively low contaminant fluxes to the water table were eliminated.

<u>Inventory Screening</u>

Inventory screening levels were selected using professional judgement. The bases for the selections were the relative magnitude of the facility inventory and the relative risk presented by each of the radionuclides.

Tritium inventories at all facilities listed in Table 4.4-2 are plotted in Figure 6.6-1. The highest inventory is MWMF with an order of magnitude of 1×10^6 Ci. The threshold was set four orders of magnitude below this level at 1×10^2 Ci. All facilities with inventories below 1×10^2 Ci were screened out except for F Canyon, which was retained because its 68 Ci inventory was only slightly below the threshold.

Table 6.6-1 Radionuclides Exceeding Threshold Dose of 0.3 mrem/yr

Radionuclide	Dose (mrem/yr)[1]	Stream
^{3}H	0.32	FMB
^{14}C	13.00	FMB
^{14}C	1.80	UTR
^{237}Np	0.70	FMB

Notes:
[1] From Table 5.5-2

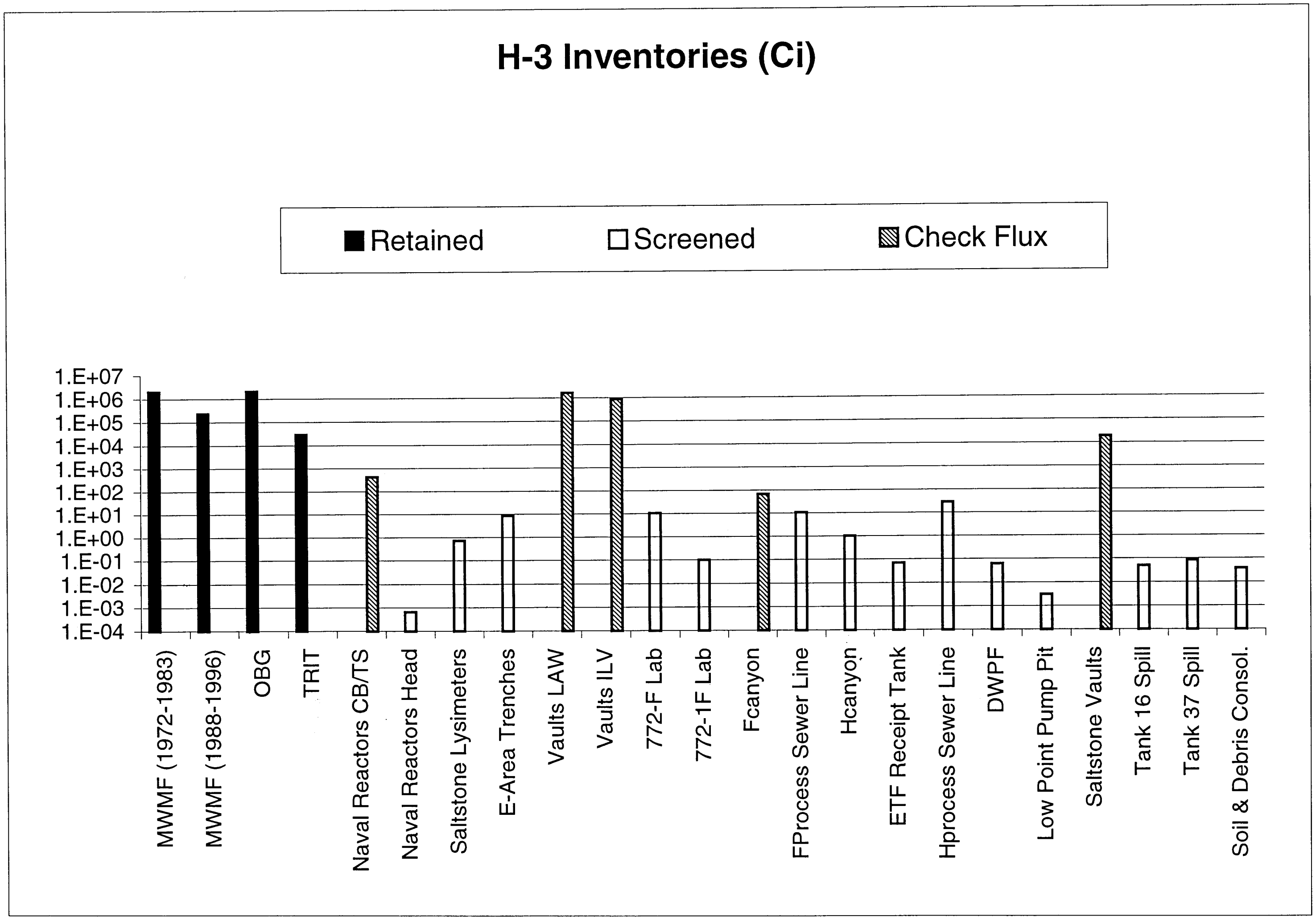

Figure 6.6-1 ^{3}H **Inventories at All Facilities**

Sources represented by clear bars in Figure 6.6-1 were eliminated during the inventory-screening phase. Sources with crosshatched bars were retained during the first screening phase. Based on flux to water table curves shown in Figures 5.2-3 through 5.2-22, elimination of sources with crosshatched bars was expected during the second screening phase. Sources with solid bars were retained during the first screening phase and their elimination was not expected during subsequent screening. Bar attributes for subsequent inventory figures are identical to the bar attributes for Figure 6.6-1.

^{14}C inventories at all facilities listed in Table 4.4-2 are plotted in Figure 6.6-2. The highest inventory is OBG with an order of magnitude of 1×10^{3} Ci. The threshold was set three orders of magnitude below this level at 1 Ci. All facilities with inventories below 1×10 Ci were screened out.

^{237}Np inventories at all facilities listed in Table 4.4-2 are plotted in Figure 6.6-3. The highest inventory, 12 Ci, is found in 235-F, the Plutonium Fabrication Facility. The threshold was set five orders of magnitude below this level at 1×10^{-4} Ci. All facilities with inventories below 1×10^{-4} Ci were screened out.

^{237}Np is a part of a decay chain that includes ^{241}Pu and ^{241}Am. Inventories for ^{241}Pu and ^{241}Am are included in Figures 6.6-4 and 6.6-5, respectively. These two figures were used only to add to the list of ^{237}Np facilities to consider in subsequent screening and analysis. The subsequent screening for ^{241}Pu and ^{241}Am was based on the flux of ^{237}Np to the water table.

In Figure 6.6-4 the highest inventory for ^{241}Pu is OBG with an order of magnitude of 1×10^{4} Ci. The threshold was set four orders of magnitude below this level at 1 Ci. The list of facilities with inventories above 1 Ci was compared with the list of retained ^{237}Np inventory facilities. Facilities added to the ^{237}Np inventory list were as follows:

- Naval Reactors
- 772-F Laboratory
- Tanks 17-20
- Tanks 25-28.

In Figure 6.6-5, the highest inventory for ^{241}Am is Tanks 21-24 with an order of magnitude of 1×10^{2} Ci. The threshold was set four orders of magnitude below this level at 1×10^{-2} Ci. The list of facilities with inventories above 1×10^{-2} Ci was compared with the list of retained ^{237}Np inventory facilities. Facilities added to the ^{237}Np inventory list were as follows:

- Naval Reactors (already added due to ^{241}Pu inventory)
- E-Area Trenches
- Soil and Debris Consolidation Facility
- Tanks 17-20 (already added due to ^{241}Pu inventory)
- Tanks 25-28 (already added due to ^{241}Pu inventory)
- H Process Sewer
- H Seepage Basin.

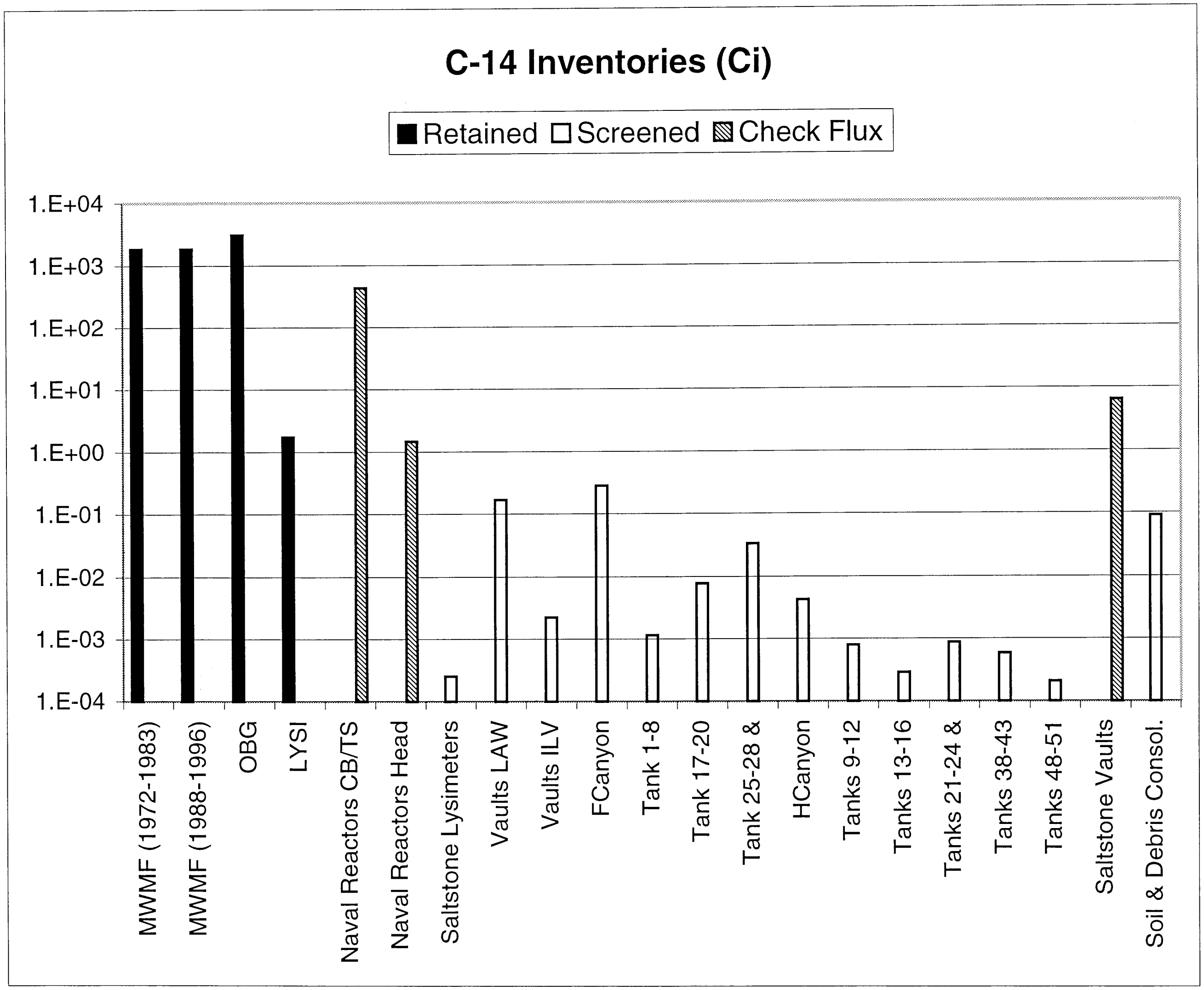

Figure 6.6-2 **^{14}C Inventories at All Facilities**

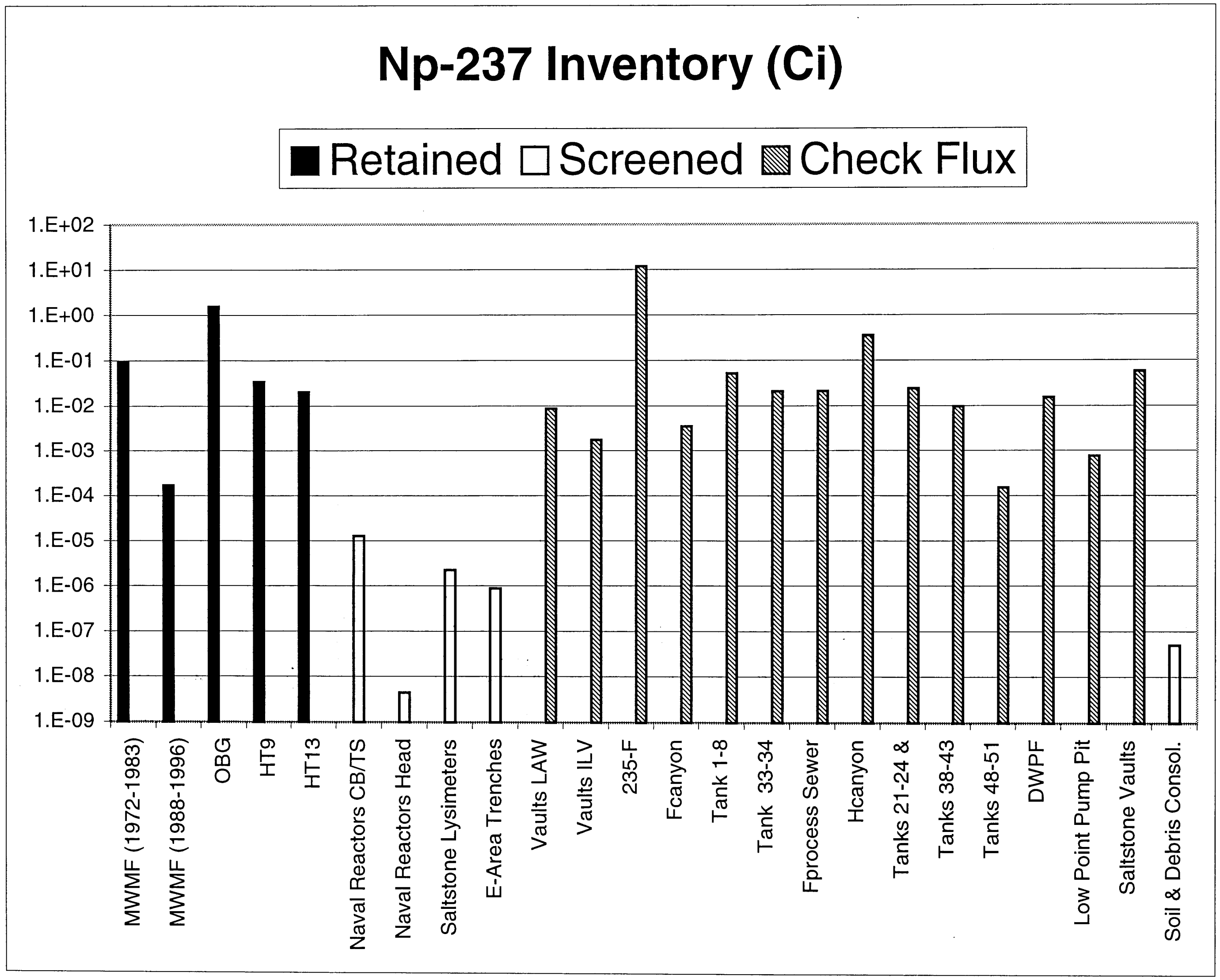

Figure 6.6-3 **^{237}Np Inventories at All Facilities**

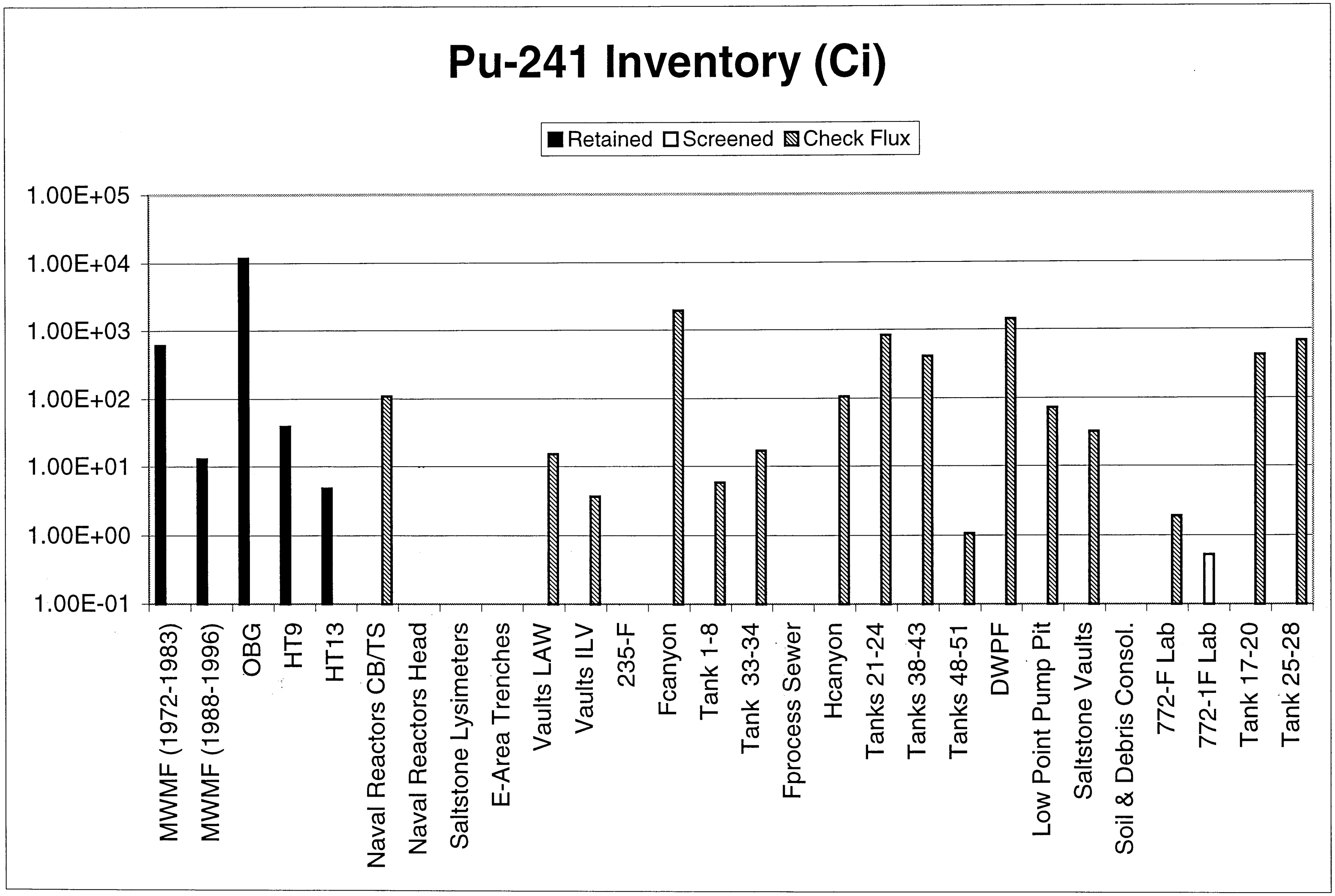

Figure 6.6-4 ^{241}**Pu Inventories at All Facilities**

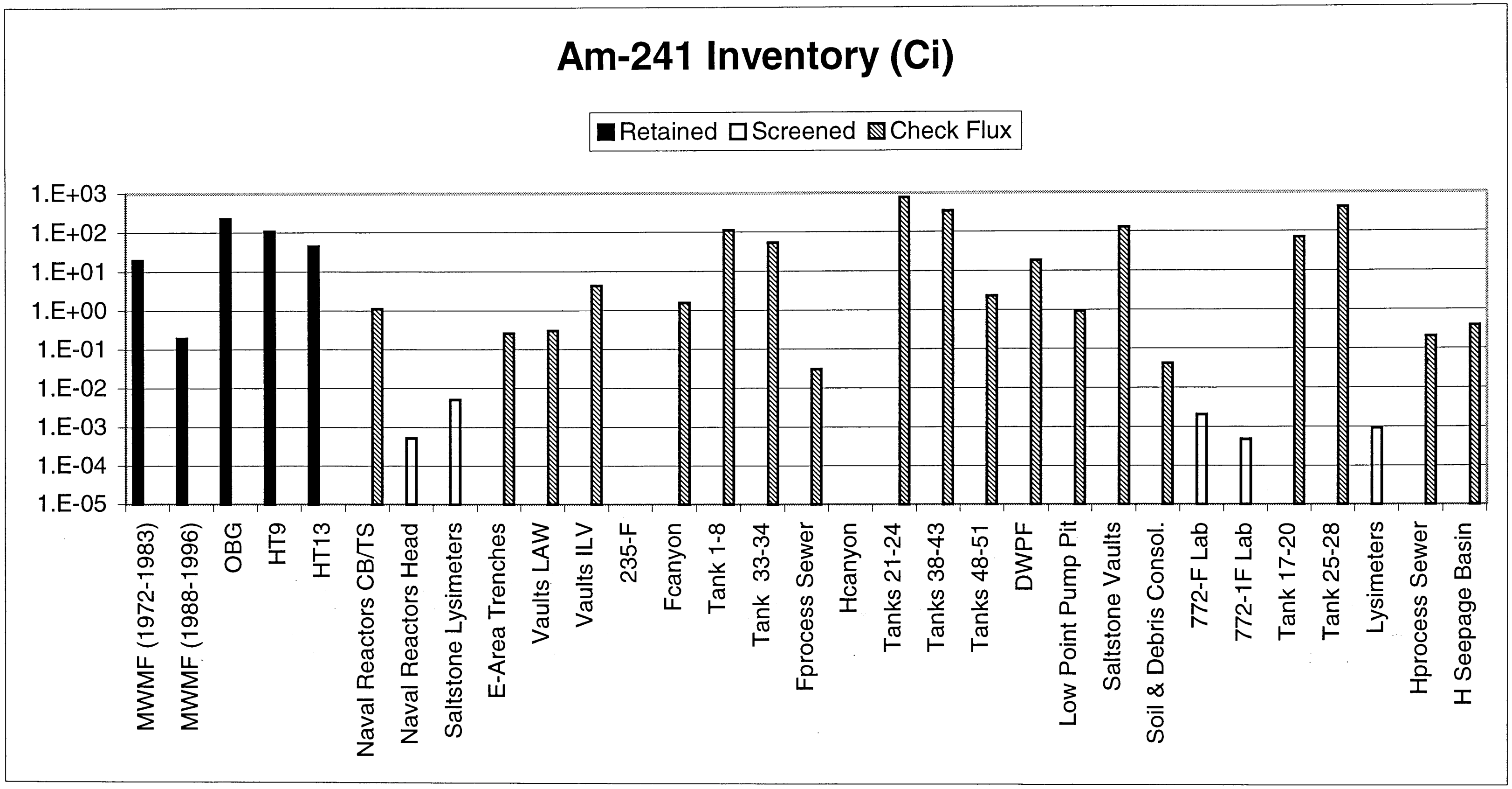

Figure 6.6-5 ^{241}**Am Inventories at All Facilities**

Contaminant Flux to the Water Table Screening

For the second screening step for significant sources, contaminant fluxes to the water table were examined. Each source with a peak flux less than .001 of the maximum peak flux of all sources (shown in bold in the Peak Flux tables below) was eliminated from future consideration. Fluxes to the water table derived from Table 4.4-5 are listed and plotted in the following tables and figures:

Contaminant	Listed	Plotted
^{3}H	Table 6.6-2	Figure 6.6-6
^{14}C	Table 6.6-3	Figure 6.6-7
^{237}Np	Table 6.6-4	Figure 6.6-8

In Tables 6.6-2 through 6.6-4, sources are grouped as to whether they were eliminated during the inventory-screening phase, eliminated during the contaminant flux screening phase, or survived both screening phases. Table 6.6-4 contains the inventories of ^{237}Np parent products for those facilities that were added to the list for future consideration based on their ^{241}Am or ^{241}Pu inventories.

Figures 6.6-6 through 6.6-8 only show the sources that passed the inventory screen. In these figures, sources with dark bars survived the water table contaminant flux screen, while sources without shading were eliminated.

Screening Summary

Screening based on flux at the water table produced two unexpected sources for retention. The 235-F and H Canyon facilities for ^{237}Np were the two exceptions. The 235-F facility had the highest ^{237}Np inventory by almost an order of magnitude leading to its retention. H Canyon had the third highest inventory, but it was retained only after slightly relaxing the screening criteria from 3.2×10^{-5} Ci/yr (based on .001 of HT13's 3.2×10^{-2} Ci/yr flux) to 1.0×10^{-5} Ci/yr.

All sources that were retained after screening are shown in Table 6.6-5 with the applicable contaminant. Table 6.6-5 also contains the data qualifier for the site that indicates the level of certainty associated with the information, with a lower value indicating more certainty.

6.6.3 Inventory Variation at Significant Sources

Approach

To examine uncertainty based on the inventory, typically a random sample is selected from an inventory probability density function (PDF). A sample is selected for each source's inventory and the samples are combined to form a realization set. That realization set feeds two computer models. The first model simulates transport of contaminants through the vadose zone, while the second model simulates transport of contaminants through the aquifer, producing a concentration and dose at each stream. Inventory sampling continues until each realization set has been selected and modeled, generating a set of doses at each stream. The set of doses forms the basis for determining the dose probabilities.

PDFs were developed for the twelve significant sources at eight locations, as presented in Table 6.6-5. The data qualifier provided a means to describe the inventory uncertainty. For data qualifiers 1 and 2, a lognormal PDF was assumed. For the rest of the sources, a logtriangular PDF was assumed. The base

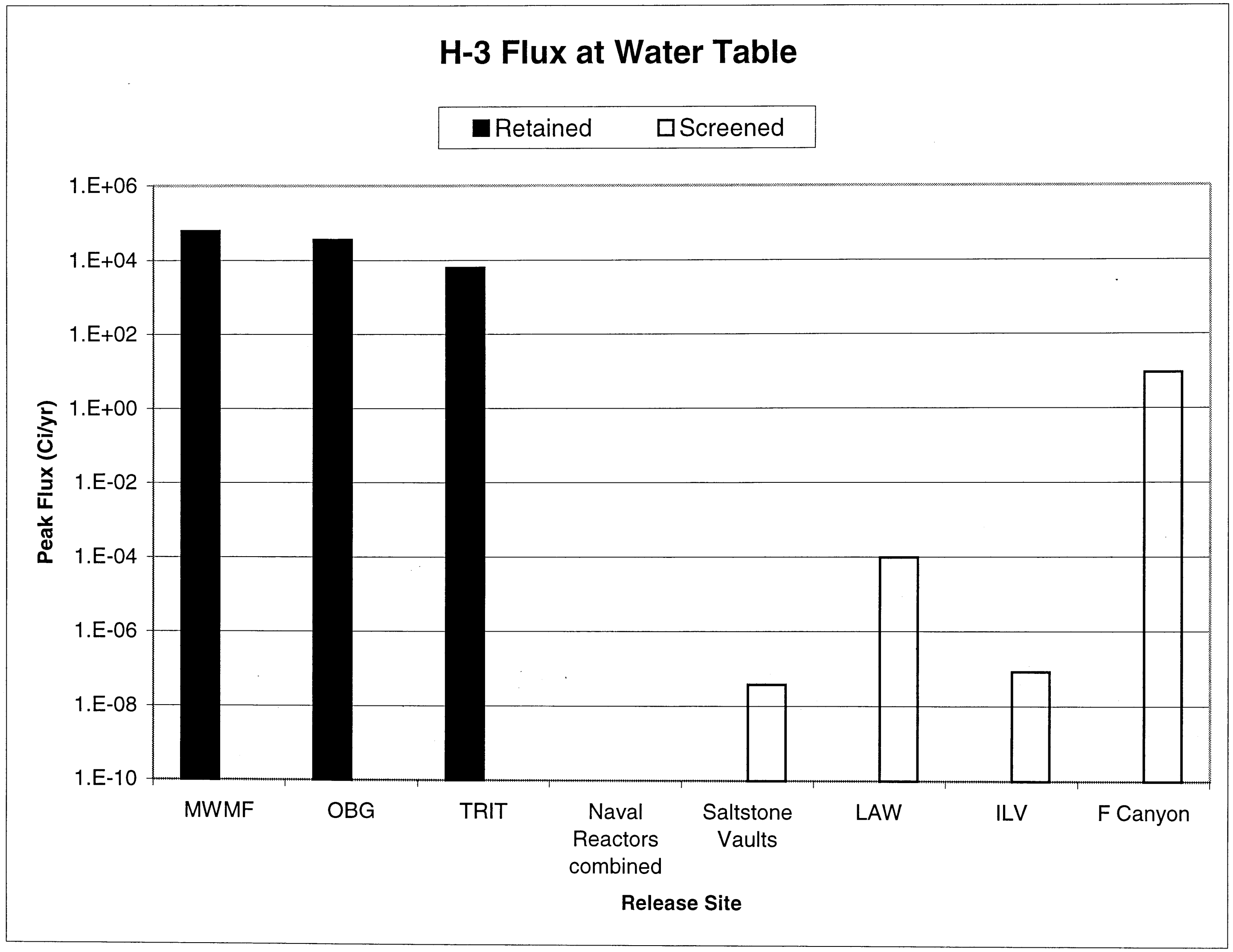

Figure 6.6-6 ^{3}H **Inventories for All Facilities Screened by Flux at Water Table**

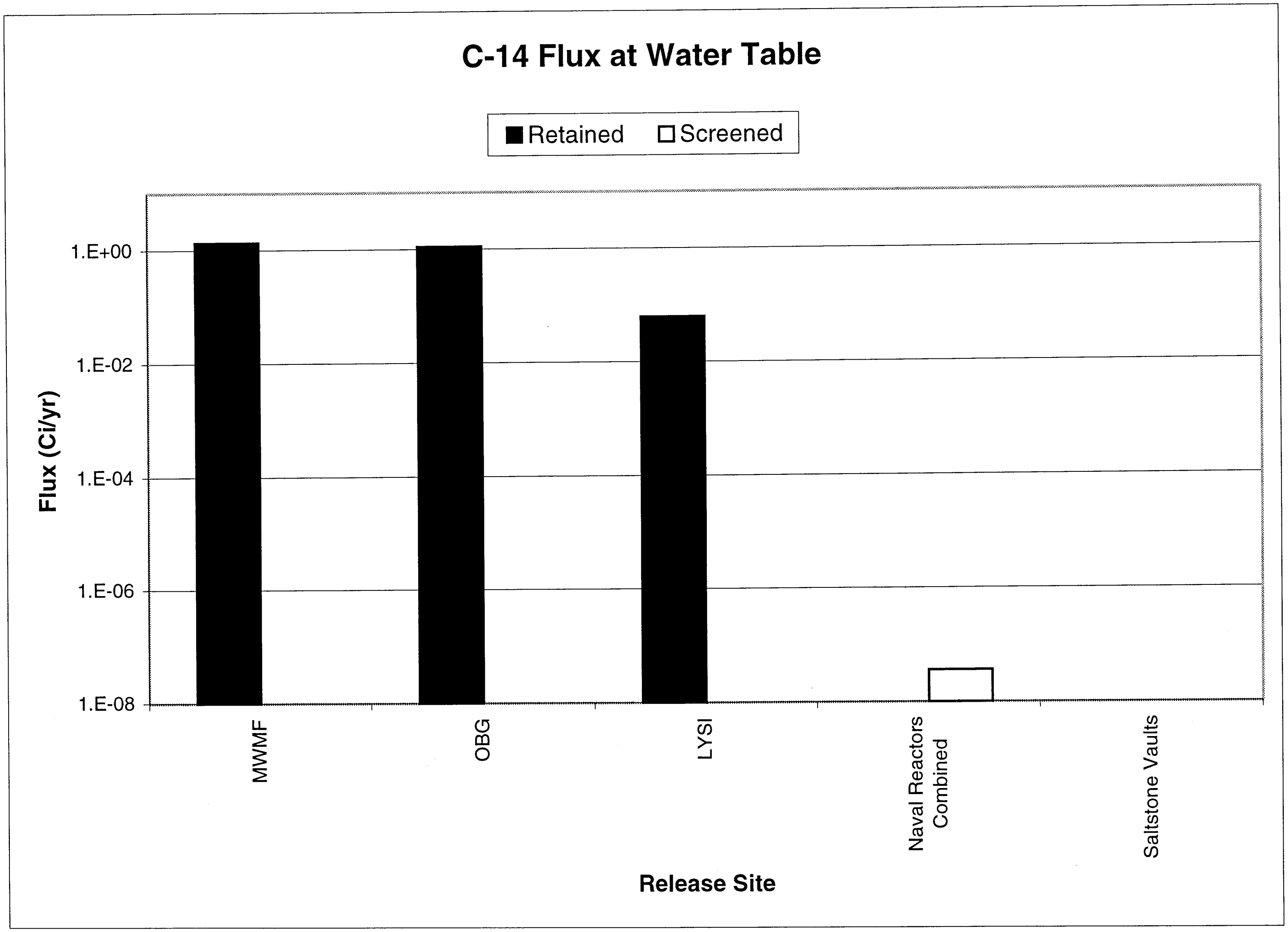

Figure 6.6-7 ^{14}C **Inventories for All Facilities Screened by Flux at Water Table**

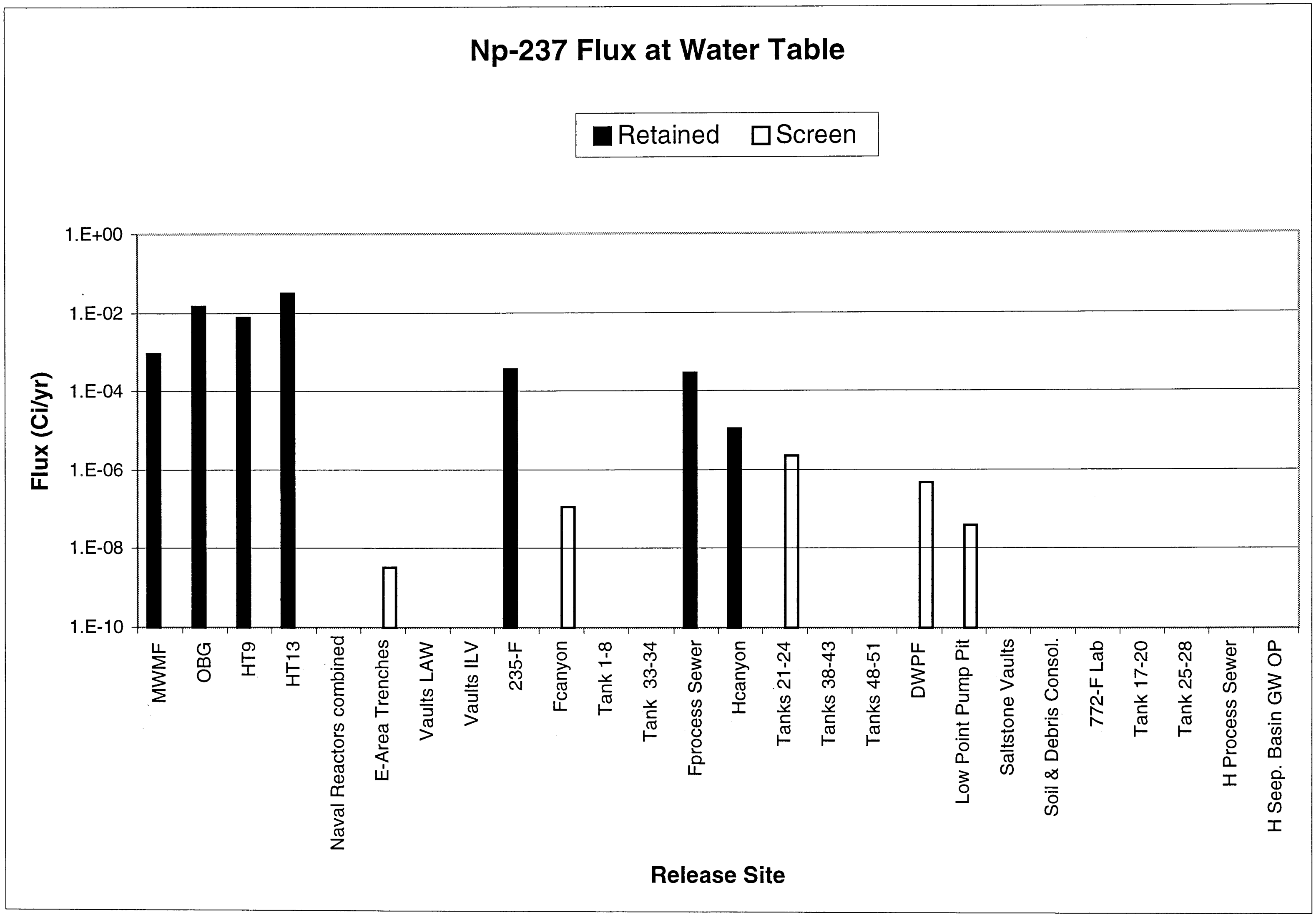

Figure 6.6-8 ^{237}**Np Inventories for All Facilities Screened by Flux at Water Table**

Table 6.6-2 ^{3}H Peak Flux to Water Table[2]

Source	Inventory (Ci)	Peak Flux at Water Table Time (yr)	Peak Flux (Ci/yr)
Sources Eliminated During Inventory Screening Phase			
Saltstone Lysimeters	7.39×10^{-1}		
E-Area Trenches	8.75		
772-F Lab	1.06×10^{1}		
772-1F Lab	1.00×10^{-1}		
F Process Sewer	1.11×10^{1}		
H Canyon	1.02		
ETF Receipt Tank	7.00×10^{-2}		
H Process Sewer	2.87×10^{1}		
DWPF	6.34×10^{-2}		
Low Point Pump Pit	3.17×10^{-3}		
Tank 16 Spill	5.00×10^{-2}		
Tank 37 Spill	8.41×10^{-2}		
Soil and Debris Consol.	3.71×10^{-2}		
Sources Eliminated During Contaminant Flux Screening Phase			
Naval Reactors CB/TS and Naval Reactors Head	4.39×10^{2}	Other Screen[3]	$<1.\times10^{-18}$
LAW Vaults	1.66×10^{6}	85[4]	9.79×10^{-5}
ILV Vaults	8.80×10^{5}	114[4]	8.54×10^{-8}
F Canyon	6.79×10^{1}	23	9.2
Saltstone Vaults	1.90×10^{4}	89[6]	3.8×10^{-8}
Sources Remaining After Both Screening Phases			
MWMF	2.29×10^{6}	35	6.25×10^{4}
OBG	2.10×10^{6}	20	3.6×10^{4}
TRIT	3.00×10^{4}	41	6.3×10^{3}

Notes:
[2]Peak time from Figure 4.4-2, Inventory from Table 4.4-2, Peak flux from Table 4.4-5
[3]From Table L.2-1 in WSRC, 1996. 13.5 Ci per barrel after 750 years decays to less than 1×10^{-18} Ci.
[4]From Table 4.1-3, WSRC, 1994.
[5]From PATHRAE-RAD computer run
[6]From Table 4.1-3, WSRC 1992.

Table 6.6-3 ^{14}C Peak Flux to Water Table[7]

Site	Inventory (Ci)	Peak Flux at Water Table: Time (yr)	Peak Flux at Water Table: Peak Flux (Ci/yr)
Sources Eliminated During Inventory Screening Phase			
Saltstone Lysimeters	2.53×10^{-4}		
LAW Vaults	1.70×10^{-1}		
ILV Vaults	2.24×10^{-3}		
F Canyon	2.85×10^{-1}		
Tank 1-8	1.15×10^{-3}		
Tank 17-20	7.80×10^{-3}		
Tank 25-28 & 44-47	3.34×10^{-2}		
H Canyon	4.28×10^{-3}		
Tank 9-12	7.97×10^{-4}		
Tank 13-16	2.88×10^{-4}		
Tanks 21-24 & 29-32 & 35-37	8.79×10^{-4}		
Tanks 38-43	5.85×10^{-4}		
Tanks 48-51	2.08×10^{-4}		
Soil and Debris Consol.	9.06×10^{-2}		
Sources Eliminated During Contaminant Flux Screening Phase			
Naval Reactors CB/TS and Naval Reactors Head	6.79×10^{2}	10000[8]	3.60×10^{-8}
Saltstone Vaults	6.50	Other Screen[9]	$<1.\times10^{-18}$
Sources Remaining After Both Screening Phases			
MWMF	3.72×10^{3}	140	1.35
OBG	3.09×10^{3}	180	1.12
LYSI	1.75	180	6.18×10^{-2}

Notes:
[7]Peak time from Figure 4.4-2, Inventory from Table 4.4-2, Peak flux from Table 4.4-5
[8]Peak time from Table L.3-1, WSRC 1996
[9]From Table 4.1-3, WSRC, 1992.

Table 6.6-4 ^{237}Np Peak Flux to Water Table[10]

Site	^{237}Np Inventory (Ci)	^{241}Am Forcing Consideration Inventory (Ci)	^{241}Pu Forcing Consideration Inventory (Ci)	Peak Flux at Water Table: Time (yr)	Peak Flux at Water Table: Peak Flux (Ci/yr)
Sources Eliminated During Inventory Screening Phase					
Saltstone Lysimeters	2.27×10^{-6}				5.69×10^{-11}
Sources Eliminated During Contaminant Flux Screening Phase					
Naval Reactors CB/TS and Naval Reactors Head	1.29×10^{-5}	1.13×10	1.09×10^{2}	Other Screen[12]	NA
E-Area Trenches	8.85×10^{-7}	2.57×10^{-1}		215[13]	3.15×10^{-9}
LAW Vaults	8.69×10^{-3}				$<1.\times10^{-18}$
ILV Vaults	1.75×10^{-3}				$<1.\times10^{-18}$
F Canyon	3.53×10^{-3}				1.09×10^{-7}
Tank 1-8	5.25×10^{-2}				$<1.\times10^{-18}$
Tank 33-34	2.11×10^{-2}				$<1.\times10^{-18}$
F Process Sewer	2.15×10^{-2}				2.91×10^{-4}
Tanks 21-24	2.45×10^{-2}				2.28×10^{-6}
Tanks 38-43	9.70×10^{-3}				$<1.\times10^{-18}$
Tanks 48-51	1.50×10^{-4}				$<1.\times10^{-18}$
DWPF	1.52×10^{-2}				4.68×10^{-7}
Low Point Pump Pit	7.60×10^{-4}				3.80×10^{-8}
Saltstone Vaults	5.80×10^{-2}				NR[14]
Soil and Debris Consol.	4.97×10^{-8}	4.18×10^{-2}			
772-F Lab			1.91		1×10^{-18}
Tank 17-20		7.17×10^{1}	4.26×10^{2}		1×10^{-18}
Tank 25-28		4.19×10^{2}	6.38×10^{2}		1×10^{-18}
H Process Sewer		2.07×10^{-1}			2×10^{-18}
H Seep. Basin GW Op Unit		3.93×10^{-1}			2×10^{-18}
Sources Remaining After Both Screening Phases					
MWMF	9.59×10^{-2}			310	9.31×10^{-4}
OBG	1.57			380	1.52×10^{-2}
HT9	3.44×10^{-2}			610	7.89×10^{-3}
HT13	2.04×10^{-2}			610	3.2×10^{-2} [11]
235-F	1.20×10^{1}			Not Plotted	3.69×10^{-4}
H Canyon	3.56×10^{-1}				1.10×10^{-5}

Notes:

[10]Peak time from Figure 4.4-2, Inventory from Table 4.4-2, Peak flux from Table 4.4-5

[11]Value is from Figure 4.4-2 which is higher than 2.62×10^{-2} shown in Table 4.4-5

[12]WSRC 1996, Table L.2-3 inventory about 1 order of magnitude below screen threshold.

[13]Table 4.3-5, WSRC 1998.

[14]Not Reported. (WSRC, 1992) only reported ^{241}Am flux to water table of < 10×10^{-6} pCi/yr.

Table 6.6-5 Significant Sources

Source	Contaminant	Data Qualifier	Qualifier Title
MWMF	^{3}H	2	Shipping and Disposal Record, Facility Inventories
OBG	^{3}H	2	Shipping and Disposal Record, Facility Inventories
TRIT	^{3}H	7	Interviews with Plant Personnel
MWMF	^{14}C	2	Shipping and Disposal Record, Facility Inventories
OBG	^{14}C	2	Shipping and Disposal Record, Facility Inventories
LYSI	^{14}C	1	Peer-reviewed Technical Reports
MWMF	^{237}Np	2	Shipping and Disposal Record, Facility Inventories
OBG	^{237}Np	2	Shipping and Disposal Record, Facility Inventories
HT9	^{237}Np	3	Process Modeling
HT13	^{237}Np	3	Process Modeling
235-F	^{237}Np	5	Process Knowledge
H Canyon	^{237}Np	5	Process Knowledge

case inventory was used as the median value for each PDF. As the data qualifier increased, the uncertainty increased and the PDF's range of inventories increased.

Parameters and distribution types describing each PDF are provided in Table 6.6-6. Case ID's with an N suffix are lognormal while case ID's with a T suffix are logtriangular. Sampling details are described in Appendix A.

After inventory sampling, this study deviated from the typical approach. A novel approach was implemented to limit the number of computer runs needed to model contaminant transport. The approach requires the recognition of two key relationships. First, total doses at a stream can be calculated by summing the releases from each source. Second, fluxes and doses at a stream from a single source are linearly related to the source inventory, so relative inventory changes produce equal relative dose changes (e.g., if the inventory doubles, then the dose doubles).

These relationships allowed total doses to be calculated in a spreadsheet-type operation after independently modeling the base case for each significant source. The uncertainty study required scaling each source's base results by the relative sample inventory (relative to the base inventory), then summing the scaled results from all sources. The steps are shown in Figure 6.6-9 and are listed in Table 6.6-7.

Validity of Approach

The basic premises for this approach are as follows:

1) The total annual dose due to a stream is directly proportional to the sum of the releases to the stream from each source for each radionuclide in that year
2) The release into a stream from an individual source is directly proportional to the inventory of that source.

The first premise allows each source to be modeled separately. It postulates that the effects from one source are independent of all other sources. This premise requires that the adsorption-desorption curve be linear and that diffusion results be additive for multiple sources. The transport computer program models the case for a linear adsorption-desorption curve, so the first requirement is satisfied. Diffusion results are not additive where plumes interact from two sources. Vadose zone transport was modeled independently for each source, so no plume interaction was permitted. In the aquifer, advection dominates such that diffusion becomes at least a second or third order effect.

For a single source, the second premise allows that source to be modeled with a base case contaminant inventory to generate a release to the stream. After calculating the release to the stream for each source separately, the releases are summed to give a total annual release. The release is divided by the annual stream flow to give a concentration, which in turn is used to calculate the dose.

To check the new approach, an initial sample equal to the base inventory was selected at each source and combined to form a check realization set. The total check doses match the earlier CA results that were obtained by simultaneously modeling each source.

Table 6.6-6 Input Probability Distributions and Parameters

Data Qualification Value	Case ID	Area/Location	Isotope	Median Activity (m)	Activity Range Factor (f)	50% Probability Range for Lognormal Distribution [m/f, mf]	100% Probability Range for Logtriangular Distribution [m/f, mf]
1	1N	Lysimeters	^{14}C	1.75	2	[0.875, 3.5]	---
2	2N	Old Burial Ground	^{3}H	2.1×10^{6}	5	[4.2×10^{5}, 1.05×10^{7}]	---
2	3N	Old Burial Ground	^{14}C	3100	5	[620, 1.55×10^{4}]	---
2	4N	Old Burial Ground	^{237}Np	1.6	5	[0.32, 8]	---
2	5N	MWMF	^{3}H	2,300,000	5	[4.6×10^{5}, 1.15×10^{7}]	---
2	6N	MWMF	^{14}C	3700	5	[740, 1.85×10^{4}]	---
2	7N	MWMF	^{237}Np	0.096	5	[0.0192, 0.48]	---
3	1T	HLW Tanks 9-12	^{237}Np	0.034	20	---	[0.0017, 0.68]
3	2T	HLW Tanks 13-16	^{237}Np	0.02	20	---	[0.001, 0.4]
5	3T	H Canyon	^{237}Np	0.36	50	---	[0.0072, 18.0]
5	4T	235-F	^{237}Np	12.0	50	---	[0.24, 600]
7	5T	Tritium Facilities	^{3}H	30,000	100	---	[300, 3.0×10^{6}]

Table 6.6-7 Simplified Uncertainty Approach

Step Number	Inventory	Operation	Results
1	Base	Sample	1,000 Sample Inventories
2	Each Source's Base Inventory Analyzed Independently	Model	Releases to Stream from Each Source for Base Inventory
3	Sample	Scale Releases by Sample Inventory / Base Inventory	Partial Stream Releases from Each Source
4	Sample	Sum Releases for all Sources × Dose Conversion Factor	Total Dose
Check	Base	Sum Base Releases for all Facilities × Dose Conversion Factor	Total Doses for Base Inventory to check against CA results that considered all inventories simultaneously

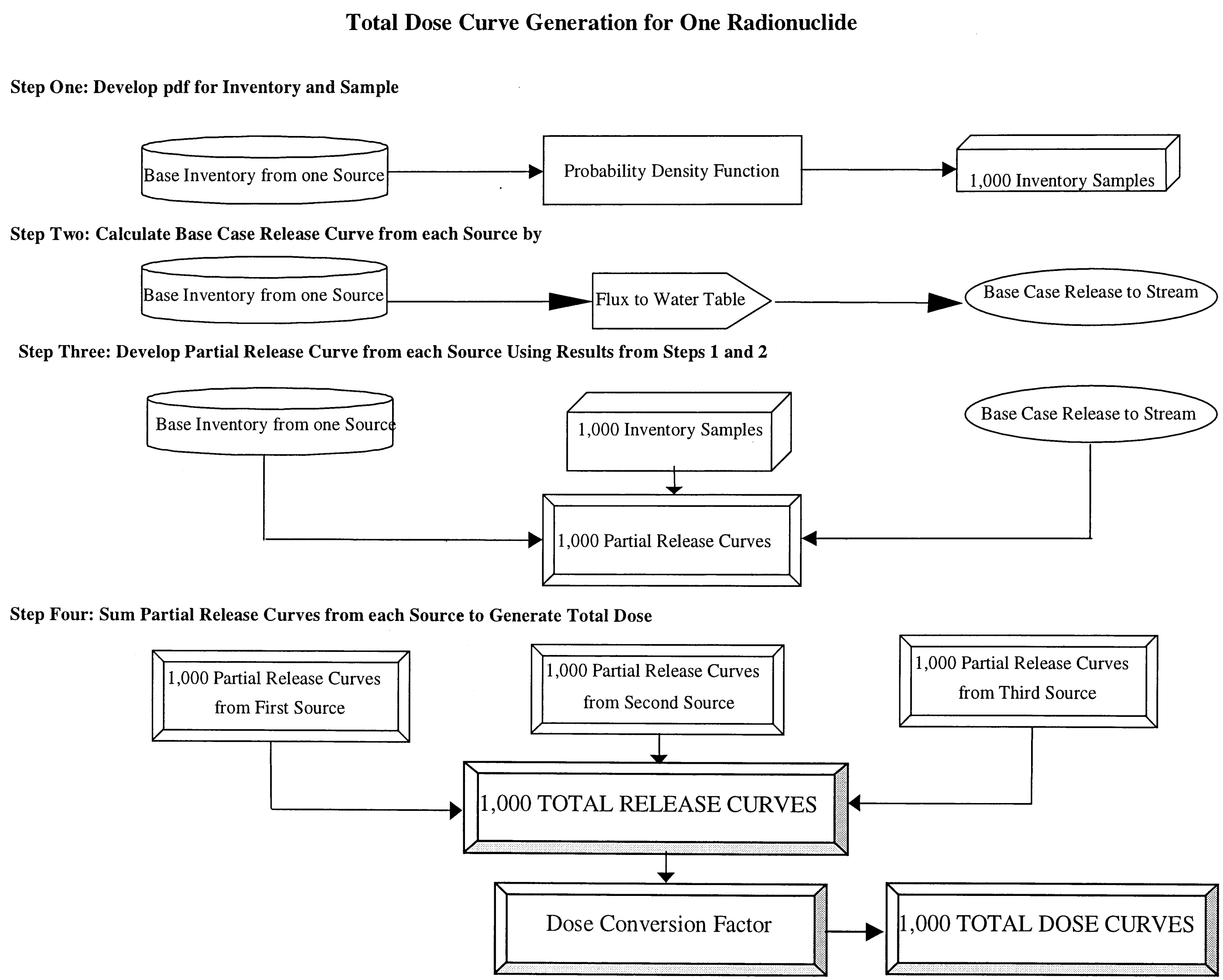

Figure 6.6-9 Simplified Uncertainty Approach

Benefits of Approach

For a single contaminant, the new uncertainty approach requires a separate computer run for each source. For a single contaminant, a traditional uncertainty analysis approach accommodates all sources in a single computer run, but the traditional approach requires separate computer runs for each realization. Because the double screening reduced the number of significant sites, the computer runs for the new approach were substantially reduced. The computer run savings are expressed in Table 6.6-8 for one thousand realizations.

6.6.4 Inventory Variation Sampling Results

For each important radionuclide significant source (see Table 6.6-5), the following sampling, scaling and summing process was implemented:

1. A PDF was developed for the inventory at each source
2. One thousand independent random samples were selected from the inventory PDF at each source

The PDF's for each radionuclide from each significant release site are shown in Figures 6.6-10 through 6.6-12.

6.6.5 Dose Results from Inventory Variations

For each important radionuclide, the base case dose curve was generated with transport modeling. The dose curve consists of a plot of doses at a stream versus time. After sampling each significant source inventory (see Table 6.6-5), the samples and the base case dose curve were combined to produce total dose curves by implementing the following method:

1. One thousand partial dose curves for each significant source were generated
2. Each partial dose curve was calculated by multiplying the base case dose curve by a random sample inventory and dividing by the base case inventory
3. Partial dose curves for each significant source were summed to generate one thousand total dose curves.

Thus, one thousand total dose curves were developed for the following scenarios:

^{3}H at FMB
^{14}C at FMB
^{237}Np at FMB
^{14}C at UTR.

The complete set of total dose curves for ^{14}C at UTR is shown in Figure 6.6-13. The other plots are not shown because of the vast amount of data required for each plot.

The total dose curves for ^{14}C at UTR slope relatively steeply from time zero to 500 years. After that time, the slope is essentially flat for the remaining 500 years. Since Figure 6.6-13 displays a linear dose axis, only the curves with very high values are distinct from the central mass. The visibly distinct curves displaying the greatest values originate from a combination of high sample inventories from the OBG and the MWMF (see Figure 6.6-11).

Table 6.6-8 Computer Run Savings

Contaminant	Number of Sources	Number of Traditional Computer Runs	Number of Computer Runs for New Approach	Percentage Savings
^{3}H	3	1000	3	99.7
^{14}C	3	1000	3	99.7
^{237}Np	6	1000	6	99.4
TOTAL	12	3000	12	99.6

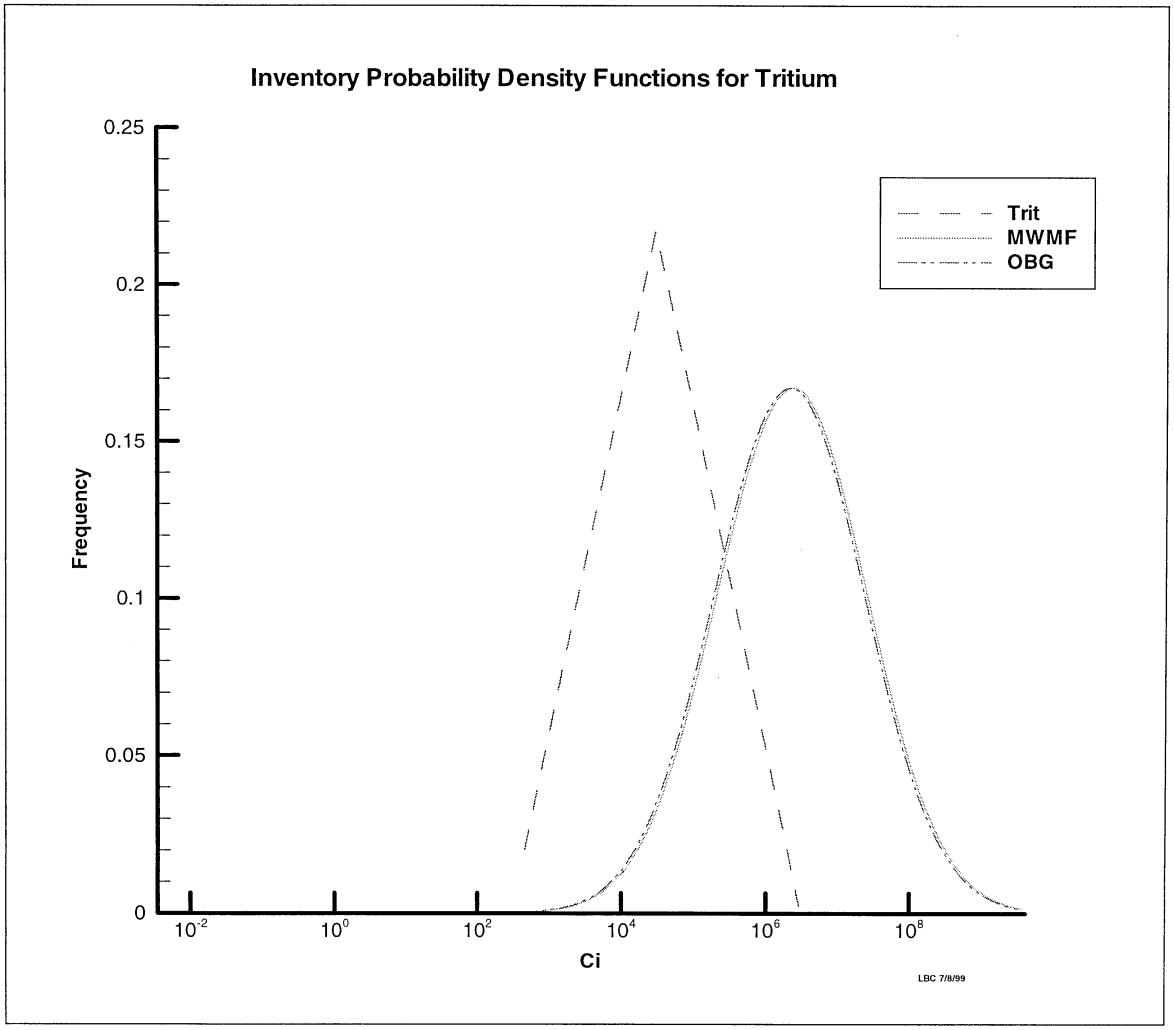

Figure 6.6-10 Probability Density Function for ^3H

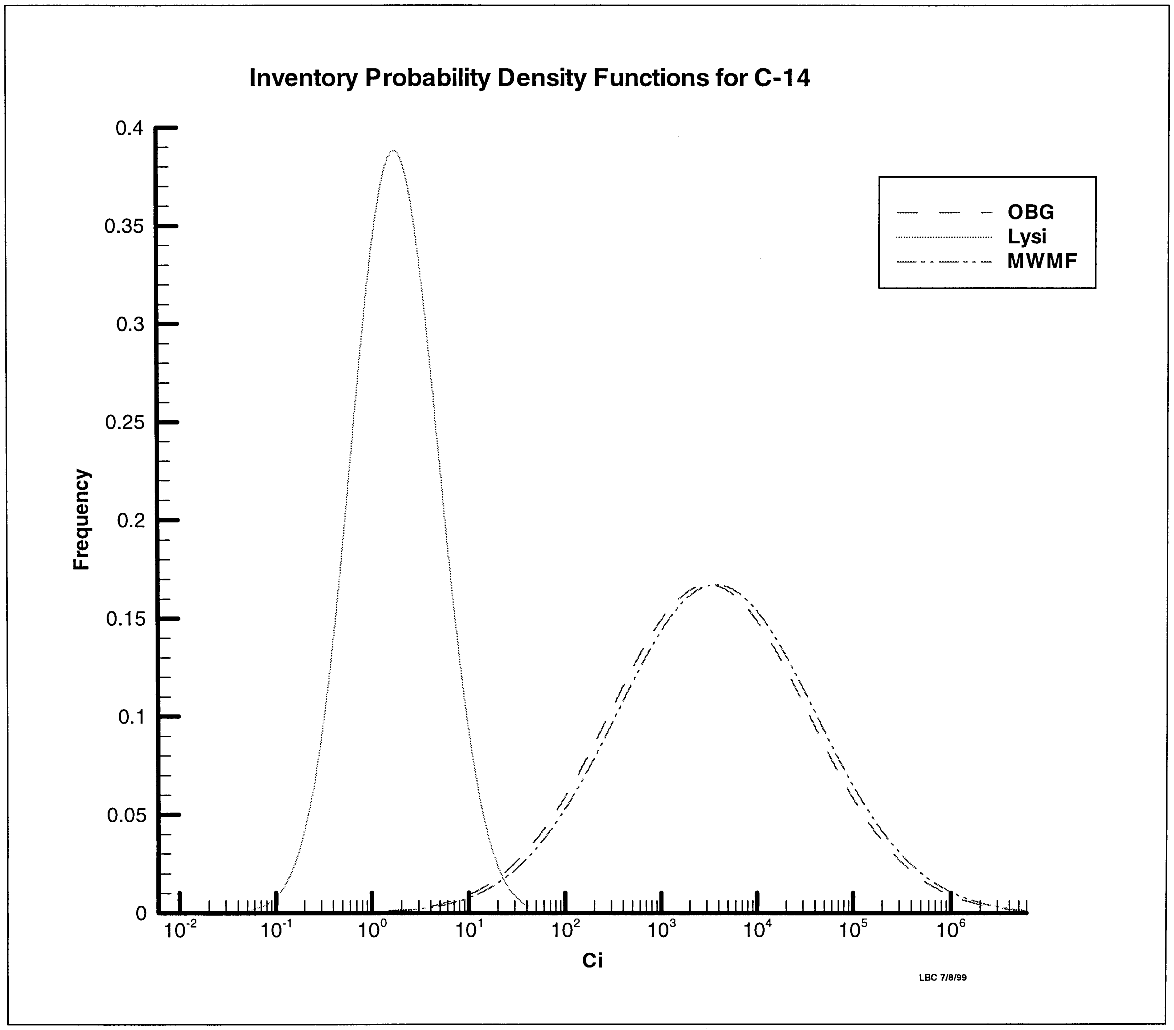

Figure 6.6-11 Probability Density Function for ^{14}C

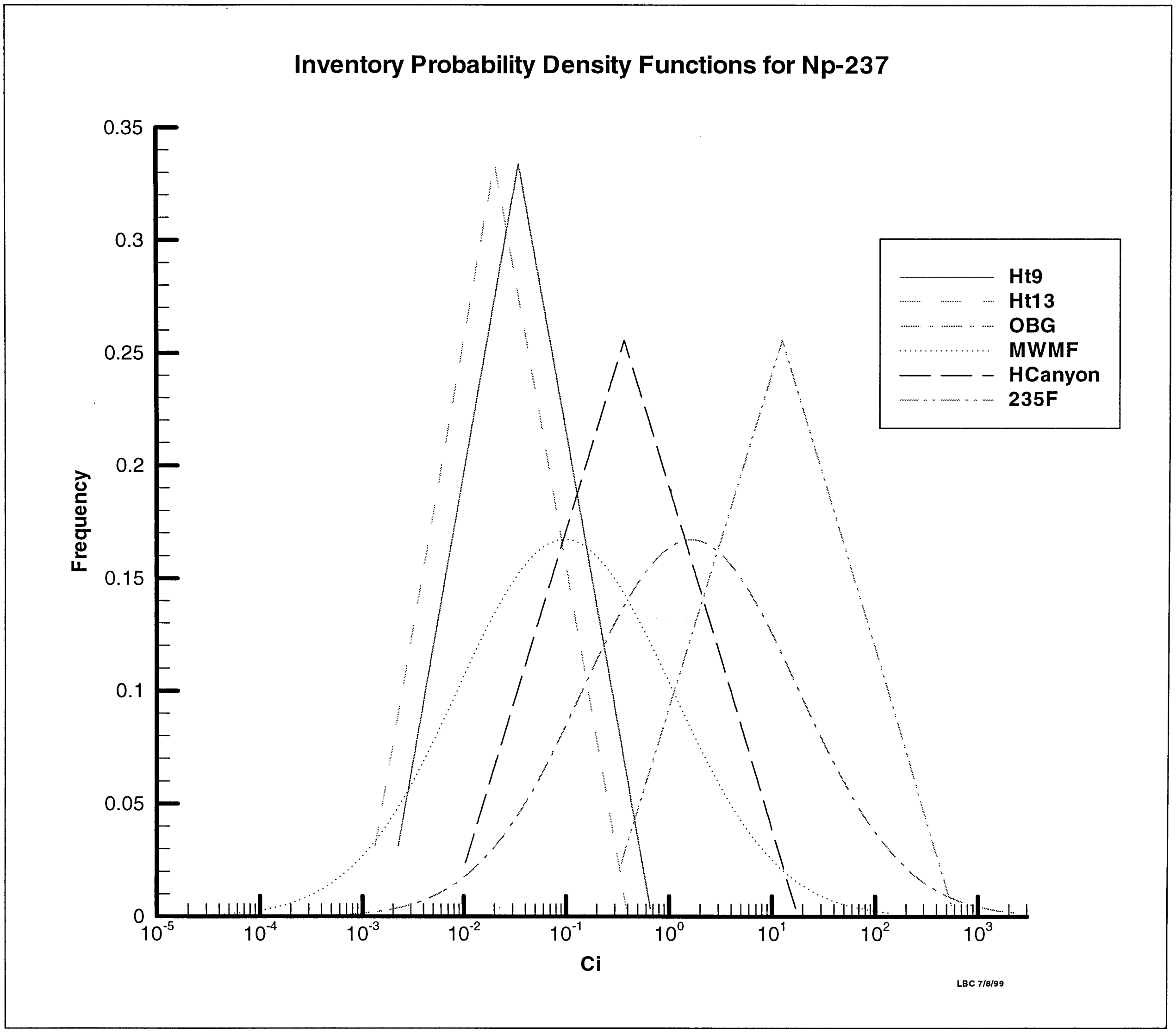

Figure 6.6-12 Probability Density Function for ^{237}Np

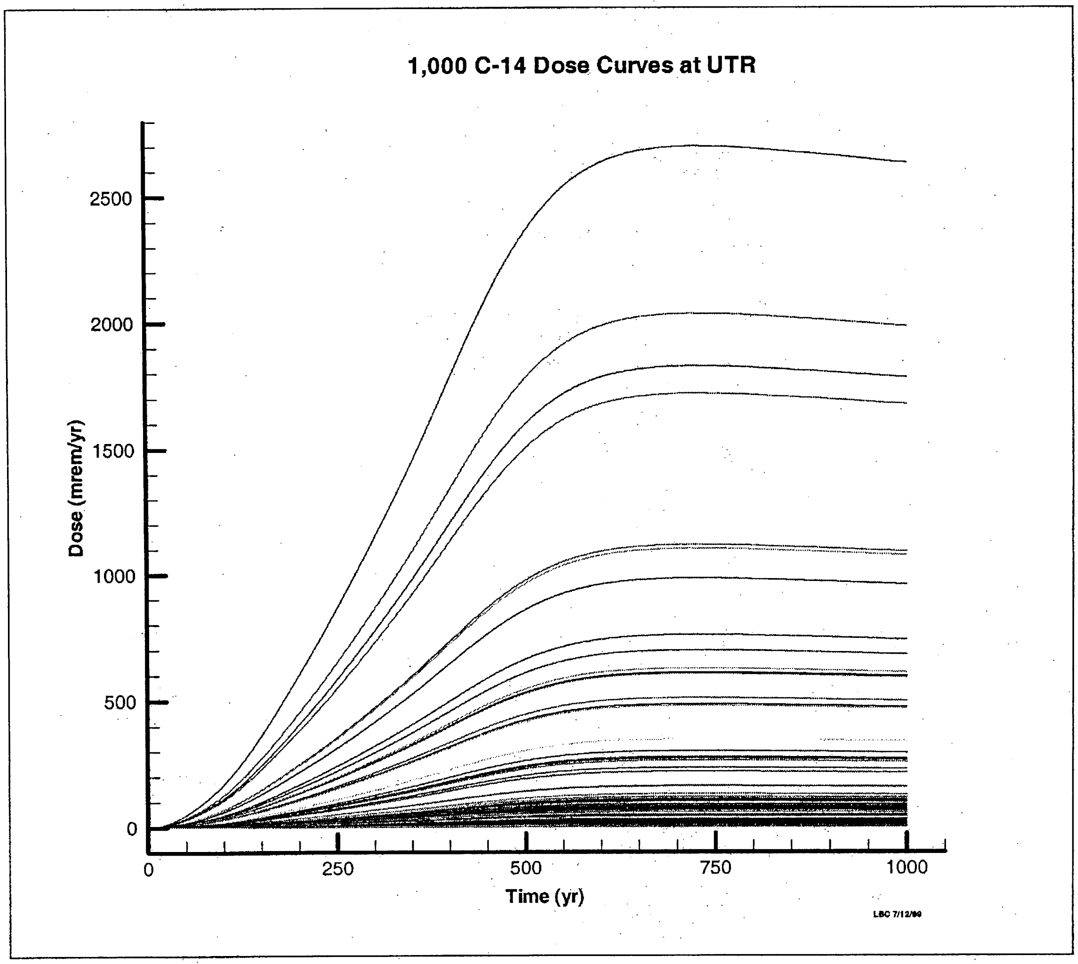

Figure 6.6-13 One Thousand Dose Response Curves for ^{14}C at UTR

Peak Dose Plots

For each scenario, the peak doses from the total dose curves were collected and sorted to produce a cumulative frequency plot. These plots are shown in Figures 6.6-14 through 6.6-17. Additionally, sorted doses were collected in bins. These histograms are shown in Figures 6.6-18 through 6.6-21.

The peak total dose cumulative frequency plot for ^{3}H at FMB (see Figure 6.6-14) and its associated histogram (see Figure 6.6-18) reveal an almost lognormal distribution of results. The most important sources based on flux to the water table are the MWMF and the OBG (see Figure 4.4-2). The inventory PDFs (see Figure 6.6-10) for the MWMF and the OBG are both lognormal, thus the results should be essentially lognormal.

The peak total dose cumulative frequency plot for ^{14}C at FMB (see Figure 6.6-15) and its associated histogram (see Figure 6.6-19) reveal an almost lognormal distribution of results, although there appears to be a slight skew to the right. The most important sources based on flux to the water table are MWMF and OBG (see Figure 4.4-1). The inventory PDFs (see Figure 6.6-11) for MWMF and OBG are both lognormal, thus the results should be essentially lognormal.

The peak total dose cumulative frequency plot for ^{237}Np at FMB (see Figure 6.6-16) and its associated histogram (see Figure 6.6-20) reveal an asymmetrical distribution of results. The peak bin occurs around 1 Ci. To the left, the distribution steps down rapidly with very little tail. To the right, the distribution steps down more gradually with much more of a tail. The most important sources based on flux to the water table are HT13, the OBG, and HT9 (see Figure 4.4-4). The inventory PDFs (see Figure 6.6-12) for HT13, the OBG, and HT9 are logtriangular, lognormal, and logtriangular, respectively, thus the results generally would be asymmetrical.

For ^{237}Np, the total dose for all base case inventories occurs at about the 33rd percentile of sampled peak doses. The other important radionuclides have an all base case total dose very near the 50th percentile of sampled peak doses. This apparent anomaly is likely caused by the interaction of three major sources with similar peaks occurring at slightly different times and by the mixture of lognormal and logtriangular inventory distributions. If one source's partial peak dose at FMB is greater than the base case peak total dose, then it does not matter what the other partial peak doses are. Because the very high peak doses are more important than the very low peak doses, the peak total dose curve tends to be skewed toward the higher end.

The time of the peak total dose at FMB for the base case was 476 years. The times of the peak total doses from the uncertainty analysis ranged from 428 years to 496 years, indicating that multiple sources were affecting the results. Table 6.6-9 shows that HT13 has the most influence, but OBG and HT9 are almost as important. Table 6.6-9 also shows that the times of the partial dose peaks are close. The time for the peak total dose decreased when OBG's influence increased and the time for the peak total dose increased when HT9's influence increased.

The peak total dose cumulative frequency plot for ^{14}C at UTR (see Figure 6.6-17) and its associated histogram (see Figure 6.6-21) reveal an almost lognormal distribution of results, although there appears to be a slight skew to the right. The most important sources based on flux to the water table are the MWMF and the OBG (see Figure 4.4-1). The inventory PDFs (see Figure 6.6-11) for the MWMF and the OBG are both lognormal, thus the results should be essentially lognormal.

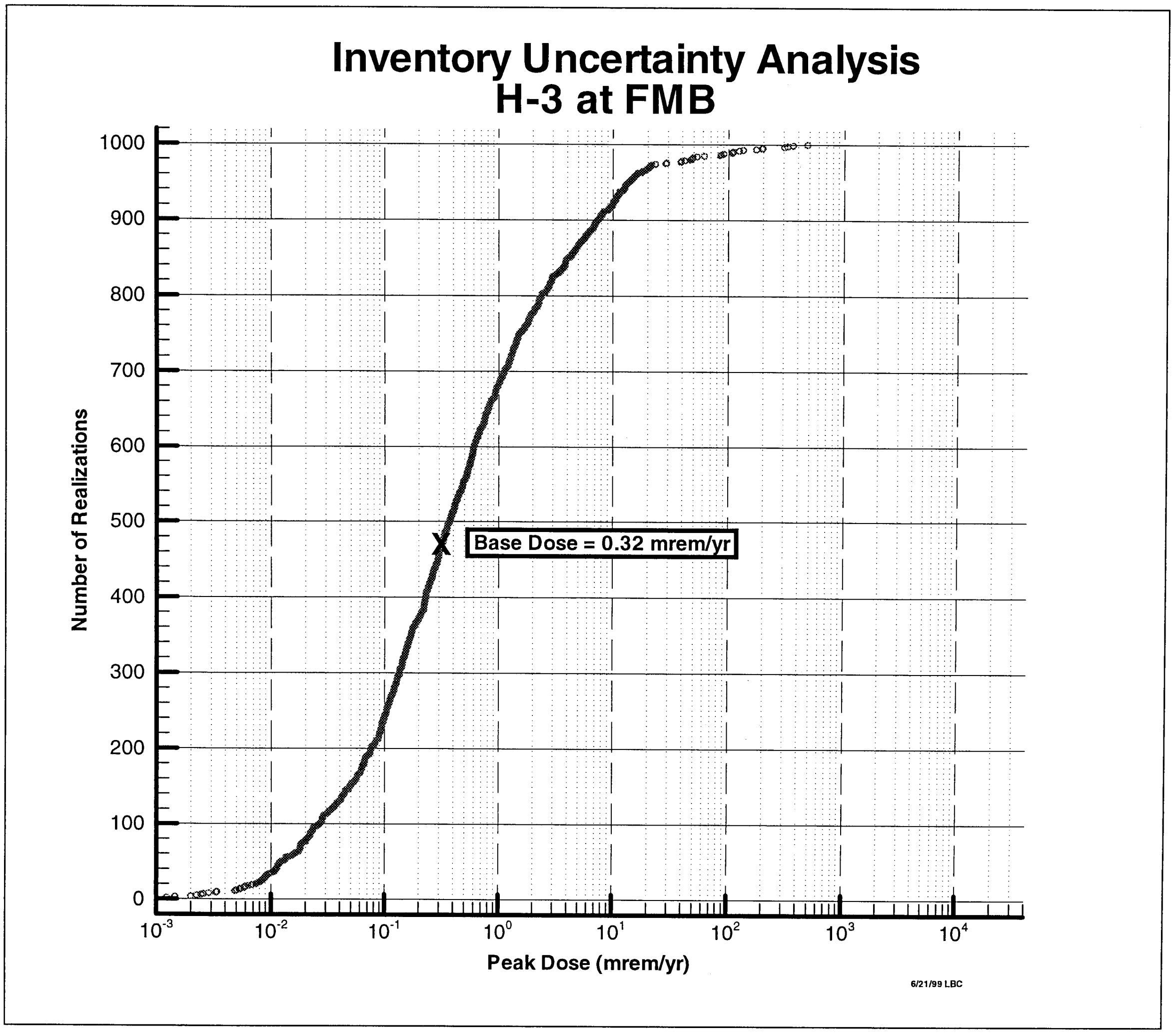

Figure 6.6-14 Peak Dose Cumulative Frequency Plot of ^{3}H at FMB

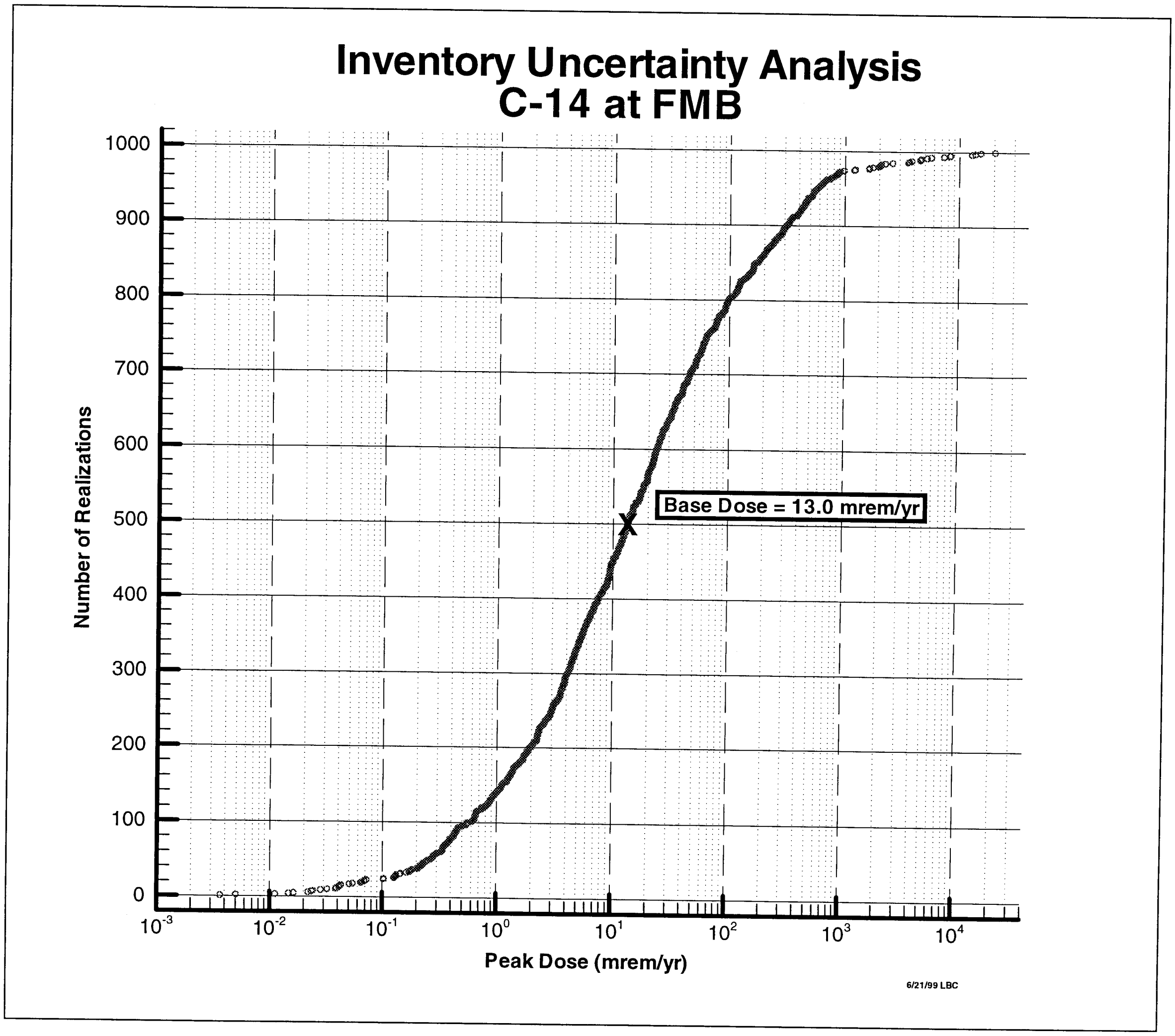

Figure 6.6-15 Peak Dose Cumulative Frequency Plot of ^{14}C at FMB

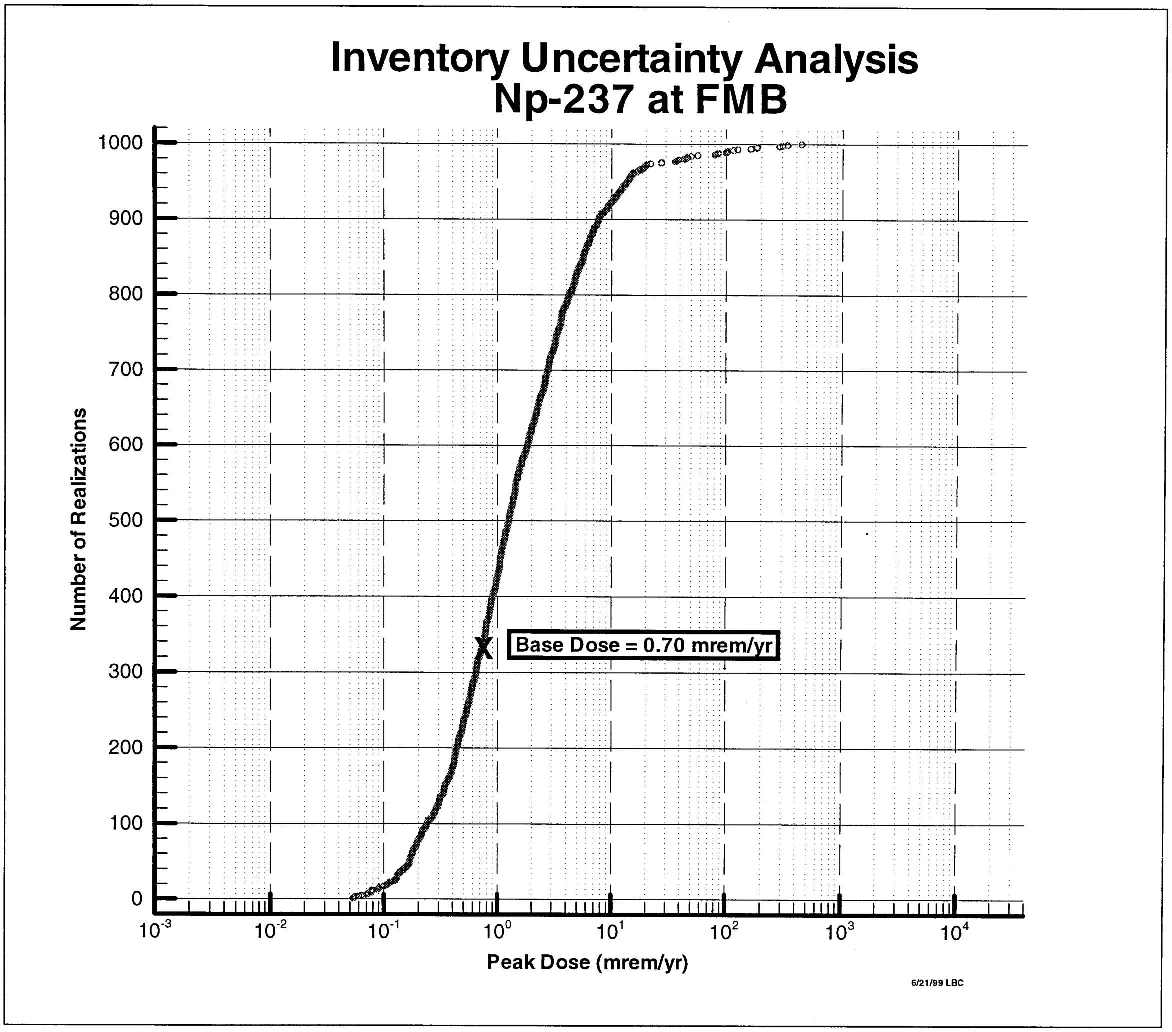

Figure 6.6-16 Peak Dose Cumulative Frequency Plot of ^{237}Np at FMB

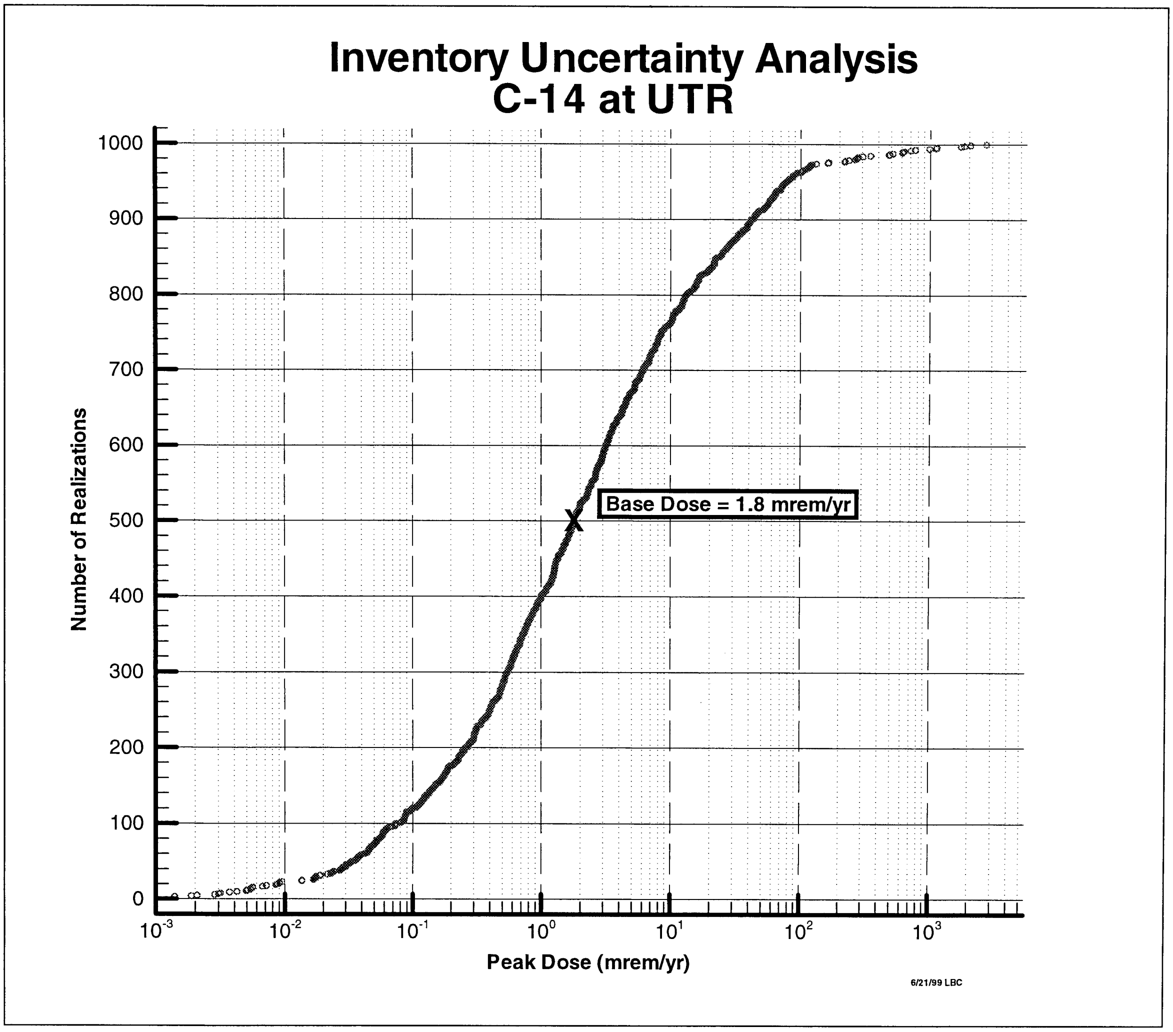

Figure 6.6-17 Peak Dose Cumulative Frequency Plot of ^{14}C at UTR

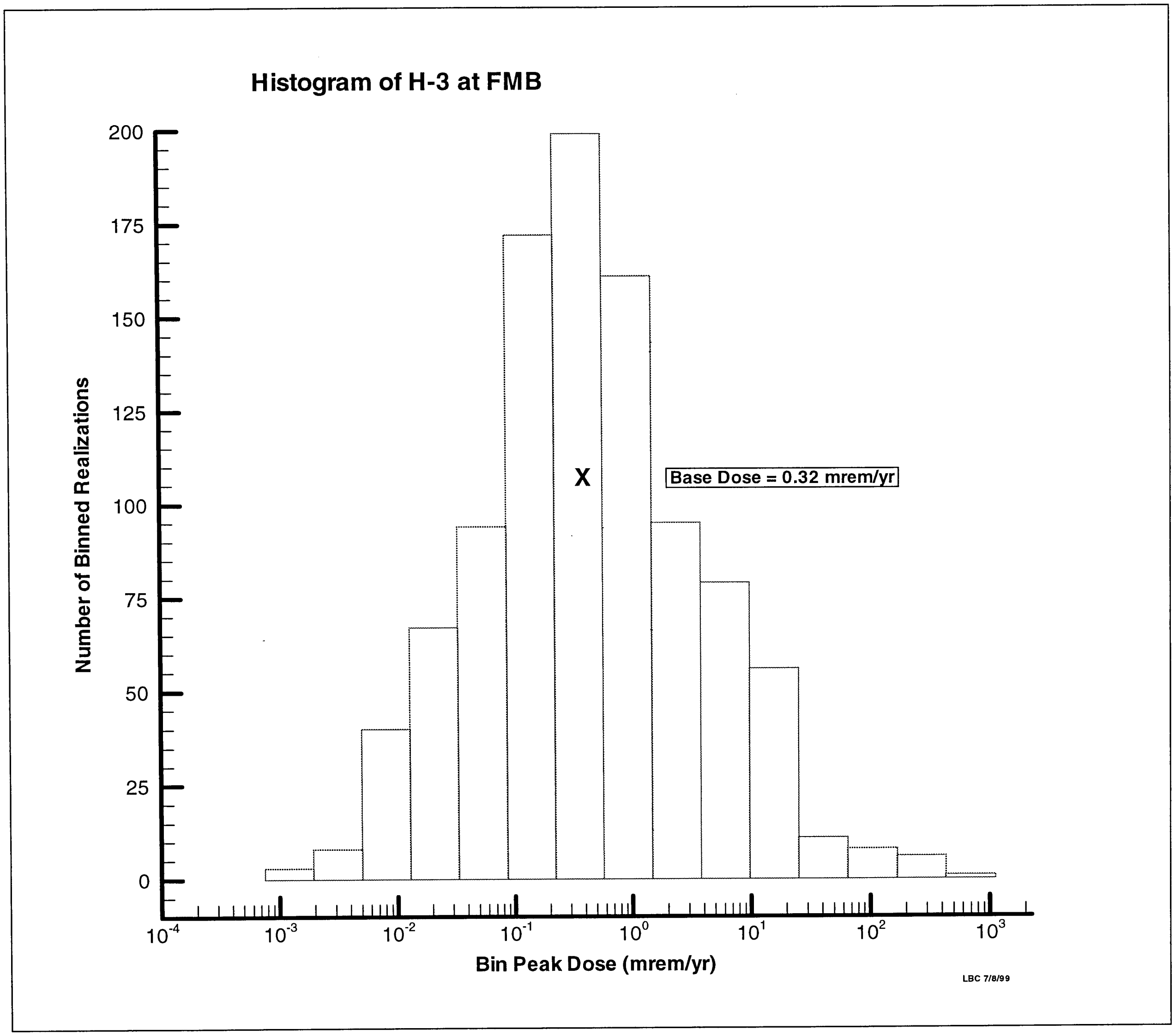

Figure 6.6-18 Peak Dose Histogram of ^3H at FMB

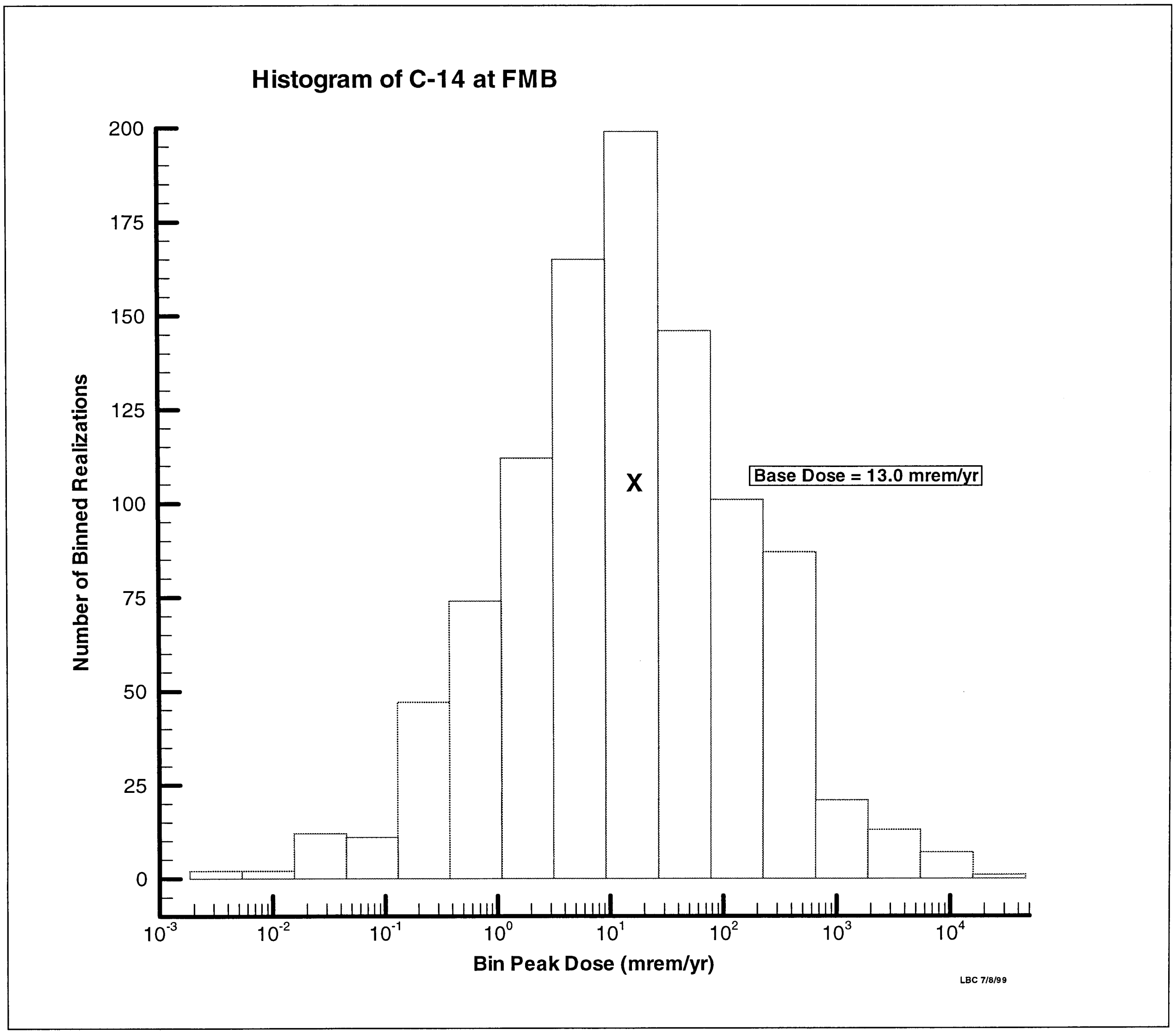

Figure 6.6-19 Peak Dose Histogram of ^{14}C at FMB

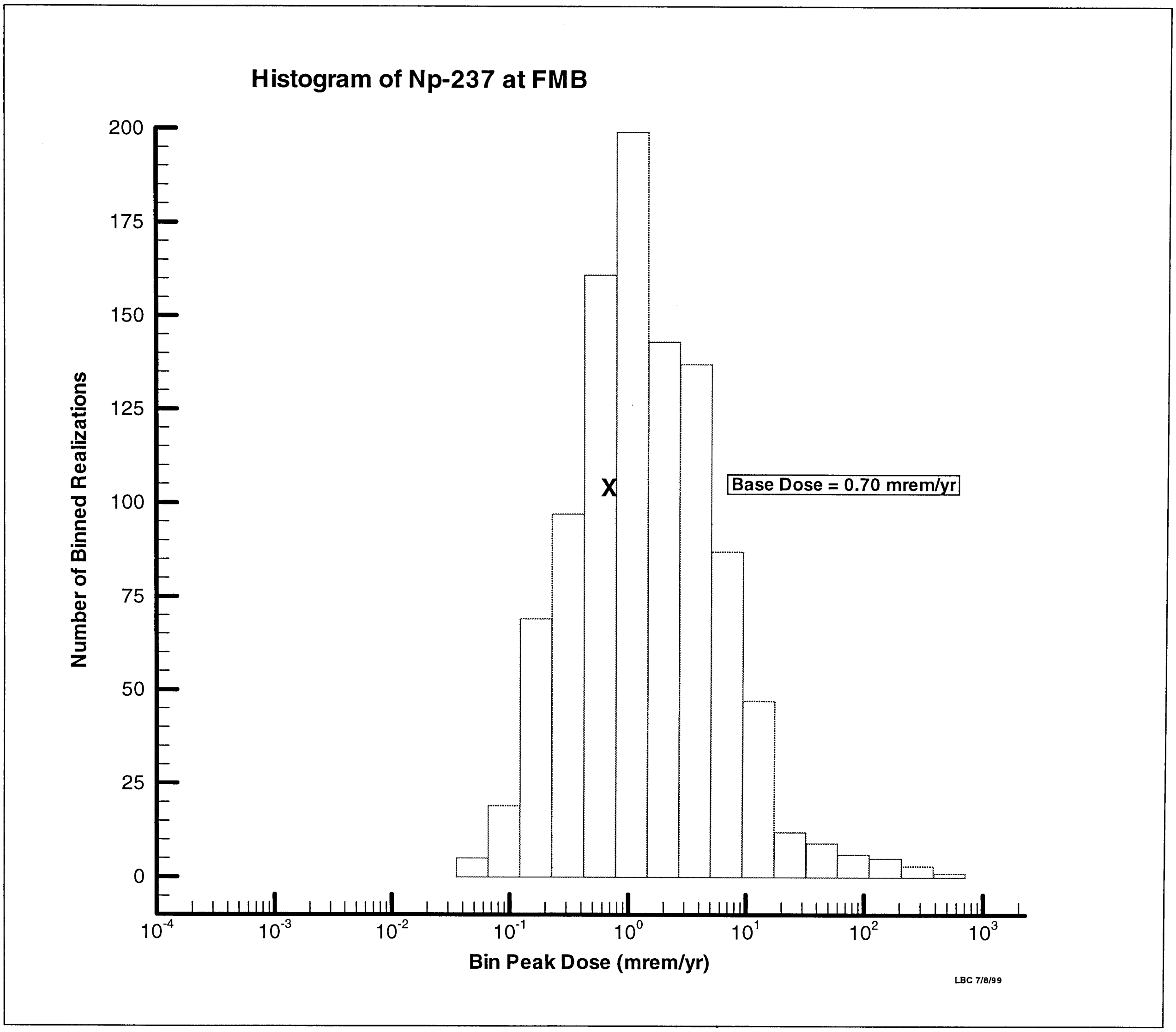

Figure 6.6-20 Peak Dose Histogram of ^{237}Np at FMB

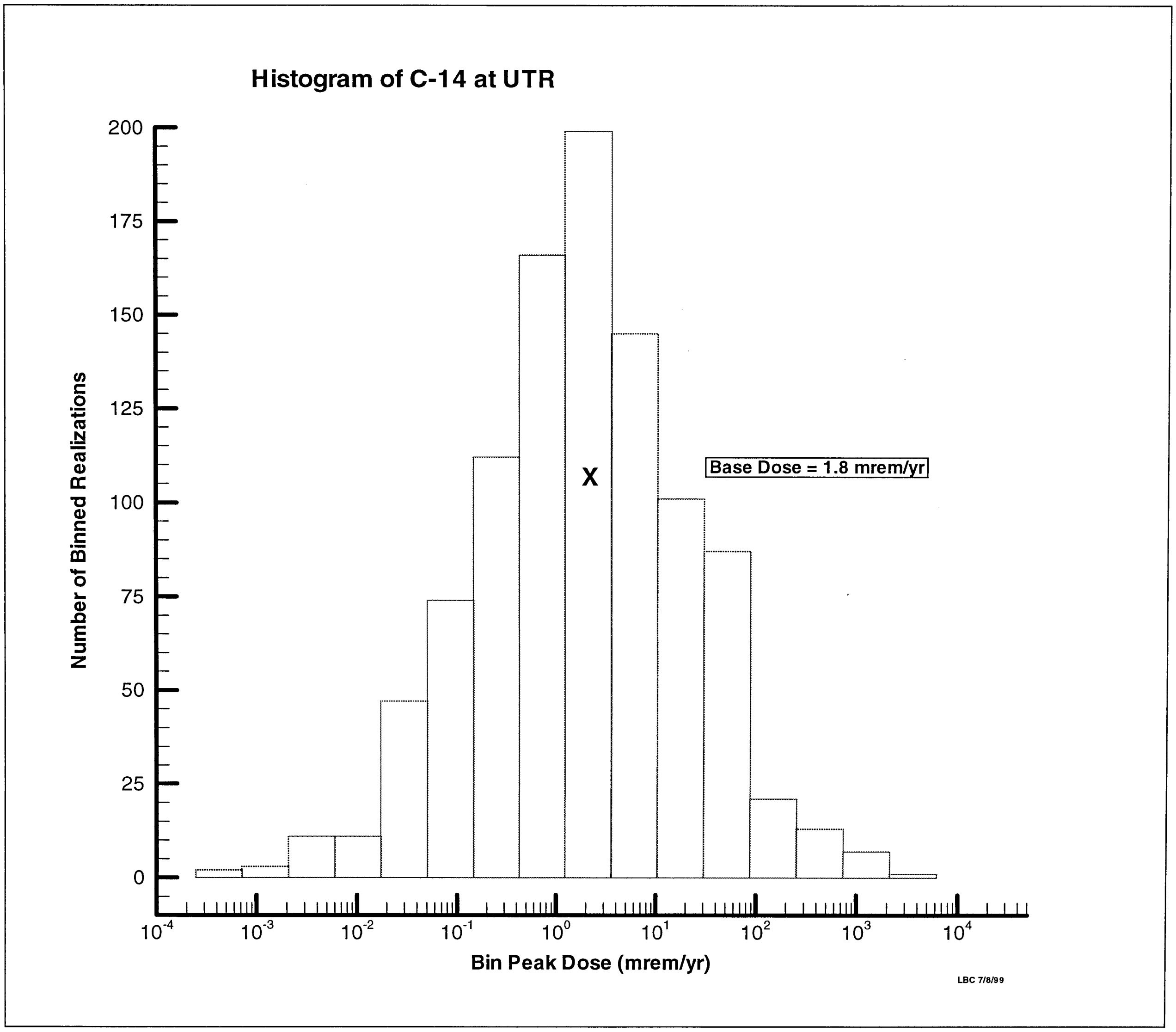

Figure 6.6-21 Peak Dose Histogram of ^{14}C at UTR

Table 6.6-9 ^{237}Np Major Peaks

Source	Peak Water Table Flux (Ci/yr)	Time (yr)	Peak Dose at FMB (mrem/yr)	Time (yr)
OBG	1.52×10^{-2}	358	2.86×10^{-1}	364
HT9	8.28×10^{-3}	316	1.14×10^{-1}	506
HT13	2.62×10^{-2}	316	3.80×10^{-1}	492

Inherent in a probabilistic analysis are "tails" that represent low probabilities of both low and high results. Such an analysis provides additional information relative to the uncertainty in the CA. Generally, the high dose tail (i.e., results exceeding the 100 mrem/year dose limit) are a small fraction of the total set of results. For example, in Figure 6.6-18, only 15 of the 1,000 results exceed 100 mrem. SRS is beginning a program conducted under the maintenance program to improve the analysis of uncertainty in PAs and CAs. As appropriate, results will be incorporated into the PAs and CAs.

2.3 References

Flach, G. P., 1998, Impact of F- and H-Area Pump-Treat-Reinject Remediation Systems on the Old Radioactive Waste Burial Ground (U), SRT-EST-98-154.

Flach, G. P. and M. K. Harris, 1997, Integrated hydrogeological model of the General Separations Area (U); Volume 2: Groundwater flow model, WSRC-TR-96-0399.

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WSRC, 1996. *Appendix L, Naval Reactor Waste Disposal (U)*, WSRC-RP-94-218, Westinghouse Savannah River Company, Aiken, SC.

WSRC, 1994. *Radiological Performance Assessment for the E-Area Vaults Disposal Facility (U)*, WSRC-RP-94-218, Westinghouse Savannah River Company, Aiken, SC.

Appendix A.

Westinghouse Savannah River Company

INTER-OFFICE MEMORANDUM

SRT-SCS-99-047

Statistical Consulting Section

June 24, 1999

To: J.R. Cook, 773-43A

From: C.P. Reeve, 773-42A

Generation of Lognormal and Logtriangular Pseudorandom Deviates Based on 50% and 100% Coverage Intervals (U)

____ ________________

S.P. Harris, Technical Reviewer Date

____ ________________

R.C. Tuckfield, Manager Date

Distribution:

B.T. Butcher, 773-43A
L.B. Collard, 773-43A
S.P. Harris, 773-42A
R.C. Tuckfield, 773-42A
E.L. Wilhite, 773-43A

1. Introduction.

The purpose of this report is to describe the methodology for generating pseudorandom numbers from probability distributions of the current inventory of three radionuclides at eight SRS facilities. For each radionuclide/facility combination, Jim Cook provided the assumed distribution type, an estimate of the median activity level, and relative ranges that included the best-estimate inventory level with 50% and 100% probability. All activities have units of *curies.*

When the lognormal distribution was assumed, the mean and standard deviation of the parent normal distribution were obtained from the input median and 50% relative range. When the logtriangular distribution was assumed, the median and range of the parent triangular distribution were obtained from the input median and 100% range. Pseudorandom deviates from the parent distributions were exponentiated to produce deviates from the desired distributions. Computational details are given in section 2.

The input parameters describing the assumed distributions are summarized in Table 1 below. Note the correspondence between the data qualification value and the range factor.

Table 1. Input Probability Distributions and Parameters

Data Qualifi-Cation Value	Case ID	Area/Location	Isotope	Median Activity (m)	Activity Range Factor (f)	50% Probability Range for Lognormal Distribution [m/f, mf]	100% Probability Range for Logtriangular Distribution [m/f, mf]
1	1N	Lysimeters	C-14	1.75	2	[0.875, 3.5]	---
2	2N	Old Burial Ground	H-3	2.1E6	5	[4.2E5, 1.05E7]	---
2	3N	Old Burial Ground	C-14	3100	5	[620, 1.55E4]	---
2	4N	Old Burial Ground	Np-237	1.6	5	[0.32, 8]	---
2	5N	MWMF	H-3	2,300,000	5	[4.6E5, 1.15E7]	---
2	6N	MWMF	C-14	3700	5	[740, 1.85E4]	---
2	7N	MWMF	Np-237	0.096	5	[0.0192, 0.48]	---
3	1T	HLW Tanks 9-12	Np-237	0.034	20	---	[0.0017, 0.68]
3	2T	HLW Tanks 13-16	Np-237	0.02	20	---	[0.001, 0.4]
5	3T	H Canyon	Np-237	0.36	50	---	[0.0072, 18.0]
5	4T	235-F	Np-237	12.0	50	---	[0.24, 600]
7	5T	Tritium Facilities	H-3	30,000	100	---	[300, 3.0E6]

2. Discussion.

2.1. Generation of Lognormal Deviates.

For cases 1N-7N in Table 1, pseudorandom deviates were generated from a *lognormal* distribution for which m/f, m, and mf are the 25%, 50% and 75% quantiles, respectively. The corresponding quantiles for the parent *normal* distribution are $\ln m - \ln f$, $\ln m$, and $\ln m+\ln f$. The central 50% of a normal distribution with mean, μ, and standard deviation, σ, is contained in the interval $[\mu - 0.6745\sigma, \mu+0.6745\sigma]$. It follows that the parent normal distribution has $\mu = \ln m$ and $\sigma = \ln f/0.6745$. If Z_i is a standard normal deviate (mean zero and standard deviation one), as shown in Figure 1a, then the desired lognormal deviate, Y_i, is obtained by

$$Y_i = \exp\{\mu + Z_i\sigma\} = \exp\{\ln m + Z_i \ln f / 0.6745\}.$$

One thousand such lognormal deviates were generated for each of cases 1N-7N. The standard normal deviates were generated by a commercially available computer subroutine.

The probability density function for the standard lognormal distribution is plotted in Figure 1b.

2.2. Generation of Logtriangular Deviates.

For cases 1T-5T in Table 1, pseudo-random deviates were generated from a *logtriangular* distribution for which *m/f*, *m*, and *mf* are the 0%, 50% and 100% quantiles, respectively. The corresponding quantiles for the parent *triangular* distribution are ln*m* – ln*f*, ln*m*, and ln*m*+ln*f*. If T_i is a standard triangular deviate (mean zero, range [-1,1]), as shown in Figure 2a, then the desired logtriangular deviate, Y_i, is obtained by

$$Y_i = \exp\{\ln m + T_i \ln f\}.$$

One thousand such logtriangular deviates were generated for each of cases 1T-5T using the following computer algorithm:

1) Generate a standard uniform deviate, U_i (mean zero and range [0,1]);
2) If $U_i \leq 0.5$, set $T_i = \sqrt{2U_i} - 1$; otherwise, set $T_i = 1 - \sqrt{2(1-U_i)}$;
3) Set $Y_i = \exp\{\ln m + T_i \ln f\}$.

The standard uniform deviates were generated by a commercially available computer subroutine.

The probability density function for the standard logtriangular distribution is plotted in Figure 2b.

Summary.

Methods have been presented for generating pseudorandom deviates from lognormal and logtriangular distributions with specified parameters. The deviates, 1000 for each case, have been transmitted to you electronically. For the purpose of graphical illustration, the value of the probability density function (p.d.f.) corresponding to each deviate was also transmitted.

Figure 1a. Probability Density Function of the Standard Normal Distribution

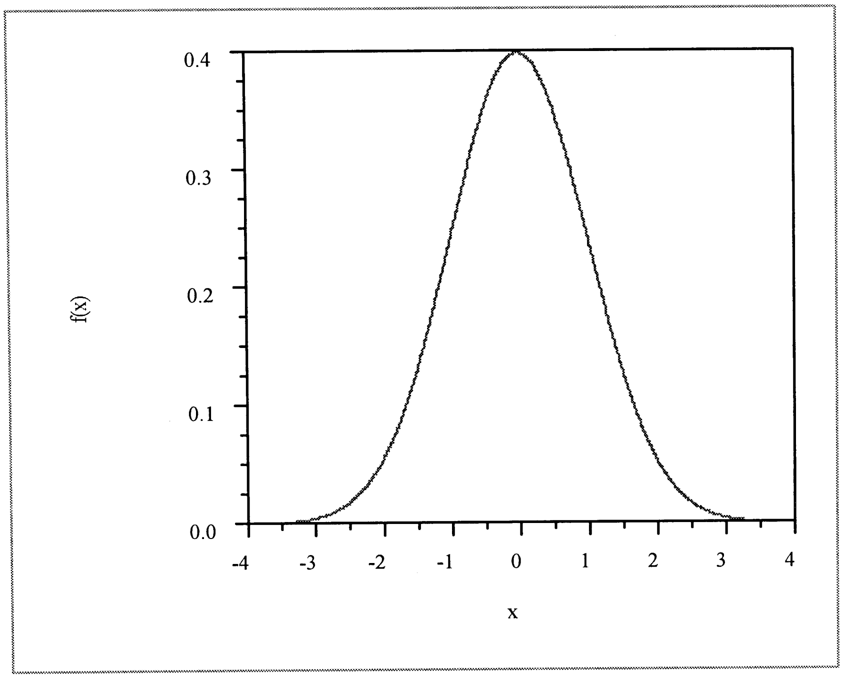


Figure 1b. Probability Density Function of the Standard Lognormal Distribution

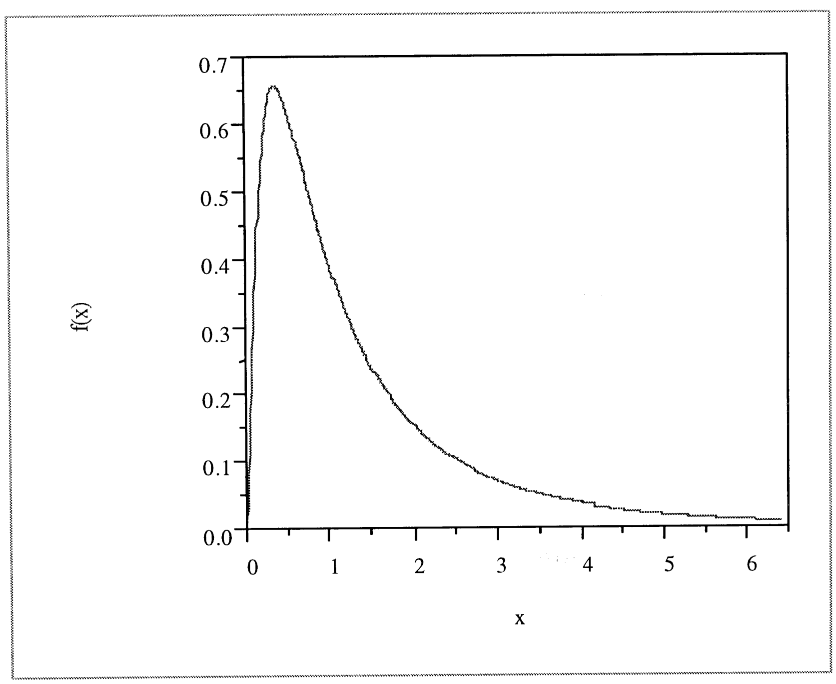


Figure 2a. Probability Density Function of the Symmetric Triangular Distribution in the Interval [-1, 1]

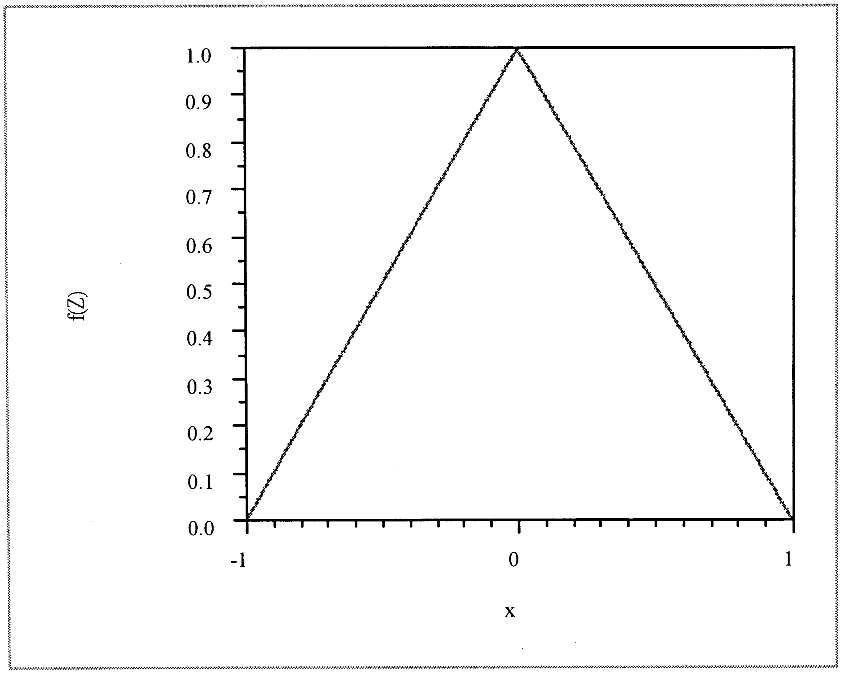
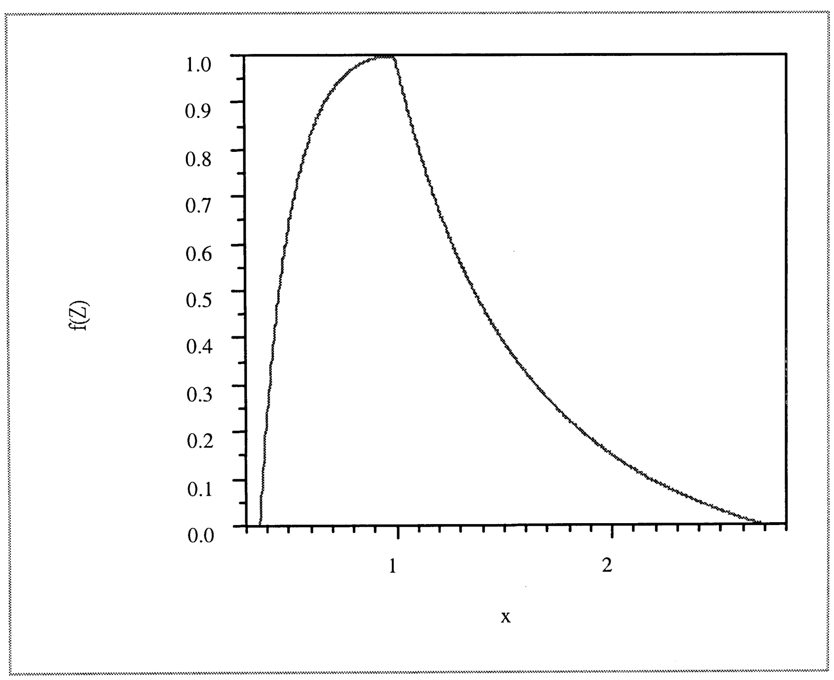


Figure 2b. Probability Density Function of the Standard Logtriangular Distribution



3.0 Condition 3

Source Term/Inventory – Provide a complete source term for the composite analysis to include a complete inventory of the Upper Three Runs watershed and a reanalysis of the source term that was arbitrarily assigned to Cs and Sr to provide a more realistic radionuclide distribution.

3.1 Complete Inventory of Upper Three Runs Watershed

Residual radioactivity left in the A and M areas of SRS will eventually migrate through the groundwater pathway and discharge to Tims Branch, and on to UTR and the Savannah River. A study to estimate the magnitude of these impacts was undertaken to place an upper bound on them.

Three major facilities could contribute residual radioactivity in future times, the closed M-Area Seepage Basin and Lost Lake complex, the Savannah River Laboratory (SRL) Seepage Basins, and the Savannah River Technology Center (SRTC) facility. The M-Area facilities were analyzed as part of the site-wide Environmental Impact Statement on Waste Management Activities and Groundwater Protection. The SRL Basins and the SRTC facility were analyzed using the PATHRAE code to calculate releases and environmental concentrations of radionuclides. The results are summarized in Table 3.1-1.

The M-Area results are taken directly from the M-Area Environmental Information Document (Pickett et al., 1987), using the "No Waste Removal and Closure" option, which most closely describes the actual actions taken at the seepage basin and Lost Lake. The SRL Basin model used the residual inventory remaining after the most contaminated upper one foot has been removed. The basins were assumed to be backfilled with 3 meters of material. No low permeability cap was assumed. The SRTC model assumed that the residual contamination was contained on a 1 meter thick concrete slab with the dimensions of the central corridor of the 773-A building (i.e., it was assumed that the radionuclides were concentrated into a smaller area than that of the entire building).

The former processing buildings in M Area (313-M, 320-M, and 321-M) were thoroughly surveyed and cleaned in preparation for privatization of the buildings. Estimates from surveys conducted as part of the preparation indicate that at most a few kilograms of uranium remain in the buildings. Because this low inventory is associated with the concrete structure, it would be modeled using a solubility limit, thus producing an extremely low source term. Comparison with the results from 247-F, which has a much higher inventory of enriched uranium, indicates that the M-Area process buildings would have been screened out and no further calculations would have been performed. Therefore, the process buildings were not analyzed further in this calculation.

The M-Area waste tanks that contained electroplating waste from the processing facilities are now inactive. All of the waste has been removed and vitrified for disposal as mixed waste. The disposal facility has not yet been determined; SRS has no plans to dispose of mixed waste. The tanks have been cleaned and are awaiting final disposition. Since the tanks are above-ground, it is expected that they will be removed and excessed as scrap metal. Therefore, neither the waste tanks nor the vitrified waste was analyzed further in this calculation. If the disposition of the tanks or vitrified waste changes, the change will be addressed per the Maintenance Program for the E-Area Vaults and Saltstone Performance Assessments, and the Composite Analysis (Attachment 1).

Table 3.1-1 Estimated Peak Concentrations and Peak Times from A and M Areas

Radionuclide	Tims Branch Peak Concentration (Ci/m^3)	Upper Three Runs Peak Concentration (Ci/m^3)	Time of Peak Concentration (yr)
M Area Facilities			
^{238}U	4.4×10^{-15}	8.6×10^{-17}	186
SRL Basins			
^{239}Pu	3.5×10^{-13}	6.8×10^{-15}	36,000
^{240}Pu	2.2×10^{-14}	4.3×10^{-16}	36,000
^{234}U	1.9×10^{-12}	3.7×10^{-14}	12,700
^{235}U	2.2×10^{-13}	4.3×10^{-15}	12,700
^{238}U	2.6×10^{-12}	5.1×10^{-14}	12,700
SRTC			
^{3}H	1.2×10^{-10}	2.4×10^{-12}	120
^{235}U	2.8×10^{-16}	5.5×10^{-18}	2,500
^{238}U	1.6×10^{-14}	3.1×10^{-16}	2,500
^{237}Np	2.8×10^{-15}	5.5×10^{-17}	3,200
^{239}Pu	7.7×10^{-13}	1.5×10^{-14}	52,000
^{240}Pu	3.8×10^{-15}	7.4×10^{-17}	49,000
^{242}Pu	7.9×10^{-16}	1.6×10^{-17}	55,000

Comparison of the results in the table with the results for UTR in Table 5.3-2 in the CA shows that the contribution to UTR from A and M Areas is many orders of magnitude less than the contribution from the GSA.

3.2 Reanalysis of the Source Term that was Arbitrarily Assigned to Cs and Sr

Facilities where the inventory was attributed to only ^{137}Cs or ^{90}Sr were reformulated using the fission product distribution table in Stewart (Stewart 1985). This resulted in additional entries in Tables 4.4-2 and 4.4-5 in the CA. The revised tables are included here (the tables have the same table numbers as in the CA). The result is that in a few cases, additional radionuclide sources would not have been screened out. In some cases the recalculated inventory produced fluxes to the water table greater that the screening value of 1 x 10^{-4} Ci/year. The additional sources are ^{79}Se in the H-Area Sand Filter, the F-Area Sand Filter and the spills at Tanks 16, 37 and 8, ^{90}Sr at the H-Area Sand Filter, and the spills at Tanks 13, 9, 16, 37, and B281-F, ^{99}Tc at all of the Solvent Tanks, the H-Area Sand Filter, the F-Area Sand Filter and the spills at Tanks 13,16, 37, and 8, and ^{129}I at the H-Area Sand Filter, the F-Area Sand Filter, and the spills at Tanks 13, 16, 37, and 8.

The magnitude of the flux to the water table results for the radionuclides and facilities listed above are less than others that were analyzed and yielded low overall impacts. The conclusion of this supplemental work is that the omission of these sources did not affect the doses presented in the CA.

Table 4.4-2 Residual Radionuclide Summary

	E Area												
	Lysimeters	MWMF		Naval Fuel Waste	Naval Reactors KAPL CB/TS	Naval Reactors KAPL Head	Old Burial Ground	Old Solvent Tanks S1-S22	Saltstone Lysimeters	Slit Trenches	Solvent Tanks S23-S30 and S32	Vaults LAW	Vaults ILV
Building Number	643-7E	643-7E and 643-28E		643-7E	643-7E	643-7E	643-E	643-E	643-7E	643-7E	643-7E	661-6E	662-6E
Site Map Page No.	10	10, I-3		10, G-12 and G-13	12,B-10	12,B-10	10, A-12	10	10	10	10	10	10
Dates of Operation	1978 - 1980	1972 - 1988	1988 - 1996	1989	1994 - 2014	1994 - 2014	1952 - 1972	1955 - 1981	1983	1995 - 2015	1981 - 1997	1995 - 2015	1995 - 2015
Total Volume	DU	DU	DU	41.9 Kg	DU	DU	DU	DU	7500 gal	26000 m^3	DU	34000 m^3	7464 m^3
Reference No.	B-22, B-49	B-16, B-49	B-16, B-49	B-40	B-26	B-26	B-16	B-36	B-24	B-43,B-32	B-39	B-35	B-32
RADIONUCLIDE													
H-3	---	2.06E+06	2.34E+05	---	4.32E+02	6.67E-04	2.12E+06		7.39E-01	8.75E+00	---	1.66E+06	8.80E+05
C-14	1.75E+00	1.86E+03	1.86E+03	---	4.33E+02	1.49E+00	3.09E+03	---	2.53E-04	---	---	1.70E-01	2.24E-03
Na-22	1.02E-03	---	---	---	---	---	---	---	---	---	---	---	---
Al-26	---	---	---	---	---	---	---	---	---	---	---	---	---
K-40	---	---	---	---	---	---	---	---	---	---	---	---	---
Sc-46	3.50E-02	---	---	---	---	---	---	---	---	---	---	---	---
Cr-51	---	---	---	---	2.40E+04	3.74E+01	---	---	---	---	---	---	---
Mn-54	2.19E-01	2.62E+01	1.20E+00	---	4.39E+03	1.49E+01	5.59E+01	---	---	---	---	---	---
Fe-55	---	---	---	---	2.88E+05	2.98E+02	---	---	---	---	---	---	---
Fe-59	---	---	---	---	2.40E+04	3.73E+01	---	---	---	---	---	---	---
Co-57	2.12E+00	---	---	---	---	---	---	---	---	---	---	---	---
Co-58	---	---	---	---	6.50E+04	5.94E+02	---	---	---	---	---	---	---
Co-60	2.94E+00	1.88E+06	7.18E+04	---	3.14E+05	1.49E+02	1.66E+06	---	7.96E-03	4.63E-02	---	8.66E+00	1.38E+01
Ni-59	---	1.74E+03	7.96E+01	---	4.99E+03	4.46E-01	3.71E+03	---	7.67E-06	---	---	1.06E-01	5.66E-02
Ni-63	---	2.37E+05	1.09E+04	---	5.76E+05	4.46E+01	5.06E+05	---	7.67E-04	---	---	---	---
Zn-65	2.60E+00	---	---	---	---	---	---	---	---	---	---	---	---
Se-79	---	1.07E-01	6.66E-03	---	3.94E-03	2.23E-07	7.21E-01	5.11E-05	1.25E-02	---	2.09E-05	2.85E-02	6.46E-03
Sr-89	---	---	---	---	---	---	---	---	---	---	---	---	---
Sr-90	3.93E-02	1.81E+04	1.02E+03	---	1.69E+01	5.94E-02	1.10E+05	7.80E+00	2.64E-02	2.88E-01	3.19E+00	1.00E+02	1.47E+04
Y-90	---	---	---	---	1.69E+01	5.94E-02	---	---	---	---	---	7.63E+01	---
Zr-93	---	---	---	---	2.40E+04	2.98E-04	---	2.43E-04	1.02E-05	---	9.94E-05	1.16E-05	---
Zr-95	7.99E-01	---	---	---	1.98E+05	1.49E+01	---	---	---	---	---	---	---
Nb-93m	---	---	---	---	2.40E+04	2.23E+00	---	---	---	---	---	---	---
Nb-94	---	---	---	---	2.08E+01	2.98E-02	---	---	---	---	---	---	---
Nb-95	1.02E+00	---	---	---	4.19E+05	3.20E+01	---	---	---	---	---	---	---
Nb-95m	---	---	---	---	4.19E+03	---	---	---	---	---	---	---	---
Mo-93	---	---	---	---	4.61E+00	---	---	---	---	---	---	---	---
Tc-99	---	3.83E+00	2.39E-01	---	4.58E-01	1.49E-03	2.59E+01	1.83E-03	2.53E+00	9.73E-04	7.52E-04	3.41E-02	2.18E-01
Ru-103	4.14E-01	---	---	---	---	---	---	---	---	---	---	---	---
Ru-106	1.12E+00	---	---	---	1.34E+00	---	---	7.07E-02	1.28E+00	---	2.89E-02	1.66E-01	---
Rh-106	---	---	---	---	---	---	---	---	---	---	---	6.89E-04	---
Pd-107	---	---	---	---	---	---	---	1.41E-05	7.67E-07	---	5.78E-06	---	---
Ag-110m	---	---	---	---	---	---	---	2.18E-05	2.27E-05	---	8.94E-06	---	---
In-113m	---	---	---	---	1.56E+04	---	---	---	---	---	---	---	---
Sn-113	---	---	---	---	1.56E+04	---	---	---	---	---	---	---	---
Sn-119m	---	---	---	---	2.60E+05	---	---	---	---	---	---	---	---

Table 4.4-2 Residual Radionuclide Summary

	E Area												
	Lysimeters	MWMF		Naval Fuel Waste	Naval Reactors KAPL CB/TS	Naval Reactors KAPL Head	Old Burial Ground	Old Solvent Tanks S1-S22	Saltstone Lysimeters	Slit Trenches	Solvent Tanks S23-S30 and S32	Vaults LAW	Vaults ILV
Building Number	643-7E	643-7E and 643-28E		643-7E	643-7E	643-7E	643-E	643-E	643-7E	643-7E	643-7E	661-6E	662-6E
Site Map Page No.	10	10, I-3		10, G-12 and G-13	12,B-10	12,B-10	10, A-12	10	10	10	10	10	10
Dates of Operation	1978 - 1980	1972 - 1988	1988 - 1996	1989	1994 - 2014	1994 - 2014	1952 - 1972	1955 - 1981	1983	1995 - 2015	1981 - 1997	1995 - 2015	1995 - 2015
Total Volume	DU	DU	DU	41.9 Kg	DU	DU	DU	DU	7500 gal	26000 m^3	DU	34000 m^3	7464 m^3
Reference No.	B-22, B-49	B-16, B-49	B-16, B-49	B-40	B-26	B-26	B-16	B-36	B-24	B-43,B-32	B-39	B-35	B-32
RADIONUCLIDE													
Sn-121m	---	---	---	---	---	---	---	---	1.02E-03	---	---	---	---
Sn-123	---	---	---	---	7.55E+03	---	---	---	---	---	---	---	---
Sn-126	---	1.46E-01	9.14E-03	---	2.34E-05	6.70E-07	9.88E-01	7.02E-05	5.12E-03	---	2.87E-05	3.31E-04	8.58E-03
Sb-125	4.93E-02	1.55E+03	7.09E+01	---	1.31E+05	1.49E+00	3.30E+03	9.00E-06	2.53E-01	---	3.68E-06	---	---
Sb-126	---	---	---	---	---	---	---	---	5.12E-04	---	---	---	---
Sb-126m	---	---	---	---	---	---	---	---	---	---	---	---	---
Te-125m	---	7.16E+02	3.41E+01	---	8.16E+04	3.43E-01	1.88E+03	3.61E-02	7.67E-03	---	1.48E-02	---	---
Te-127	---	---	---	---	---	---	---	---	---	---	---	---	---
Te-127m	---	---	---	---	---	---	---	---	---	---	---	---	---
I-129	---	9.94E-02	6.21E-03	---	6.30E-06	5.94E-06	6.72E-01	4.77E-05	7.67E-04	1.15E-06	1.95E-05	4.43E-05	1.39E-04
Cs-134	5.56E-02	2.24E+03	1.40E+02	---	---	---	1.52E+04	1.08E+00	2.53E-03	---	4.41E-01	---	---
Cs-135	---	---	---	---	1.11E-04	---	---	3.68E-05	1.51E-06	---	1.50E-05	---	---
Cs-137	1.30E+00	2.29E+04	1.43E+03	---	1.66E+01	5.94E-02	1.55E+05	1.10E+01	1.02E+00	1.02E+01	4.50E+00	3.10E+02	2.52E+04
Ba-137m	---	---	---	---	1.66E+01	5.94E-02	---	---	---	---	---	7.08E+01	---
Ce-144	4.44E+00	---	---	---	1.58E+01	---	---	1.94E-02	---	---	7.94E-03	6.14E+00	---
Pr-144	---	---	---	---	---	---	---	---	---	---	---	5.96E+00	---
Pr-144m	---	---	---	---	---	---	---	---	---	---	---	2.99E-05	---
Pm-147	---	---	---	---	9.54E+00	---	---	9.96E-01	1.53E-01	---	4.07E-01	3.31E+01	---
Sm-151	---	3.11E+02	1.94E+01	---	1.73E-01	---	2.10E+03	1.49E-01	7.67E-02	---	6.10E-02	---	---
Eu-152	---	---	---	---	---	---	---	---	2.27E-04	---	---	---	---
Eu-154	4.80E-03	1.21E+03	7.58E+01	---	2.16E-01	---	8.20E+03	5.82E-01	2.53E-02	1.53E-02	2.38E-01	---	---
Eu-155	---	4.37E+01	2.73E+00	---	1.23E-01	---	2.95E+02	2.09E-02	1.25E-02	---	8.57E-03	---	---
Hf-181	---	---	---	---	2.40E+04	1.49E+01	---	---	---	---	---	---	---
Ta-182	---	---	---	---	5.66E+04	8.42E-02	---	---	---	---	---	---	---
Pb-212	---	---	---	---	---	---	---	---	---	9.35E-03	---	---	---
Pb-214	---	---	---	---	---	---	---	---	---	---	---	---	---
Bi-214	---	---	---	---	---	---	---	---	---	---	---	---	---
Ra-226	---	---	---	---	---	---	---	---	---	---	---	---	---
Ra-228	---	---	---	---	---	---	---	---	---	---	---	---	---
Ac-228	---	---	---	---	---	---	---	---	---	---	---	---	---
Th-228	---	---	---	---	---	---	---	---	5.12E-08	---	---	---	---
Th-230	---	---	---	---	---	---	---	---	---	---	---	---	---
Th-231	---	---	---	---	---	---	---	---	5.12E-06	---	---	---	---
Th-232	---	2.46E+00	1.46E+00	---	8.51E-11	1.42E-10	3.61E+00	---	---	---	---	3.17E-02	---
Th-234	---	---	---	---	---	---	---	---	7.67E-08	---	---	---	---
Pa-234	---	---	---	---	---	---	---	---	1.51E-07	---	---	---	---

Table 4.4-2 Residual Radionuclide Summary

	E Area												
	Lysimeters	MWMF		Naval Fuel Waste	Naval Reactors KAPL CB/TS	Naval Reactors KAPL Head	Old Burial Ground	Old Solvent Tanks S1-S22	Saltstone Lysimeters	Slit Trenches	Solvent Tanks S23-S30 and S32	Vaults LAW	Vaults ILV
Building Number	643-7E	643-7E and 643-28E		643-7E	643-7E	643-7E	643-E	643-E	643-7E	643-7E	643-7E	661-6E	662-6E
. Site Map Page No.	10	10, I-3		10, G-12 and G-13	12,B-10	12,B-10	10, A-12	10	10	10	10	10	10
Dates of Operation	1978 - 1980	1972 - 1988	1988 - 1996	1989	1994 - 2014	1994 - 2014	1952 - 1972	1955 - 1981	1983	1995 - 2015	1981 - 1997	1995 - 2015	1995 - 2015
Total Volume	DU	DU	DU	41.9 Kg	DU	DU	DU	DU	7500 gal	26000 m^3	DU	34000 m^3	7464 m^3
Reference No.	B-22, B-49	B-16, B-49	B-16, B-49	B-40	B-26	B-26	B-16	B-36	B-24	B-43,B-32	B-39	B-35	B-32
RADIONUCLIDE													
U-232	---	---	---	---	1.39E-06	2.23E-06	---	---	1.76E-06	---	---	2.48E-05	---
U-233	---	1.55E+00	4.90E-01	---	---	---	2.33E-01	---	1.02E-07	---	---	1.75E-03	2.56E-04
U-234	---	2.79E+01	2.25E+01	4.46E+00	8.90E-06	---	1.98E+01	---	1.02E-05	---	---	7.79E-01	1.12E-04
U-235	---	1.06E+00	4.99E-01	8.81E-02	6.59E-07	---	6.14E-01	---	---	---	---	1.23E-02	3.00E-06
U-236	---	4.70E+00	1.18E+00	1.41E-04	1.35E-05	---	2.85E+00	---	---	---	---	3.59E-02	5.84E-06
U-238	---	4.16E+01	4.63E+00	---	7.46E-05	---	1.57E+01	---	7.67E-08	---	---	6.29E-02	1.55E-04
Np-237	---	9.57E-02	1.68E-04	---	1.29E-05	4.46E-09	1.57E+00	---	2.27E-06	8.85E-07	---	8.69E-03	1.75E-03
Np-239	---	---	---	---	---	---	---	---	---	---	---	---	---
Pu-236	---	---	---	---	---	---	---	---	---	---	---	---	---
Pu-238	3.38E+00	3.97E+03	3.05E+02	---	8.61E-01	3.73E-04	1.62E+04	2.75E+02	1.90E-03	5.16E-03	1.13E+02	6.01E+00	1.43E+01
Pu-239	1.80E+00	6.09E+01	9.03E-01	---	3.97E-01	5.94E-05	1.30E+03	5.50E+01	5.17E-03	---	2.25E+01	1.54E+00	2.15E+00
Pu-240	---	1.51E+01	2.67E-01	---	3.55E-01	3.73E-05	3.11E+02	---	1.25E-05	---	---	3.04E-01	4.60E-02
Pu-241	---	6.14E+02	1.30E+01	---	1.09E+02	1.49E-02	1.19E+04	---	1.25E-03	---	---	1.52E+01	3.68E+00
Pu-242	---	1.25E-03	---	---	1.30E-03	4.46E-07	---	---	---	---	---	3.00E-05	7.66E-05
Pu-244	---	---	---	---	8.87E-11	6.70E-14	---	---	---	---	---	2.59E-15	---
Am-241	9.27E-04	2.01E+01	1.97E-01	---	1.13E+00	5.21E-04	2.30E+02	---	5.12E-03	2.57E-01	---	3.01E-01	4.38E+00
Am-242	---	---	---	---	---	---	---	---	2.53E-06	---	---	---	---
Am-242m	---	---	---	---	7.17E-06	2.98E-06	---	---	2.53E-06	---	---	---	---
Am-243	---	---	9.95E-04	---	7.71E-03	4.46E-06	---	---	1.51E-06	---	---	1.73E-07	---
Cm-242	---	---	---	---	1.67E+01	9.67E-03	---	---	2.53E-06	---	---	---	---
Cm-243	---	---	---	---	2.24E-06	3.73E-06	---	---	1.02E-06	---	---	---	---
Cm-244	---	1.82E+04	3.79E+03	---	6.14E-01	5.21E-04	2.54E+04	2.20E+02	2.53E-05	---	9.00E+01	---	---
Cm-245	---	---	---	---	3.27E-05	3.73E-08	---	---	---	---	---	1.45E-09	---
Cm-246	---	---	---	---	1.26E-05	1.49E-08	---	---	---	---	---	5.77E-10	---
Cm-247	---	---	---	---	2.53E-11	4.46E-14	---	---	---	---	---	6.14E-12	---
Cm-248	---	---	---	---	5.96E-11	1.42E-13	---	---	---	---	---	5.50E-15	---
Cf-249	---	---	---	---	3.97E-10	7.46E-13	---	---	---	---	---	2.88E-14	---
Cf-251	---	---	---	---	8.47E-12	2.98E-14	---	---	---	---	---	1.40E-09	---
Cf-252	---	1.79E+01	3.39E+01	---	---	---	7.53E+00	---	---	---	---	---	---

Table 4.4-2 Residual Radionuclide Summary

		F AREA											
						F-Area Tanks							
		235-F	772-F Lab	772-1F Lab	Canyon (Separations)	Tank # 1-8	Tank # 17-20	Tank# 25-28 and 44-47	Tank # 33-34	Naval Fuel Materials Facility	Inactive Process Sewer Lines	Sand Filters	Seepage Basin GW Op. Unit
	Building Number	235-F	772-F	772-1F	221-F	NA	NA	NA	NA	247-F	081-F	294-F	DU
	Site Map Page No.	11, D-12	11, D11	11, B-8	12,G-5	13,G-6	13,E-4	13,G-4	13,I-6	11, B-10	Closed	11, E-10	Closed
	Dates of Operation	DU	DU	DU	Early 50s-2005	DU	DU	DU	DU	DU	1955 - 1982	1975 - 1990	1954 - 1988
	Total Volume	DU	DU	DU	DU	800 gal	5000 gal	8000 gal	200 gal	17,071 g	DU	DU	DU
	Reference No.	B-52	B-48	B-48	B-15	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-51	B-33	B-45	B-18
	H-3	---	1.06E+01	1.00E-01	6.79E+01	---	---	---	---	---	1.11E+01	---	---
	C-14	---	---	---	2.85E-01	1.15E-03	7.80E-03	3.34E-02	0.00E+00	---	---	---	---
	Na-22	---	---	---	---	---	---	---	---	---	---	---	---
	Al-26	---	---	---	---	---	---	---	---	---	---	---	---
	K-40	---	---	---	---	---	---	---	---	---	---	---	---
	Sc-46	---	---	---	---	---	---	---	---	---	---	---	---
	Cr-51	---	---	---	---	---	---	---	---	---	---	---	---
	Mn-54	---	---	---	---	---	---	---	---	---	---	---	---
	Fe-55	---	---	---	---	---	---	---	---	---	---	---	---
	Fe-59	---	---	---	---	---	---	---	---	---	---	---	---
	Co-57	---	---	---	---	---	---	---	---	---	---	---	---
	Co-58	---	---	---	---	---	---	---	---	---	---	---	---
R	Co-60	---	---	---	1.14E+02	6.44E+01	6.15E+00	1.70E+02	2.89E+02	---	---	---	---
A	Ni-59	---	---	---	3.36E-01	1.07E+00	4.59E-01	1.96E+00	4.35E-01	---	---	---	---
D	Ni-63	---	---	---	---	0.00E+00	0.00E+00	0.00E+00	0.00E+00	---	---	---	---
I	Zn-65	---	---	---	---	---	---	---	---	---	---	---	---
O	Se-79	---	3.25E-06	2.63E-07	7.64E-02	7.23E-01	3.79E-02	1.62E-01	3.12E-01	---	---	2.23E-02	---
N	Sr-89	---	---	---	---	---	---	---	---	---	---	---	---
U	Sr-90	---	4.96E-01	4.01E-02	9.29E+03	3.48E+04	2.10E+03	1.29E+04	2.43E+04	---	5.22E+01	3.40E+03	1.03E+00
C	Y-90	---	---	---	9.29E+03	3.48E+04	2.10E+03	1.29E+04	2.43E+04	---	---	---	---
L	Zr-93	---	---	---	---	---	---	---	---	---	---	1.06E-01	---
I	Zr-95	---	---	---	---	---	---	---	---	---	---	---	---
D	Nb-93m	---	---	---	---	---	---	---	---	---	---	---	---
E	Nb-94	---	---	---	---	---	---	---	---	---	---	---	---
	Nb-95	---	---	---	---	---	---	---	---	---	---	---	---
	Nb-95m	---	---	---	---	---	---	---	---	---	---	---	---
	Mo-93	---	---	---	---	---	---	---	---	---	---		---
	Tc-99	---	1.17E-04	9.44E-06	2.85E+00	1.25E+01	6.58E-01	2.81E+00	5.39E+00	---	2.21E-01	8.02E-01	8.80E-02
	Ru-103	---	---	---	---	---	---	---	---	---	---	---	---
	Ru-106	---	---	3.63E-04	5.95E+02	1.46E-01	5.53E-03	3.05E+02	3.15E+02	---	2.21E-01	1.02E+01	---
	Rh-106	---	4.50E-03	---	5.95E+02	1.46E-01	5.53E-03	3.05E+02	3.15E+02	---	---	---	---
	Pd-107	---	---	---	---	---	---	---	---	---	---	6.17E-03	---
	Ag-110m	---	---	---	---	---	---	---	---	---	---	9.53E-03	---
	In-113m	---	---	---	---	---	---	---	---	---	---	---	---
	Sn-113	---	---	---	---	---	---	---	---	---	---	---	---
	Sn-119m	---	---	---	---	---	---	---	---	---	---	---	---

Table 4.4-2 Residual Radionuclide Summary

	F AREA											
					F-Area Tanks							
	235-F	772-F Lab	772-1F Lab	Canyon (Separations)	Tank # 1-8	Tank # 17-20	Tank# 25-28 and 44-47	Tank # 33-34	Naval Fuel Materials Facility	Inactive Process Sewer Lines	Sand Filters	Seepage Basin GW Op. Unit
Building Number	235-F	772-F	772-1F	221-F	NA	NA	NA	NA	247-F	081-F	294-F	DU
Site Map Page No.	11, D-12	11, D11	11, B-8	12,G-5	13,G-6	13,E-4	13,G-4	13,I-6	11, B-10	Closed	11, E-10	Closed
Dates of Operation	DU	DU	DU	Early 50s-2005	DU	DU	DU	DU	DU	1955 - 1982	1975 - 1990	1954 - 1988
Total Volume	DU	DU	DU	DU	800 gal	5000 gal	8000 gal	200 gal	17,071 g	DU	DU	DU
Reference No.	B-52	B-48	B-48	B-15	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-51	B-33	B-45	B-18
RADIONUCLIDE												
Sn-121m	---	---	---	---	---	---	---	---	---	---	---	---
Sn-123	---	---	---	---	---	---	---	---	---	---	---	---
Sn-126	---	4.46E-06	3.61E-07	9.17E-03	1.35E+00	7.05E-02	3.01E-01	5.80E-01	---	---	3.06E-02	---
Sb-125	---	5.72E-07	4.62E-08	---	3.71E+01	4.00E+00	5.74E+02	8.72E+02	---	---	3.93E-03	---
Sb-126	---	---	---	---	---	---	---	---	---	---	---	---
Sb-126m	---	---	---	---	---	---	---	---	---	---	---	---
Te-125m	---	2.30E-03	1.86E-04	---	---	---	---	---	---	---	1.58E+01	---
Te-127	---	---	---	---	---	---	---	---	---	---	---	---
Te-127m	---	---	---	---	---	---	---	---	---	---	---	---
I-129	---	3.03E-06	2.45E-07	5.20E-03	5.96E-05	3.12E-06	1.33E-05	2.57E-05	---	---	2.08E-02	3.57E-02
Cs-134	---	6.85E-02	5.53E-03	---	1.87E-01	1.82E-02	7.95E+00	1.12E+01	---	---	4.70E+02	---
Cs-135	---	---	---	---	8.28E-03	4.34E-04	1.85E-03	3.57E-03	---	---	1.60E-02	---
Cs-137	---	7.00E-01	5.65E-02	1.73E+04	2.42E+03	1.45E+02	8.72E+02	1.64E+03	---	6.92E+01	4.81E+03	1.49E+01
Ba-137m	---	---	---	1.64E+04	2.29E+03	1.37E+02	8.25E+02	1.56E+03	---	---	---	---
Ce-144	---	1.23E-03	9.97E-05	7.09E+02	1.60E-02	3.21E-04	3.84E+02	3.57E+02	---	---	4.16E+00	---
Pr-144	---	---	---	7.09E+02	1.60E-02	3.21E-04	3.84E+02	3.57E+02	---	---	---	---
Pr-144m	---	---	---	1.02E+01	---	---	---	---	---	---	---	---
Pm-147	---	6.34E-02	5.12E-03	8.16E+03	6.32E+02	6.78E+01	1.12E+04	1.68E+04	---	---	4.35E+02	---
Sm-151	---	9.48E-03	7.66E-04	---	---	---	---	---	---	---	6.50E+01	---
Eu-152	---	---	---	---	---	---	---	---	---	3.81E-02	---	---
Eu-154	---	3.70E-02	2.99E-03	---	1.28E+02	1.01E+01	1.35E+02	2.43E+02	---	---	2.54E+02	---
Eu-155	---	1.33E-03	1.08E-04	---	---	---	---	---	---	7.63E-03	9.14E+00	---
Hf-181	---	---	---	---	---	---	---	---	---	---	---	---
Ta-182	---	---	---	---	---	---	---	---	---	---	---	---
Pb-212	---	---	---	---	---	---	---	---	---	---	---	---
Pb-214	---	---	---	---	---	---	---	---	---	---	---	---
Bi-214	---	---	---	---	---	---	---	---	---	---	---	---
Ra-226	---	---	---	---	---	---	---	---	---	8.72E-02	---	---
Ra-228	---	---	---	---	---	---	---	---	---	---	---	---
Ac-228	---	---	---	---	---	---	---	---	---	---	---	---
Th-228	---	---	---	---	---	---	---	---	---	---	---	---
Th-230	---	---	---	---	---	---	---	---	---	---	---	---
Th-231	---	---	---	---	---	---	---	---	---	---	---	---
Th-232	---	---	---	---	---	---	---	---	---	6.59E-02	---	---
Th-234	---	---	---	---	---	---	---	---	---	---	---	---
Pa-234	---	---	---	---	---	---	---	---	---	---	---	---

Table 4.4-2 Residual Radionuclide Summary

		F AREA										
					F-Area Tanks							
		235-F	772-F Lab	772-1F Lab	Canyon (Separations)	Tank # 1-8	Tank # 17-20	Tank# 25-28 and 44-47	Tank # 33-34	Naval Fuel Materials Facility	Inactive Process Sewer Lines	Sand Filters	Seepage Basin GW Op. Unit
Building Number		235-F	772-F	772-1F	221-F	NA	NA	NA	NA	247-F	081-F	294-F	DU
Site Map Page No.		11, D-12	11, D11	11, B-8	12,G-5	13,G-6	13,E-4	13,G-4	13,I-6	11, B-10	Closed	11, E-10	Closed
Dates of Operation		DU	DU	DU	Early 50s-2005	DU	DU	DU	DU	DU	1955 - 1982	1975 - 1990	1954 - 1988
Total Volume		DU	DU	DU	DU	800 gal	5000 gal	8000 gal	200 gal	17,071 g	DU	DU	DU
Reference No.		B-52	B-48	B-48	B-15	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-51	B-33	B-45	B-18
	U-232	---	---	---	---	6.22E-04	1.17E-04	5.82E-04	3.22E-04	---	---	---	---
	U-233	---	---	---	4.97E-08	---	---	---	---	---	---	---	---
	U-234	---	---	---	4.70E-02	---	---	---	---	1.82E+00	7.90E-02	---	5.58E-02
	U-235	---	---	---	8.62E-04	9.80E-04	7.88E-04	1.23E-03	3.24E-04	3.59E-02	---	---	1.96E-02
	U-236	---	---	---	9.38E-03	---	---	---	---	---	---	---	---
	U-238	---	---	---	1.68E-02	3.04E-02	5.07E-02	1.13E-01	2.63E-02	5.74E-05	8.99E-01	---	9.81E-02
	Np-237	1.20E+01	---	---	3.53E-03	5.25E-02	---	---	2.11E-02	---	2.15E-02	---	---
	Np-239	---	---	---	---	---	---	---	---	---	---	---	---
R	Pu-236	---	---	---	---	---	---	---	---	---	---	---	---
A	Pu-238	1.02E+04	2.07E-01	1.63E-03	1.25E+01	---	---	---	---	---	2.72E-01	---	4.44E-01
D	Pu-239	---	1.23E-02	2.75E-03	1.56E+02	8.10E+00	3.01E+01	1.38E+02	3.62E+00	---	7.57E+00	2.35E+01	1.75E+00
I	Pu-240	---	2.90E-03	6.51E-04	3.60E+01	1.93E+00	7.51E+00	3.09E+01	8.08E-01	---	---	---	---
O	Pu-241	---	1.91E+00	5.25E-01	1.95E+03	5.81E+00	4.26E+02	6.83E+02	1.70E+01	---	---	---	---
N	Pu-242	---	---	---	2.43E-03	1.15E-03	1.16E-02	6.36E-03	1.67E-04	---	---	---	---
U	Pu-244	---	---	---	---	---	---	---	---	---	---	---	---
C	Am-241	---	2.18E-03	4.90E-04	1.58E+00	1.12E+02	7.17E+01	4.19E+02	5.37E+01	---	3.00E-02	---	3.29E-02
L	Am-242	---	---	---	---	---	---	---	---	---	---	---	---
I	Am-242m	---	---	---	1.98E-02	1.22E-01	---	---	5.87E-02	---	---	---	---
D	Am-243	---	---	---	1.94E+00	---	---	---	---	---	---	---	---
E	Cm-242	---	---	---	---	---	---	---	---	---	---	---	---
	Cm-243	---	---	---	---	---	---	---	---	---	---	---	---
	Cm-244	---	---	---	2.11E+02	2.86E-02	1.84E-03	1.39E-02	2.60E-02	---	5.72E-02	---	---
	Cm-245	---	---	---	1.72E-02	2.05E-08	1.09E-09	4.64E-09	8.84E-09	---	---	---	---
	Cm-246	---	---	---	2.69E-02	---	---	---	---	---	---	---	---
	Cm-247	---	---	---	1.27E-07	---	---	---	---	---	---	---	---
	Cm-248	---	---	---	---	---	---	---	---	---	---	---	---
	Cf-249	---	---	---	---	---	---	---	---	---	---	---	---
	Cf-251	---	---	---	---	---	---	---	---	---	---	---	---
	Cf-252	---	---	---	---	---	---	---	---	---	---	---	---

Table 4.4-2 Residual Radionuclide Summary

		H AREA											
					H-Area Tanks								
		Canyon (Separations)	ETF Receipt Tank	Inactive Process Sewer Lines	Tank # 9-12	Tank #13-16	Tank # 21-24, 29-32, and 35-37	Tank # 38-43	Tank # 48-51	New Solvent Tanks H33-H36	Sand Filter	Seepage Basin GW Op. Unit	Tritium Processing
	Building Number	221-H	241-H	081-H	NA	NA	NA	NA	NA	DU	294-H	DU	232H,233H, 234H
	Site Map Page No.	15,F-5	16,F-8 17,G-11	Closed	16,F-12	17,A-5	14,F-6	14,H-9	14,I-9	14,D-12	15,H-10	Closed	15,C-1
	Dates of Operation	Early 50s-2005	1977-Present	1955 - 1982	DU	DU	DU	DU	DU	1997 - 2028	1975 - 1990	1954 - 1988	1955 - 2005
	Total Volume	DU	1000 L	DU	400 gal	400 gal	2000 gal	600 gal	1300 gal	DU	DU	DU	DU
	Reference No.	B-15	B-42	B-33	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-39	B-45	B-18	B-17
RADIONUCLIDE	H-3	1.02E+00	7.00E-02	2.87E+01	---	---	---	---	---	---	---	---	3.00E+04
	C-14	4.28E-03	---	---	7.97E-04	2.88E-04	8.79E-04	5.85E-04	2.08E-04	---	---	---	---
	Na-22	---	---	---	---	---	---	---	---	---	---	---	---
	Al-26	---	---	---	---	---	---	---	---	---	---	---	---
	K-40	---	---	---	---	---	---	---	---	---	---	---	---
	Sc-46	---	---	---	---	---	---	---	---	---	---	---	---
	Cr-51	---	---	---	---	---	---	---	---	---	---	---	---
	Mn-54	---	---	---	---	---	---	---	---	---	---	---	---
	Fe-55	---	---	---	---	---	---	---	---	---	---	---	---
	Fe-59	---	---	---	---	---	---	---	---	---	---	---	---
	Co-57	---	---	---	---	---	---	---	---	---	---	---	---
	Co-58	---	---	---	---	---	---	---	---	---	---	---	---
	Co-60	1.71E+00	1.00E-04	5.15E-01	4.78E+01	2.47E+01	4.36E+02	3.22E+02	7.10E-01	---	---	---	---
	Ni-59	5.04E-03	---	---	5.30E-01	4.45E-01	8.87E-01	4.02E-01	1.37E-02	---	---	---	---
	Ni-63	---	---	---	---	---	---	---	---	---	---	---	---
	Zn-65	---	---	---	---	---	---	---	---	---	---	---	---
	Se-79	1.15E-03	---	---	3.09E-01	2.75E-01	5.10E-01	2.31E-01	2.11E-03	1.86E-04	2.24E-02	---	---
	Sr-89	---	---	---	---	---	---	---	---	---	---	---	---
	Sr-90	1.39E+02	4.00E-04	1.94E+01	1.67E+04	1.40E+04	3.92E+04	2.08E+04	1.41E+02	2.84E+01	3.41E+03	5.35E+01	---
	Y-90	1.39E+02	---	---	1.67E+04	1.40E+04	3.92E+04	2.08E+04	1.41E+02	---	---	---	---
	Zr-93	---	---	---	---	---	---	---	---	8.83E-04	1.06E_01	---	---
	Zr-95	---	---	---	---	---	---	---	---	---	---	---	---
	Nb-93m	---	---	---	---	---	---	---	---	---	---	---	---
	Nb-94	---	---	---	---	---	---	---	---	---	---	---	---
	Nb-95	---	---	---	---	---	---	---	---	---	---	---	---
	Nb-95m	---	---	---	---	---	---	---	---	---	---	---	---
	Mo-93	---	---	---	---	---	---	---	---	---	---	---	---
	Tc-99	4.28E-02	---	---	5.29E+00	4.70E+00	8.65E+00	3.92E+00	3.62E-02	6.68E-03	8.04E-01	6.31E-01	---
	Ru-103	---	---	---	---	---	---	---	---	---	---	---	---
	Ru-106	8.92E+00	5.00E-03	---	8.02E-03	4.76E-04	2.36E+01	4.63E+01	1.04E-03	2.57E-01	1.02E+01	---	---
	Rh-106	8.92E+00	---	---	8.02E-03	4.76E-04	2.36E+01	4.63E+01	1.04E-03	---	---	---	---
	Pd-107	---	---	---	---	---	---	---	---	5.14E-05	6.18E-03	---	---
	Ag-110m	---	---	---	---	---	---	---	---	7.94E-05	9.55E-03	---	---
	In-113m	---	---	---	---	---	---	---	---	---	---	---	---
	Sn-113	---	---	---	---	---	---	---	---	---	---	---	---
	Sn-119m	---	---	---	---	---	---	---	---	---	---	---	---

Table 4.4-2 Residual Radionuclide Summary

		H AREA											
					H-Area Tanks								
		Canyon (Separations)	ETF Receipt Tank	Inactive Process Sewer Lines	Tank # 9-12	Tank #13-16	Tank # 21-24, 29-32, and 35-37	Tank # 38-43	Tank # 48-51 .	New Solvent Tanks H33-H36	Sand Filter	Seepage Basin GW Op. Unit	Tritium Processing
	Building Number	221-H	241-H	081-H	NA	NA	NA	NA	NA	DU	294-H	DU	232H,233H, 234H
	Site Map Page No.	15,F-5	16,F-8 17,G-11	Closed	16,F-12	17,A-5	14,F-6	14,H-9	14,I-9	14,D-12	15,H-10	Closed	15,C-1
	Dates of Operation	Early 50s-2005	1977-Present	1955 - 1982	DU	DU	DU	DU	DU	1997 - 2028	1975 - 1990	1954 - 1988	1955 - 2005
	Total Volume	DU	1000 L	DU	400 gal	400 gal	2000 gal	600 gal	1300 gal	DU	DU	DU	DU
	Reference No.	B-15	B-42	B-33	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-39	B-45	B-18	B-17
	Sn-121m	---	---	---	---	---	---	---	---	---	---	---	---
	Sn-123	---	---	---	---	---	---	---	---	---	---	---	---
	Sn-126	1.38E-04	---	---	4.12E-01	3.66E-01	5.16E-01	2.13E-01	2.89E-03	2.55E-04	3.07E-02	---	---
	Sb-125	---	6.00E-05	---	4.85E+00	1.61E+00	1.90E+02	1.78E+02	2.37E-01	3.27E-05	3.93E-03	---	---
	Sb-126	---	---	---	---	---	---	---	---	---	---	---	---
	Sb-126m	---	---	---	---	---	---	---	---	---	---	---	---
	Te-125m	---	6.00E-05	---	---	---	---	---	---	1.31E-01	1.58E+01	---	---
	Te-127	---	---	---	---	---	---	---	---	---	---	---	---
	Te-127m	---	---	---	---	---	---	---	---	---	---	---	---
	I-129	7.79E-05	2.70E-05	1.28E-01	2.14E-05	1.90E-05	3.11E-05	1.36E-05	1.48E-07	1.73E-04	2.08E-02	1.54E+00	---
	Cs-134	---	---	6.54E-02	3.98E-01	7.70E-02	4.21E+01	4.27E+01	1.97E-02	3.92E+00	4.71E+02	---	---
R	Cs-135	---	---	---	3.50E-03	3.11E-03	5.72E-03	2.59E-03	2.39E-05	1.43E-04	1.61E-02	---	---
A	Cs-137	2.60E+02	6.00E-03	6.97E+00	4.60E+02	8.46E+02	2.15E+03	1.12E+03	8.40E+00	4.00E+01	4.81E+03	1.51E+02	---
D	Ba-137m	2.46E+02	---	---	9.43E+02	8.00E+02	2.03E+03	1.06E+03	7.94E+00	---	---	---	---
I	Ce-144	1.06E+01	---	---	4.29E-03	2.05E-04	6.05E+01	2.45E+02	5.05E-04	7.06E-02	4.16E+00	---	---
O	Pr-144	1.06E+01	---	---	4.29E-03	2.05E-04	6.05E+01	2.45E+02	5.05E-04	---	---	---	---
N	Pr-144m	1.53E-01	---	---	---	---	---	---	---	---	---	---	---
U	Pm-147	1.22E+02	5.00E-04	---	1.30E+02	3.84E+01	5.86E+03	5.55E+03	5.83E+00	3.62E+00	4.35E+02	---	---
C	Sm-151	---	---	---	---	---	---	---	---	5.42E-01	6.52E+01	---	---
L	Eu-152	---	---	---	---	---	---	---	---	---	---	---	---
I	Eu-154	---	---	---	1.72E+02	1.10E+02	8.80E+02	5.98E+02	2.04E+00	2.12E+00	2.55E+02	---	---
D	Eu-155	---	---	---	---	---	---	---	---	7.62E-02	9.16E+00	---	---
E	Hf-181	---	---	---	---	---	---	---	---	---	---	---	---
	Ta-182	---	---	---	---	---	---	---	---	---	---	---	---
	Pb-212	---	---	---	---	---	---	---	---	---	---	---	---
	Pb-214	---	---	---	---	---	---	---	---	---	---	---	---
	Bi-214	---	---	---	---	---	---	---	---	---	---	---	---
	Ra-226	---	---	7.63E-02	---	---	---	---	---	---	---	---	---
	Ra-228	---	---	---	---	---	---	---	---	---	---	---	---
	Ac-228	---	---	---	---	---	---	---	---	---	---	---	---
	Th-228	---	---	---	---	---	---	---	---	---	---	---	---
	Th-230	---	---	---	---	---	---	---	---	---	---	---	---
	Th-231	---	---	---	---	---	---	---	---	---	---	---	---
	Th-232	---	---	7.38E-02	8.70E-04	1.12E-03	2.12E-04	2.82E-04	5.55E-09	---	---	---	---
	Th-234	---	---	---	---	---	---	---	---	---	---	---	---
	Pa-234	---	---	---	---	---	---	---	---	---	---	---	---

Table 4.4-2 Residual Radionuclide Summary

	H AREA											
				H-Area Tanks								
	Canyon (Separations)	ETF Receipt Tank	Inactive Process Sewer Lines	Tank # 9-12	Tank #13-16	Tank # 21-24, 29-32, and 35-37	Tank # 38-43	Tank # 48-51	New Solvent Tanks H33-H36	Sand Filter	Seepage Basin GW Op. Unit	Tritium Processing
Building Number	221-H	241-H	081-H	NA	NA	NA	NA	NA	DU	294-H	DU	232H,233H, 234H
Site Map Page No.	15,F-5	16,F-8 17,G-11	Closed	16,F-12	17,A-5	14,F-6	14,H-9	14,I-9	14,D-12	15,H-10	Closed	15,C-1
Dates of Operation	Early 50s-2005	1977-Present	1955 - 1982	DU	DU	DU	DU	DU	1997 - 2028	1975 - 1990	1954 - 1988	1955 - 2005
Total Volume	DU	1000 L	DU	400 gal	400 gal	2000 gal	600 gal	1300 gal	DU	DU	DU	DU
Reference No.	B-15	B-42	B-33	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-30, B-21,B-46	B-39	B-45	B-18	B-17
RADIONUCLIDE												
U-232	---	---	---	1.10E-04	9.65E-05	4.25E-05	4.55E-06	3.06E-06	---	---	---	---
U-233	7.46E-10	---	---	2.60E-02	3.01E-02	5.60E-02	4.96E-03	3.86E-04	---	---	---	---
U-234	4.44E-02	---	1.91E-01	3.80E-03	5.49E-03	2.57E-02	1.60E-02	2.45E-04	---	---	1.53E-01	---
U-235	6.42E-04	---	---	2.29E-04	1.91E-04	4.94E-04	2.64E-04	2.68E-05	---	---	1.06E-01	---
U-236	9.54E-03	---	---	4.17E-04	5.22E-04	5.43E-03	3.60E-03	4.71E-05	---	---	---	---
U-238	2.80E-05	---	1.91E-01	4.42E-03	2.51E-03	2.13E-03	1.74E-03	1.16E-03	---	---	1.35E-01	---
Np-237	3.56E-01	---	---	3.44E-02	2.04E-02	2.45E-02	9.70E-03	1.50E-04	---	---	---	---
Np-239	---	---	---	2.28E+02	5.74E+01	2.06E+03	8.15E+02	1.31E+00	---	---	---	---
Pu-236	---	---	---	---	---	---	---	---	---	---	---	---
Pu-238	1.02E+03	4.00E-05	3.27E-01	2.02E+02	5.08E+01	1.82E+03	7.22E+02	1.16E+00	5.00E+01	2.35E+01	1.16E+00	---
Pu-239	6.90E+00	1.00E-05	5.50E+00	4.30E+00	2.78E+00	1.66E+01	7.95E+00	7.14E-01	1.00E+01	---	4.06E+00	---
Pu-240	3.10E+00	---	---	1.99E+00	1.07E+00	1.19E+01	4.97E+00	1.82E-01	---	---	---	---
Pu-241	1.06E+02	---	---	3.90E+01	4.79E+00	8.37E+02	4.14E+02	1.07E+00	---	---	---	---
Pu-242	3.15E-02	---	---	3.07E-03	5.10E-04	2.59E-02	1.16E-02	2.12E-04	---	---	---	---
Pu-244	---	---	---	---	---	---	---	---	---	---	---	---
Am-241	---	---	2.07E-01	1.08E+02	4.53E+01	7.72E+02	3.40E+02	2.33E+00	---	---	3.93E-01	---
Am-242	---	---	---	---	---	---	---	---	---	---	---	---
Am-242m	---	---	---	3.83E-02	3.40E-02	5.21E-02	2.13E-02	7.84E-05	---	---	---	---
Am-243	---	---	---	---	---	---	---	---	---	---	---	---
Cm-242	---	---	---	---	---	---	---	---	---	---	---	---
Cm-243	---	---	---	---	---	---	---	---	---	---	---	---
Cm-244	---	---	2.72E-02	1.14E-01	8.50E-02	4.03E-01	2.37E-01	9.67E-04	4.00E+01	---	---	---
Cm-245	---	---	---	1.04E-05	9.19E-06	2.75E-05	1.38E-05	6.59E-08	---	---	---	---
Cm-246	---	---	---	---	---	---	---	---	---	---	---	---
Cm-247	---	---	---	---	---	---	---	---	---	---	---	---
Cm-248	---	---	---	---	---	---	---	---	---	---	---	---
Cf-249	---	---	---	---	---	---	---	---	---	---	---	---
Cf-251	---	---	---	---	---	---	---	---	---	---	---	---
Cf-252	---	---	---	---	---	---	---	---	---	---	---	---

Table 4.4-2 Residual Radionuclide Summary

	S Area		Z Area	Various Spills								
	DWPF	Low Point Pump Pit	Saltstone Vaults	Spill at Tank 13	Spill at Tank 9	Spill at Tank 16	Spill at Tank 37	Spill at B281-3F	Spill at Tank 3	Spill at Tank 8	Spill at B281-3H	Soil and Debris Consol. Facility
Building Number	292-S	511-S	451-1,6,7	Tank 13	Tank 9	Tank 16	Tank 37	B281-3F	Tank 3	Tank 8	B281-3H	TBA
Site Map Page No.	19,F-4	19,J-5	20, G-8	DU	DU	DU	DU	DU	DU	DU	DU	DU
Dates of Operation	1996-2038	1996-2038	1992 - 2038	Dec-83	May-67	Sep-60	Feb-89	Startup - 1973	Aug-75	Apr-61	Startup - 1973	TBA
Total Volume	1000 gal	50 gal	DU	100 gal	DU	DU	DU	DU	DU	DU	DU	562,500 yd^3
Reference No.	B-41	B-41	B-23	B-4, B-38	B-4, B-38	B-38	B-5, B-37	B-4	B-38	B-4, B-38	B-4	B-25, B-27, B-28, B-34
RADIONUCLIDE												
H-3	6.34E-02	3.17E-03	1.90E+04	---	---	5.00E-02	8.41E-02	---	---	---	---	---
C-14	---	---	6.50E+00	---	---	---	---	---	---	---	---	7.10E-03
Na-22	---	---	---	---	---	---	---	---	---	---	---	---
Al-26	---	---	---	---	---	---	---	---	---	---	---	---
K-40	---	---	---	---	---	---	---	---	---	---	---	2.90E-03
Sc-46	---	---	---	---	---	---	---	---	---	---	---	---
Cr-51	---	---	---	---	---	---	---	---	---	---	---	---
Mn-54	---	---	---	---	---	---	---	---	---	---	---	---
Fe-55	---	---	---	---	---	---	---	---	---	---	---	---
Fe-59	---	---	---	---	---	---	---	---	---	---	---	---
Co-57	---	---	---	---	---	---	---	---	---	---	---	---
Co-58	---	---	---	---	---	---	---	---	---	---	---	5.90E-04
Co-60	2.94E+02	1.47E+01	2.00E+02	---	---	---	---	---	---	---	---	1.20E-04
Ni-59	2.39E+00	1.20E-01	2.00E-01	---	---	---	---	---	---	---	---	---
Ni-63	2.97E+02	1.49E+01	2.00E+01	---	---	---	---	---	---	---	---	---
Zn-65	---	---	---	---	---	---	---	---	---	---	---	---
Se-79	2.34E-01	1.17E-02	3.20E+02	1.46E-03	2.57E-04	3.86E-03	6.51E-03	4.65E-05	1.73E-04	1.24E-02	1.39E-04	---
Sr-89	---	---	---	---	---	---	---	---	---	---	---	---
Sr-90	5.17E+04	2.59E+03	6.80E+02	2.23E+02	3.92E+01	2.00E+00	3.36E+00	7.09E+00	2.64E+01	1.89E+03	2.13E+01	7.23E-01
Y-90	5.32E+04	2.66E+03	---	---	---	---	---	---	---	---	---	---
Zr-93	1.94E+00	9.70E-02	2.60E-01	6.96E-03	1.22E-03	1.82E-02	3.09E-02	2.21E-04	8.21E-04	5.90E-02	6.62E-04	---
Zr-95	1.74E-02	8.70E-04	---	---	---	---	---	---	---	---	---	---
Nb-93m	---	---	---	---	---	---	---	---	---	---	---	---
Nb-94	---	---	---	---	---	---	---	---	---	---	---	---
Nb-95	3.67E-02	1.84E-03	---	---	---	---	---	---	---	---	---	---
Nb-95m	---	---	---	---	---	---	---	---	---	---	---	---
Mo-93	---	---	---	---	---	---	---	---	---	---	---	---
Tc-99	4.26E+00	2.13E-01	6.50E+04	5.26E-02	9.24E-03	1.39E-01	2.34E-01	1.67E-03	6.21E-03	4.46E-01	5.01E-03	5.93E-04
Ru-103	---	---	---	---	---	---	---	---	---	---	---	---
Ru-106	2.69E+03	1.35E+02	3.30E+04	2.02E+00	3.55E-01	5.33E+00	9.00E+00	6.43E-02	2.39E-01	1.72E+01	1.93E-01	---
Rh-106	2.64E+03	1.32E+02	---	---	---	---	---	---	---	---	---	---
Pd-107	1.57E-02	7.85E-04	2.00E-02	4.05E-04	7.11E-05	1.07E-03	1.80E-03	1.29E-05	4.78E-05	3.43E-03	3.86E-05	---
Ag-110m	2.25E+01	1.13E+00	5.80E-01	6.26E-04	1.10E-04	1.65E-03	2.78E-03	1.99E-05	7.39E-05	5.30E-03	5.96E-05	---
In-113m	---	---	---	---	---	---	---	---	---	---	---	---
Sn-113	---	---	---	---	---	---	---	---	---	---	---	---
Sn-119m	---	---	---	---	---	---	---	---	---	---	---	---

Table 4.4-2 Residual Radionuclide Summary

		S Area		Z Area	Various Spills								
		DWPF	Low Point Pump Pit	Saltstone Vaults	Spill at Tank 13	Spill at Tank 9	Spill at Tank 16	Spill at Tank 37	Spill at B281-3F	Spill at Tank 3	Spill at Tank 8	Spill at B281-3H	Soil and Debris Consol. Facility
	Building Number	292-S	511-S	451-1,6,7	Tank 13	Tank 9	Tank 16	Tank 37	B281-3F	Tank 3	Tank 8	B281-3H	TBA
	Site Map Page No.	19,F-4	19,J-5	20, G-8	DU	DU	DU	DU	DU	DU	DU	DU	DU
	Dates of Operation	1996-2038	1996-2038	1992 - 2038	Dec-83	May-67	Sep-60	Feb-89	Startup - 1973	Aug-75	Apr-61	Startup - 1973	TBA
	Total Volume	1000 gal	50 gal	DU	100 gal	DU	DU	DU	DU	DU	DU	DU	562,500 yd^3
	Reference No.	B-41	B-41	B-23	B-4, B-38	B-4, B-38	B-38	B-5, B-37	B-4	B-38	B-4, B-38	B-4	B-25, B-27, B-28, B-34
	Sn-121m	5.13E-02	2.57E-03	2.60E+01	---	---	---	---	---	---	---	---	---
	Sn-123	4.55E-01	2.28E-02	---	---	---	---	---	---	---	---	---	---
	Sn-126	2.58E-01	1.29E-02	1.30E+02	2.01E-03	3.53E-04	5.29E-03	8.93E-03	6.38E-05	2.37E-04	1.70E-02	1.91E-04	---
	Sb-125	1.43E+03	7.15E+01	6.50E+03	2.58E-04	4.52E-05	6.79E-04	1.14E-03	8.18E-06	3.04E-05	2.18E-03	2.45E-05	---
	Sb-126	---	---	1.30E+01	---	---	---	---	---	---	---	---	---
	Sb-126m	2.60E-01	1.30E-02	---	---	---	---	---	---	---	---	---	---
	Te-125m	3.42E+02	1.71E+01	2.00E+02	1.03E+00	1.82E-01	2.72E+00	4.60E+00	3.28E-02	1.22E-01	8.76E+00	9.85E-02	---
	Te-127	1.49E-01	7.45E-03	---	---	---	---	---	---	---	---	---	---
	Te-127m	1.53E-01	7.65E-03	---	---	---	---	---	---	---	---	---	---
	I-129	1.24E-02	6.20E-04	2.00E+01	1.37E-03	2.40E-04	3.60E-03	6.07E-03	4.33E-05	1.61E-04	1.16E-02	1.30E-04	1.90E-02
	Cs-134	3.03E+02	1.52E+01	6.50E+01	3.08E+01	5.41E+00	8.13E+01	1.37E+02	9.79E-01	3.64E+00	2.61E+02	2.94E+00	---
R	Cs-135	---	---	3.90E-02	1.05E-03	1.85E-04	2.77E-03	4.68E-03	3.34E-05	1.24E-04	8.92E-03	1.00E-04	---
A	Cs-137	2.86E+03	1.43E+02	2.65E+04	3.15E+02	5.53E+01	8.30E+02	1.40E+03	1.00E+01	3.72E+01	2.67E+03	3.00E+01	3.00E+00
D	Ba-137m	2.70E+03	1.35E+02	---	---	---	---	---	---	---	---	---	---
I	Ce-144	1.69E+04	8.45E+02	3.20E+00	5.56E-01	9.76E-02	1.46E+00	2.47E+00	1.76E-02	6.56E-02	4.71E+00	5.29E-02	---
O	Pr-144	1.69E+04	8.45E+02	---	---	---	---	---	---	---	---	---	---
N	Pr-144m	2.04E+02	1.02E+01	---	---	---	---	---	---	---	---	---	---
U	Pm-147	4.15E+04	2.08E+03	3.90E+03	2.85E+01	5.01E+00	7.51E+01	1.27E+02	9.05E-01	3.37E+00	2.42E+02	2.72E+00	---
C	Sm-151	4.19E+02	2.10E+01	2.00E+03	4.27E+00	7.49E-01	1.12E+01	1.90E+01	1.36E-01	5.04E-01	3.62E+01	4.07E-01	---
L	Eu-152	6.37E+00	3.19E-01	5.80E+00	---	---	---	---	---	---	---	---	---
I	Eu-154	1.07E+03	5.35E+01	6.50E+02	1.67E+01	2.93E+00	4.39E+01	7.41E+01	5.29E-01	1.97E+00	1.41E+02	1.59E+00	7.10E-04
D	Eu-155	8.21E+02	4.11E+01	3.20E+02	6.00E-01	1.05E-01	1.58E+00	2.67E+00	1.90E-02	7.08E-02	5.08E+00	5.71E-02	1.50E-03
E	Hf-181	---	---	---	---	---	---	---	---	---	---	---	---
	Ta-182	---	---	---	---	---	---	---	---	---	---	---	---
	Pb-212	---	---	---	---	---	---	---	---	---	---	---	---
	Pb-214	---	---	---	---	---	---	---	---	---	---	---	---
	Bi-214	---	---	---	---	---	---	---	---	---	---	---	---
	Ra-226	---	---	---	---	---	---	---	---	---	---	---	8.20E-04
	Ra-228	---	---	---	---	---	---	---	---	---	---	---	---
	Ac-228	---	---	---	---	---	---	---	---	---	---	---	2.40E-03
	Th-228	---	---	1.30E-03	---	---	---	---	---	---	---	---	2.00E-03
	Th-230	---	---	---	---	---	---	---	---	---	---	---	1.20E-03
	Th-231	---	---	1.30E-01	---	---	---	---	---	---	---	---	---
	Th-232	---	---	---	---	---	---	---	---	---	---	---	---
	Th-234	---	---	2.00E-03	---	---	---	---	---	---	---	---	---
	Pa-234	---	---	3.90E-03	---	---	---	---	---	---	---	---	---

Table 4.4-2 Residual Radionuclide Summary

	S Area		Z Area	Various Spills								
	DWPF	Low Point Pump Pit	Saltstone Vaults	Spill at Tank 13	Spill at Tank 9	Spill at Tank 16	Spill at Tank 37	Spill at B281-3F	Spill at Tank 3	Spill at Tank 8	Spill at B281-3H	Soil and Debris Consol. Facility
Building Number	292-S	511-S	451-1,6,7	Tank 13	Tank 9	Tank 16	Tank 37	B281-3F	Tank 3	Tank 8	B281-3H	TBA
Site Map Page No.	19,F-4	19,J-5	20, G-8	DU	DU	DU	DU	DU	DU	DU	DU	DU
Dates of Operation	1996-2038	1996-2038	1992 - 2038	Dec-83	May-67	Sep-60	Feb-89	Startup - 1973	Aug-75	Apr-61	Startup - 1973	TBA
Total Volume	1000 gal	50 gal	DU	100 gal	DU	DU	DU	DU	DU	DU	DU	562,500 yd^3
Reference No.	B-41	B-41	B-23	B-4, B-38	B-4, B-38	B-38	B-5, B-37	B-4	B-38	B-4, B-38	B-4	B-25, B-27, B-28, B-34
RADIONUCLIDE												
U-232	1.46E-01	7.30E-03	4.50E-02	---	---	---	---	---	---	---	---	---
U-233	---	---	2.60E-03	---	---	---	---	---	---	---	---	---
U-234	4.60E-01	2.30E-02	2.60E-01	---	---	---	---	---	---	---	---	8.54E-02
U-235	---	---	---	---	---	---	---	---	---	---	---	8.59E-03
U-236	3.34E-02	1.67E-03	---	---	---	---	---	---	---	---	---	6.29E-06
U-238	---	---	2.00E-03	---	---	---	---	---	---	---	---	8.54E-02
Np-237	1.52E-02	7.60E-04	5.80E-02	---	---	---	---	---	---	---	---	4.25E-07
Np-239	---	---	---	---	---	---	---	---	---	---	---	5.90E-05
Pu-236	1.06E-01	5.30E-03	---	---	---	---	---	---	---	---	---	---
Pu-238	1.29E+03	6.45E+01	4.90E+01	---	---	---	---	---	---	---	---	2.07E-02
Pu-239	1.21E+01	6.05E-01	1.31E+02	---	---	2.00E-01	3.36E-01	---	---	---	---	1.25E-01
Pu-240	7.70E+00	3.85E-01	3.20E-01	---	---	---	---	---	---	---	---	4.67E-05
Pu-241	1.45E+03	7.25E+01	3.20E+01	---	---	---	---	---	---	---	---	2.10E-03
Pu-242	1.06E-02	5.30E-04	---	---	---	---	---	---	---	---	---	---
Pu-244	---	---	---	---	---	---	---	---	---	---	---	---
Am-241	1.86E+01	9.30E-01	1.30E+02	---	---	---	---	---	---	---	---	3.12E-02
Am-242	2.45E-02	1.23E-03	6.50E-02	---	---	---	---	---	---	---	---	---
Am-242m	2.47E-02	1.24E-03	6.50E-02	---	---	---	---	---	---	---	---	---
Am-243	---	---	3.90E-02	---	---	---	---	---	---	---	---	---
Cm-242	6.03E-02	3.02E-03	6.50E-02	---	---	---	---	---	---	---	---	---
Cm-243	---	---	2.60E-02	---	---	---	---	---	---	---	---	---
Cm-244	2.80E-01	1.40E-02	6.50E-01	---	---	---	---	---	---	---	---	---
Cm-245	---	---	---	---	---	---	---	---	---	---	---	---
Cm-246	---	---	---	---	---	---	---	---	---	---	---	---
Cm-247	---	---	---	---	---	---	---	---	---	---	---	---
Cm-248	---	---	---	---	---	---	---	---	---	---	---	---
Cf-249	---	---	---	---	---	---	---	---	---	---	---	---
Cf-251	---	---	---	---	---	---	---	---	---	---	---	---
Cf-252	---	---	---	---	---	---	---	---	---	---	---	---

Table 4.4-5 Results of Flux to the Water Table Calculations up to 1,000 Years

		F-Canyon Ci/yr	FSAND Ci/yr	HLT1-8 Ci/yr	HLT17-20 Ci/yr	HLT25-28 Ci/yr	HLT33-34 Ci/yr	235-F Ci/yr	772-F Ci/yr	H Canyon Ci/yr
RADIONUCLIDE	H-3	9.20E+00	2.21E+00						1.57E+00	2.95E-01
	C-14	6.07E-06		<10E-18	<10E-18	2.98E-07				9.25E-08
	Ni-59	<10E-18		<10E-18	<10E-18	<10E-18	<10E-18			<10E-18
	Se-79	2.38E-03	6.93E-04	<10E-18	<10E-18	<10E-18	<10E-18		1.09E-07	3.59E-05
	Sr-90	<1E-18	6.73E-10	<1E-18	<1E-18	<1E-18	<1E-18		<1E-18	6.83E-06
	Zr-93		<1E-18							
	Tc-99	3.74E-01	6.44E-02	3.28E-01	1.70E-02	7.36E-02	1.43E-01		1.65E-05	5.62E-03
	Sn-126	<1.E-18	<1E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18		<1.E-18	<1.E-18
	I-129	3.69E-04	9.37E-04	<1.E-18	5.94E-08	2.53E-07	5.12E-07		2.33E-07	
	Cm-246	<1.E-18								
	Cf-252									
	Ra-226	5.10E-12		<1.E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18	5.33E-15
	Th-228									
	Th-230	1.54E-16		<1.E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18
	Th-232	<1.E-18		<1.E-18	<1.E-18	<1.E-18	<1.E-18		<1.E-18	<1.E-18
	U-233	3.82E-12								5.73E-14
	U-234	3.80E-06		<1.E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18		3.59E-06
	U-235	6.63E-08	<1.E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18			4.94E-08
	U-236	7.21E-07		<1.E-18	<1.E-18	<1.E-18	<1.E-18		<1.E-18	<1.E-18
	U-238	1.29E-06		<1.E-18	<1.E-18	<1.E-18	<1.E-18		<1.E-18	2.15E-09
	Np-237	1.09E-07		<1.E-18	<1.E-18	<1.E-18	<1.E-18	3.69E-04	<1.E-18	1.10E-05
	Pu-238	<1.E-18						<1.E-18	<1.E-18	<1.E-18
	Pu-239	<1.E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18		<1.E-18	<1.E-18
	Pu-240	<1.E-18		<1.E-18	<1.E-18	<1.E-18	<1.E-18		<1.E-18	<1.E-18
	Pu-241	<1.E-18		<1.E-18	<1.E-18	<1.E-18	<1.E-18		<1.E-18	<1.E-18
	Pu-242	<1.E-18		<1.E-18	<1.E-18	<1.E-18	<1.E-18			<1.E-18
	Am-241	<1.E-18		<1.E-18	<1.E-18	<1.E-18	<1.E-18		<1.E-18	
	Am-243	<1.E-18								
	Cm-244	<1.E-18		<1.E-18	<1.E-18					

Table 4.4-5 Results of Flux to the Water Table Calculations up to 1,000 Years

		HSAND Ci/yr	HLT9-12 Ci/yr	HLT13-16 Ci/yr	HLT21-24 Ci/yr	HLT38-43 Ci/yr	HLT48-51 Ci/yr	ETFTANKS Ci/yr	TRIT Ci/yr	FSEEP Ci/yr
	H-3							2.52E-02	6.30E+03	
	C-14		1.95E-07	7.07E-08	1.16E-07	1.36E-07	3.69E-09			
	Ni-59		8.36E-06	7.02E-06	<10E-18	<10E-18	<10E-18			
	Se-79	6.98E-04	9.73E-04	8.68E-04	1.84E-03	<1E-18	<1E-18			
	Sr-90	1.58E-04	1.22E-02	1.03E-02	<1.E-18	<1.E-18	<1.E-18	3.84E-10		
	Zr-93	<1E-18								
	Tc-99	6.57E-02	1.65E-01	1.46E-01	2.58E-01	1.06E-01	9.49E-04			6.05E-05
	Sn-126	<1E-18	4.70E-05	4.18E-05	<1.E-18	<1.E-18	<1.E-18			
R	I-129	9.52E-04	4.08E-07	3.68E-07	6.73E-07	2.59E-07	2.83E-09	5.06E-06		6.80E-04
A	Cm-246									
D	Cf-252									
I	Ra-226		1.51E-10	2.34E-10	2.74E-18	<1.E-18	<1.E-18	2.73E-13		4.66E-09
O	Th-228									
N	Th-230		8.16E-10	1.27E-09	4.03E-17	<1.E-18	<1.E-18	1.87E-12		1.88E-08
U	Th-232		1.04E-15	1.62E-15	<1.E-18	<1.E-18	<1.E-18			
C	U-233		2.36E-05	2.73E-05	2.79E-05	<1.E-18	<1.E-18			
L	U-234		3.65E-06	5.48E-06	1.35E-05	<1.E-18	<1.E-18			
I	U-235	<1.E-18	2.08E-07	1.74E-07	2.47E-07	<1.E-18	<1.E-18			
D	U-236		3.90E-04	2.96E-04	2.71E-06	<1.E-18	<1.E-18			
E	U-238		4.04E-06	2.39E-06	1.06E-06	<1.E-18	<1.E-18			
	Np-237		7.89E-03	2.62E-02	2.28E-06	<1.E-18	<1.E-18			
	Pu-238		1.25E-08	1.94E-08	<1.E-18	<1.E-18	<1.E-18			
	Pu-239	6.88E-07	1.94E-08	3.01E-08	<1.E-18	<1.E-18	<1.E-18			
	Pu-240		4.18E-05	3.18E-05	<1.E-18	<1.E-18	<1.E-18			
	Pu-241		8.60E-17	1.34E-16	<1.E-18	<1.E-18	<1.E-18			
	Pu-242		<1.E-18	1.98E-09	<1.E-18	<1.E-18	<1.E-18			
	Am-241		5.97E-18	7.70E-18	<1.E-18	<1.E-18	<1.E-18			
	Am-243									
	Cm-244		<1.E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18			

Table 4.4-5 Results of Flux to the Water Table Calculations up to 1,000 Years

		HSEEP	ST33-36	OBG Ci/yr	LYSIM Ci/yr	SSLYSIM Ci/yr	MWMF Ci/yr	ST1-22 Ci/yr	ST23-32 Ci/yr	247-F Ci/yr
RADIONUCLIDE	H-3			3.60E+04		1.28E-01	6.25E+04			
	C-14			1.12E+00	6.18E-02	4.43E-09	1.35E+00			
	Ni-59			<10E-18		<10E-18	<10E-18			
	Se-79		5.80E-06	6.98E-03		3.89E-04	1.10E-03	1.60E-06	6.52E-07	
	Sr-90		2.91E-08	6.28E-04	4.95E-11	8.94E-12	7.90E-04	2.46E-05	4.67E-07	
	Zr-93		<1E-18			<1.E-18		<1E-18	1.95E-04	
	Tc-99	4.34E-04	6.38E-04	3.66E-02		2.62E-01	5.75E-03	4.75E-04	1.95E-04	
	Sn-126		<1E-18	<1.E-18		<1.E-18	<1.E-18	<1E-18	<1.0E-18	
	I-129	2.93E-02	3.26E-05	3.93E-02		4.35E-05	6.21E-03	8.96E-06	3.67E-06	
	Cm-246									
	Cf-252			<1.E-18			<1.E-18			
	Ra-226	3.01E-09	7.60E-08	1.42E-07	<1.E-18	1.57E-06	3.50E-07			
	Th-228		9.42E-10			<1.E-18				
	Th-230	1.80E-08	5.54E-07	5.30E-12	<1.E-18	<1.E-18	7.21E-11			
	Th-232			<1.E-18		<1.E-18	<1.E-18			
	U-233			3.16E-03		6.38E-12	2.77E-02			
	U-234			<1.E-18	<1.E-18	<1.E-18	<1.E-18			3.61E-04
	U-235			8.33E-03	<1.E-18	<1.E-18	<1.E-18			2.76E-06
	U-236		7.90E-09	3.87E-02		<1.E-18	7.98E-02			4.42E-09
	U-238			2.13E-01		4.81E-12	6.27E-01			
	Np-237			1.52E-02	<1.E-18	5.69E-11	9.31E-04			
	Pu-238		4.17E-08	<1.E-18	<1.E-18	<1.E-18	<1.E-18	1.04E-08	1.52E-08	
	Pu-239		1.34E-08	<1.E-18	<1.E-18	<1.E-18	<1.E-18	9.14E-09	1.18E-08	
	Pu-240			<1.E-18		<1.E-18	<1.E-18			
	Pu-241			<1.E-18		<1.E-18	<1.E-18			
	Pu-242						<1.E-18			
	Am-241			<1.E-18	<1.E-18	<1.E-18	<1.E-18			
	Am-243					<1.E-18	<1.E-18			
	Cm-244			<1.E-18		<1.E-18	<1.E-18			

Table 4.4-5 Results of Flux to the Water Table Calculations up to 1,000 Years

		247-F	DWPF Ci/yr	LOWPT Ci/yr	ILV Ci/yr	LAW Ci/yr	KAPL Ci/yr	SALT Ci/yr	SLIT Ci/yr
RADIONUCLIDE	H-3		1.70E-02	1.08E-03	8.54E-08	9.79E-05	<1.E-18	3.80E-08	8.29E-01
	C-14				8.38E-12	4.03E-06	3.60E-08	<1.E-18	
	Ni-59		<10E-18	<10E-18	2.83E-06	2.32E-05	9.16E-03	NA	
	Se-79		7.30E-03	3.65E-04	<1.E-18	<1.E-18	NA	<1.E-18	
	Sr-90		9.49E-04	3.85E-03	1.89E-12	<1.E-18	<1.E-18	<1.E-18	2.31E-05
	Zr-93		<1.E-18	<1.E-18	NA	NA	9.38E-06	NA	
	Tc-99		5.59E-01	3.93E-02	1.73E-04	6.10E-05	<1.E-18	<1.E-18	3.14E-05
	Sn-126		<1.E-18	<1.E-18	<1.E-18	<1.E-18	NA	<1.E-18	
	I-129		8.79E-04	6.75E-05	2.71E-07	1.90E-07	<1.E-18	<1.E-18	2.93E-02
	Cm-246				NA	<1.E-18	NA	NA	
	Cf-252				<1.E-18	<1.E-18	NA	NA	
	Ra-226		<1.E-18	<1.E-18	NA	NA	NA	NA	
	Th-228				NA	NA	NA	NA	
	Th-230		<1.E-18	<1.E-18	NA	NA	NA	NA	
	Th-232		<1.E-18	<1.E-18	1.06E-05	3.00E-05	NA	NA	
	U-233				<1.E-18	<1.E-18	NA	NA	
	U-234	3.61E-04	3.72E-05	3.02E-06	<1.E-18	<1.E-18	NA	NA	
	U-235	2.76E-06	7.69E-05	<1.E-18	<1.E-18	<1.E-18	NA	NA	
	U-236	4.42E-09	2.57E-06	2.09E-07	<1.E-18	<1.E-18	NA	NA	
	U-238				<1.E-18	<1.E-18	NA	NA	
	Np-237		4.68E-07	3.80E-08	<1.E-18	<1.E-18	NA	NA	3.15E-09
	Pu-238		<1.E-18	<1.E-18	<1.E-18	<1.E-18	NA	<1.E-18	<1.E-18
	Pu-239		<1.E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18	NA	
	Pu-240		<1.E-18	<1.E-18	<1.E-18	<1.E-18	NA	NA	
	Pu-241		<1.E-18	<1.E-18	<1.E-18	<1.E-18	<1.E-18	NA	
	Pu-242		<1.E-18	<1.E-18	<1.E-18	<1.E-18	NA	NA	
	Am-241		<1.E-18	<1.E-18	<1.E-18	<1.E-18	NA	<1.E-18	<1.E-18
	Am-243				NA	<1.E-18	NA	NA	
	Cm-244		<1.E-18	<1.E-18	NA	NA	NA	NA	

Table 4.4-5 Results of Flux to the Water Table Calculations up to 1,000 Years

		Tank 13 Spill	Tank 9 Spill	Tank 16 Spill	Tank 37 Spill	B281-F Spill	Tank 3 Spill	Tank 8 Spill	B281-H Spill
RADIONUCLIDE	H-3			2.36E-02	3.82E-02				
	C-14								
	Ni-59								
	Se-79	4.35E-05	7.65E-06	1.15E-04	1.94E-04	1.38E-06	4.87E-06	3.49E-04	4.13E-06
	Sr-90	8.84E-01	1.55E-01	7.93E-03	9.19E-04	1.41E-06	6.71E-16	4.80E-14	4.23E-06
	Zr-93	8.20E-07	3.19E-07	7.40E-06	8.05E-06	<1E-18	<1E-18	<1E-18	<1E-18
	Tc-99	3.62E-04	6.37E-05	9.58E-04	1.61E-03	1.15E-05	3.42E-05	2.46E-03	3.45E-05
	Sn-126	2.41E-06	4.23E-07	6.35E-06	<1E-18	<1E-18	<1E-18	<1E-18	<1E-18
	I-129	2.43E-04	4.26E-05	6.40E-04	1.09E-03	7.77E-06	2.61E-05	1.88E-03	2.33E-05
	Cm-246								
	Cf-252								
	Ra-226								
	Th-228								
	Th-230								
	Th-232								
	U-233								
	U-234								
	U-235			<1E-18	<1E-18				
	U-236								
	U-238								
	Np-237								
	Pu-238								
	Pu-239			5.72E-08	5.72E-08				
	Pu-240								
	Pu-241								
	Pu-242								
	Am-241								
	Am-243								
	Cm-244								

3.3 References

J. B. Pickett, W. P. Colven and H. W. Bledsoe, Environmental Information Document, M-Area Settling Basin and Vicinity, DPST-85-703, March 1987.

Stewart, Donald C., 1985. Data for Radioactive Waste Management and Nuclear Applications. John Wiley & Sons, New York, New York. 1985.

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4.0 Condition 4

Include in the addendum to the composite analysis the assumptions and justification for the assumptions used in the analysis.

Table 4-1 lists the assumptions in the Savannah River Site Composite Analysis (CA) with their associated justification. Where the justification for the assumption is stated in the CA, or where the justification is self-evident, the assumption was omitted from the list.

Table 4-1 Assumptions and Justifications for the Savannah River Composite Analysis E-Area Vaults and Saltstone Facility

Location in CA Document	Assumption	Justification
Section 1.0 Summary and Conclusions		
Page 1-1	…prepare a CA that evaluates the impact to a hypothetical future member of the public from all radioactive sources that potentially interact with LLW disposal facilities. Therefore, the CA considered interaction of radionuclide sources in the GSA with the active E and Z Area disposal facilities.	The intent of the USDOE requirement for a CA is to consider the potential impact of other sources on the operations of a LLW disposal facility. SRS chose to restrict the CA to those sources within the GSA because it is those sources that would influence operations at the LLW disposal facilities. Radionuclides from other sources at the SRS, such as the reactor areas, will eventually migrate through groundwater to surface streams and will ultimately mix with contaminants from the GSA in the Savannah River. However, by the time the contaminants have mixed in the River, dilution will be so great that the calculated impact will be small.
	Therefore, the mouth of Upper Three Runs is the appropriate point to assess the effect of sources that potentially interact with E and Z Areas.	The mouth of UTR is the closest point to the GSA that a hypothetical future member of the public could reasonably be expected to be exposed to radionuclides from the GSA, given the current SRS land use plan.
	The CA included for completeness the assessment of the mouth of Four Mile Branch and the Savannah River at the Highway 301 bridge.	The GSA includes facilities that drain to FMB as well as UTR. Even though a local groundwater divide effectively prohibits those sources which drain to FMB from influencing UTR, it was felt to be worthwhile to include the analysis of the FMB watershed for completeness since the FMB watershed includes past waste disposal facilities.
Section 2.1 Introduction, Purpose and Scope		
Page 2-2/3	Z-Area and E-Area LLW disposal facilities and other sources of radioactive material in the vicinity of these facilities. Total projected dose from all sources will be compared with the USDOE primary dose limit of 100 mrem per year. The ALARA concept will also be explored in terms of estimated maximum individual doses, collective doses, and alternative controls. For example, if projected maximum individual dose is in excess of 30 mrem per year, an options analysis to identify alternatives that would reduce future doses would be explored.	The USDOE guidance for conducting the CA requires inclusion of the active LLW disposal facilities (i.e., Z-Area and E-Area) and other sources of radioactive material. The guidance further requires that the total projected dose from all of the sources determined to interact be compared with the USDOE primary dose limit of 100 mrem in a year. The guidance also requires consideration of the ALARA concept. It also requires development of an options analysis if the projected maximum individual dose exceeds 30 mrem in a year.

Section 2.2 Description of the GSA		
Page 2-3	The GSA contains major processing and waste management areas that will contain residual radioactivity after USDOE operations at SRS cease. The areas are E Area, F Area, H Area, S Area, and Z Area.	It was assumed that several of the facilities within the GSA (e.g., former LLW burial grounds, seepage basins, HLW storage tanks) would not be "clean-closed" (i.e., all radioactivity removed prior to closure). Thus, it was assumed that some residual radioactive material would remain in the GSA when all operations and clean up activities had been completed.
Section 2.3.5.2 Floridan Aquifer System		
Page 2-23	Because of relative hydrologic isolation due to the Meyers Branch confining system, only the Floridan aquifer system is of interest in the Composite Analysis of potential groundwater contamination from operations at the GSA. The Floridan aquifer system is comprised of the lowermost Gordon aquifer unit, the Gordon confining unit, and the uppermost Upper Three Runs aquifer, which contains the water table.	Within the GSA, the Meyers Branch confining system separates the Floridan aquifer system from the underlying aquifer systems. Because of higher hydraulic head in the lower aquifer systems than in the Floridan system, water tends to migrate upward from the lower aquifer systems into the Floridan system. Thus, sources of radioactive material within the GSA cannot contaminate the lower aquifer systems.
Section 2.4.1 Points of Assessment		
Page 2-34	Two media could be contaminated by radionuclides contained in facilities located in the GSA: groundwater and surface water which is recharged by groundwater. Contamination of the ground surface is not expected, and thus air and soil are not routes of potential contaminant transport.	The PAs for the E-Area and Z-Area LLW disposal facilities showed very little potential for migration of non-volatile radionuclides from the disposed waste to the ground surface. Thus, the only potential for migration is via groundwater. Due to the local hydrogeology, the two streams, FMB and UTR capture groundwater within the GSA, thus, groundwater and surface water are two media that could be contaminated by radionuclides contained in facilities located in the GSA. Although the PA for E-Area evaluated the migration of volatile tritium from disposed waste, it was judged that such migration would not contribute significantly in the CA because of dilution resulting from the transport to the much more distant point of assessment.
Page 2-34/5	Land-use planning for the SRS (Appendix A) indicates that release of the site to the public for unrestricted use will not occur over the time period of this analysis; therefore, on-site use by the public of potentially-contaminated groundwater is not a reasonable expectation.	The SRS Land Use Plan foresees no return of any portion of the SRS to unrestricted use by the public. The Plan foresees only heavy industrial use for the GSA. Therefore, the future public will have no access to groundwater within the GSA.
Page 2-35	Contaminated surface water is considered a potential source of exposure to a hypothetical future member of the public in this analysis.	Due to the local hydrogeology, the two streams, FMB and UTR capture groundwater within the GSA. Both FMB and UTR drain to the Savannah River which borders the SRS. Thus, the future public could be exposed to contaminated surface water.

	While land-use plans are expected to restrict use of the SRS during the time period of the analysis, the confluence of on-site streams with the Savannah River poses a potential means of public access to contaminated environmental media. Thus, the points of assessment for this analysis are the mouths of UTR and FMB and the Savannah River.	The SRS Land Use Plan foresees no return of any portion of the SRS to unrestricted use by the public. However, the Land Use Plan does not include restricted access to the Savannah River adjacent to SRS. Thus, the future public will have access to the mouths of UTR and FMB (the confluence of the streams and the river) and these points are logical points of assessment.
	Thus, the mouths of UTR and FMB, at the furthest downstream point where stream water remains undiluted with Savannah River water, are points for the assessment of potential dose to a hypothetical future member of the public.	To provide an appropriate degree of conservatism in the analysis, it was assumed that the public would have access to water in UTR and FMB at the mouths of the streams but before dilution of the stream water with water from the Savannah River.
Page 2-35	Additionally, the Savannah River will continue to be a point of public access.	Since the Savannah River is now accessible to the public and the SRS Land Use Plan does not include restrictions on access to the River, the Savannah River is logically a point of public access.
	…this composite analysis evaluates the dose to a hypothetical future member of the public at the highway 301 bridge, 20 km downstream of the SRS.	Dose was evaluated due to exposure to Savannah River water at the highway 301 bridge for convenience of comparison with data from the SRS Environmental Monitoring Program.
	Concentrations of radioactive material at the mouths of UTR and FMB will potentially include contributions from sources outside the GSA. At the highway-301 bridge, all sources of residual radioactive material on the SRS could potentially contribute to calculated dose. The composite analysis, however, has only considered the sources within the GSA because it is those sources that could influence decisions regarding operations of the LLW disposal facilities.	Several sources of radioactive material outside the GSA could contribute to contamination of UTR (e.g., M-Area seepage basin, SRL seepage basins) and to FMB (e.g., C-Reactor). Eventually, at the highway-301 bridge, several miles downstream of the SRS, all sources of residual radioactive material on the entire SRS would contribute to the potential dose to a hypothetical future member of the public. Nonetheless, SRS decided, for this first iteration of the CA, to only consider sources within the GSA. This decision was made for several reasons. First, it was judged that the sources outside the GSA would make a relatively small contribution to the total dose. Second, if a source outside the GSA contributed a significant amount to the total dose and the total dose warranted some action, the action would not involve operations of the LLW disposal facilities. Rather, the action would involve remediation of the contributing source. Third, USDOE guidance is that the CA is an interim requirement focussed on the active LLW disposal facilities. USDOE is developing a comprehensive environmental management systems approach which will consider all potential sources of residual radioactive material on a site. Thus, this first iteration of the CA need not include all sources of residual radioactive material on the SRS.

Page 2-36	Two other locations were selected to assess the sensitivity of the composite analysis to future land use decisions. These locations are on Upper Three Runs and Four Mile Branch, just downstream of the recharge points from groundwater passing under the GSA. These locations were selected because they represent points at which maximum surface water concentrations are expected to occur.	USDOE guidance for the CA indicates that sensitivity analysis should be focussed on land use. Alteration of the SRS Land Use Plan to permit public access to UTR and FMB on the current SRS reservation was considered credible but unlikely. However, it was not considered credible that the Plan would be altered to allow public access within the GSA. Thus, in the sensitivity analysis, the public was assumed to have access to the streams up to the edge of the GSA, but not within the GSA.
Page 2-37	For the assessment of potential collective dose to future populations, the population within an 80-km radius of the center of the SRS is assumed to participate in recreational activities at the highway 301-bridge location on the Savannah River. Two additional locations on the Savannah River are also used: 1) 160 km downstream of the SRS at the Beaufort-Jasper, SC water treatment plant; and 2) 160 km downstream of the SRS at the Port Wentworth, GA water treatment plant. These locations were selected because they represent present populations considered in the SRS annual environmental monitoring public report (WSRC, 1996c).	The SRS annual environmental report assesses the potential dose to the current population within 80-km of the SRS, which is consistent with NRC Regulatory Guide 1.109. The report also assesses potential dose to downstream river water users. It was decided that the CA should consider the same populations.
Section 2.4.2 Time of Assessment		
Page 2-37	…the Composite Analysis for the SRS GSA considers maximum doses that may potentially be received by a hypothetical future member of the public within a time period of at least 1,000 years. For long-lived and strongly-sorbing radionuclides, the actual peak dose may occur at times beyond 1,000 years due to slow transit times in soil and groundwater. For these radionuclides, a dose at 1,000 years is estimated, along with a peak dose and the time of occurrence of the peak dose.	USDOE guidance for the CA requires that doses within 1,000 years following closure of the LLW disposal facilities be considered. The SRS CA calculated doses over this 1,000-year period. Additionally, for completeness, the CA presented the calculated maximum dose, and the time of the maximum, for doses occurring beyond the 1,000-year period.
Section 3.2.4 Data Quality Objectives, DQO Development, Step 4: Define the Study Boundaries		
Page 3-6	Due to the projected Composite Analysis completion date of September 1997, no data provided after first quarter of 1997 were used in this Composite Analysis.	To allow completion of the CA on the schedule that had been committed to, it was necessary to establish a time-frame after which no further data would be included. The first quarter of 1997 was selected.
	There is no way to statistically validate the historical records; rather, many different sources of data were exploited to limit uncertainty.	Since it was judged to be impossible to develop statistical validation of the historical data records, it was decided to use as many different sources of data to limit the uncertainty.

	The scope of the Composite Analysis is confined to residual radionuclide inventories and releases. Releases that contain no radioactive contaminants were not considered.	USDOE guidance on the CA restricts the analysis to radiological constituents only.
Section 3.2.7 Step 7: Optimize the Design		
Page 3-7	After consideration of these two alternatives, a program of collecting historical residual radionuclide data for the GSA was identified as the most effective and timely method for compiling the initial inventory for the Composite Analysis.	The cost and lengthy time that would be required to characterize existing contamination by collecting samples and analyzing them resulted in the selection of historical data to develop the necessary source characterization.
Section 4.1.1 Source Term Development, Potential Sources of Radioactive Material, E-Area		
Page 4-4	For these tanks a total of 550 Ci of alpha emitters and 11 Ci of beta-gamma emitters are estimated to be present, based on an assumed inventory of 25 Ci of alpha emitters and 0.5 Ci of beta-gamma emitters in each tank. The alpha activity is assumed to be 40 percent ^{244}Cm, 50 percent ^{238}Pu, and 10 percent ^{239}Pu. It is also assumed that there are 0.5 Ci of beta-gamma emitters in each tank for a total of 11 Ci. The beta-gamma activity is assumed to be ^{137}Cs (Cole 1996a).	Since there are 22 tanks, the total inventory is 22 times the estimated average inventory. The assumed distribution of alpha emitters is based on spectroscopic analysis of tank residues. The review team challenged the assumption that all of the beta-gamma activity is ^{137}Cs, which is based on the solvent tank remediation team's analyses. The inventory has been reassessed, based on fission-product distributions, to estimate the inventory of a number of other radionuclides.
Page 4-5/6	For the purposes of this radionuclide inventory estimate a total of 225 Ci of alpha emitters and 4.5 Ci of beta-gamma emitters are estimated to be in these nine tanks, based on an assumed residual activity of 25 Ci of alpha emitters and 0.5 Ci of beta-gamma emitters in each tank. The alpha activity is assumed to be 40 percent ^{244}Cm, 50 percent ^{238}Pu, and 10 percent ^{239}Pu. The beta-gamma activity is assumed to be ^{137}Cs (Shappell 1996).	See above.

Section 4.1.2 F and H Areas		
Page 4-8/9	The F- and H-Area Sand Filters are part of the off-gas system for the F- and H-Area separations facilities. The sand filters are contaminated with radionuclides; therefore, they may contribute to the Composite Analysis. For the purposes of this study, the two old sand filters were assumed to have operated from 1960 through 1990 and the two new sand filters operated from 1975 through 1990. Measurements show that during canyon operations each of the filters accumulate a total of 2000 Ci/year of beta-gamma activity and 0.5 Ci/year of alpha activity. The beta-gamma activity is assumed to be composed of 32.8 percent ^{106}Ru, 12.6 percent ^{137}Cs, and 54.6 percent ^{144}Ce (Sykes and Harper 1968). The alpha activity is assumed to be composed of ^{239}Pu in the F-Area Sand Filter and ^{238}Pu in the H-Area Sand Filter.	The assumed period of operation was conservatively assigned, based on operating history, to fully encompass, and slightly exceed, the actual period of operation. The distribution of fission products in the sand filters is based on analysis of the air stream being filtered. The alpha activity distribution is based on the operational history of the two facilities. In response to Condition 3, the fission product distribution was reassessed to include longer-lived species such as ^{99}Tc.
Page 4-9	Since ^{65}Zn has a half-life of less than one year, it will not be a significant contributor to the residual radionuclide inventory estimate for the tritium production facilities.	Zinc-65 has a half-life of 244 days. Even if zinc migrated through the subsurface environment at the same rate as tritium, it would go through several tens of half lives before migrating to UTR. Thus, it would have essentially decayed away.
	For the purposes of this residual radionuclide inventory estimate, the amount of residual radionuclides remaining after D&D is assumed to be 10,000 Ci of tritium for each of the three tritium production buildings (Hsu 1996).	The estimated residual tritium is based on the Process Waste Assessment prepared for the facility and the assumption that quantities exceeding a gram of tritium would be recovered due to the value of the tritium.
Page 4-10	For the purposes of this residual radionuclide inventory, the majority of the tanks are assumed to have 378 L (100 gal) of sludge remaining after cleaning; a few of the tanks are assumed to have as much as 7570 L (2000 gal) of sludge remaining prior to filling with grout (d'Entremont 1997; Hester 1996a; Hester 1996b). Ancillary equipment such as piping and pumps will add 20 percent to the residual radionuclide total for the tanks. The density of the sludge is expected to be about 0.234 kg/L (1.95 lb/gal).	The estimated residual waste is based on operational history and construction details of each tank, and the experience gained in waste removal operations to date. The additional inventory provided by the ancillary equipment is based on operational history at the tank farms. The assumed sludge density is based on measurements of sludge retrieved for development of the DWPF.
Page 4-11	For the purposes of this residual radionuclide inventory, 1000 L (264 gal) of contaminated ETF influent is assumed to remain in the ETF Receiving Tank after D&D activities for the tank are completed.	The residual radionuclide inventory is based on the design and operational history of the tanks and the SRS experience in cleaning HLW tanks.

	Using the dimensions of the ETF Basins and a conservative estimate of 7.6 cm (3 in) of sediment left in the basins, the residual radionuclide contribution of ETF Basins is less than 1 Ci; therefore, the contribution is insignificant and the ETF Basins have not been included in this inventory estimate.	It was assumed that closure of the ETF basins would allow no more than three inches of sediment to remain in the basins. Using the dimensions of the basins and the three-inch thickness, as well as the concentration of radionuclides observed in the sediment, the sediment could contain no more than 1 curie of radioactivity. Thus, the basins were screened from further consideration.
Page 4-12	For the purposes of this residual radionuclide inventory estimate, the amount of residual radionuclides associated with the process sewer lines was calculated by Mr. Clifford Cole, Sr. (Cole 1996c). Mr. Cole conservatively assumed that the highest contamination level reported represents a homogenous concentration of radionuclides in the soil along each sewer line. Mr. Cole also assumed that each sewer line is 1524 m (5,000 ft) long, the excavation is 3 m (10 ft) wide by 3 m (10 ft) deep, and the soil density is 1920 kg/m^3 (120 lb/ft^3).	The highest observed contamination was imputed to all of the soil associated with the sewer line. The dimensions of the sewer line were conservatively assigned.
Page 4-13	For the purposes of this residual radionuclide inventory estimate, 25 Ci of alpha emitters and 10 Ci of beta/gamma emitters will remain in each tank after they have been emptied and decontaminated For these four tanks, a total inventory of 100 Ci of alpha emitters and 40 Ci of beta/gamma emitters is assumed. The alpha activity is assumed to be composed of 40 percent ^{244}Cm, 50 percent ^{238}Pu, and 10 percent ^{239}Pu. The beta/gamma activity is assumed to be due to only ^{137}Cs.	The residual inventory is based on the maximum observed concentration of radionuclides in the tanks and the estimated volume of residual material. The isotopic distribution of alpha emitters is based on analysis of material removed from the tanks. The assignment of the beta/gamma activity to only ^{137}Cs was derived from the remediation work plans. The review team challenged this assignment. A revised assignment, based on fission product yields, is provided in the response to Condition 3.
Section 4.1.3 S Area		
Page 4-14	For the purposes of this residual radionuclide estimate, 3,785 L (1000 gal) of typical DWPF sludge slurry is assumed to remain in the DWPF canyon building and 189 L (50 gal) of typical DWPF sludge slurry is assumed to remain in the Low Point Pump Pit after D&D activities are completed.	The volume of residual waste in the DWPF and the Low Point Pump Pit is based on the design of the facilities and operational history to date.
Section 4.1.5 Spills within the GSA		
Page 4-14	For the purposes of this residual radionuclide inventory estimate, all spills with an activity of less than one Curie are considered to be insignificant and have not been included.	One Curie is a very small fraction of the total residual radioactive material in the significant sources (those listed in Table 4.4-2), thus, it was judged appropriate to neglect sources less than one Curie.

Section 4.1.6 Other RCRA/CERCLA Sites		
Page 4-16	During the course of work on the Composite Analysis, management determined that a separate disposal facility for Environmental Restoration waste was not warranted. The inventories for the four facilities described above were added to that of the E-Area trenches.	Since a separate disposal facility for ER waste would not be built, it was assumed that the waste originally assumed to be consigned to the ER disposal facility would be disposed in the E-Area trenches.
	The sediments in the streams that bound the GSA, Four Mile Branch and Upper Three Runs, have potentially been contaminated with radionuclides released to the environment during operations at the SRS. As with other potential sources of radioactive material, only the sediments within the GSA are considered because it is those sources that could influence decisions regarding operations of the LLW disposal facilities.	Since the focus of the CA is the management of the active LLW disposal facilities, it was assumed that only those sources within the GSA would influence decisions on the operation of the LLW disposal facility. If a source outside the GSA were to contribute significantly to the CA dose, the actions taken would be to remediate the source rather than to alter operations of the LLW disposal facility.
Section 4.2 Excluded Sources		
Page 4-17	Facilities that have never been associated with the processing, management, or disposal of radioactive materials or waste such as the Burma Road Rubble Pit, the H-Area Acid/Caustic Basin, and the 284-10F Maintenance Shop. Such facilities are assumed to be free of radionuclide contamination.	Operational histories of each facility on the SRS are known. For those facilities that are known not to have radioactive material, it was judged reasonable to exclude them from the CA.
	Administration buildings such as offices, control rooms, laundry rooms, or clothing change rooms. Although these facilities may support other facilities that manage or dispose of radioactive materials or waste, sufficient controls are assumed to be in place to ensure that these facilities are free of radionuclide contamination.	Radiological control requirements to protect workers ensure that such facilities will have little, if any, residual radioactive material.
	Temporary storage facilities such as material staging areas, waste storage buildings or pads, or equipment storage areas. These facilities are assumed to be free of radionuclide contamination because either the probability of radioactive contamination is low or they can be completely decontaminated of all residual radionuclides.	Such facilities are unlikely to have been contaminated to any extent. Since the facilities are temporary storage or staging areas, the probability of leaking containers is small. Since they are storage facilities, radiological control requirements ensure periodic surveillance and clean-up of any released radioactive material.
Page 4-17	Radionuclides reported as "Gross Alpha" and "Other Alpha" are assumed to be ^{239}Pu.	This is based on isotopic analysis of samples. Additionally, the activity due to ^{238}Pu is assigned to ^{239}Pu to maximize the consequent dose (the half-life of ^{238}Pu is only 88 years, with plutonium's expected high sorption on soil, the ^{238}Pu would essentially decay away before migrating to a point of public access.

	Radionuclides reported as "Non-Volatile Beta" are assumed to be ^{90}Sr. Radionuclides reported as "Other Beta-Gamma" are assumed to be ^{137}Cs.	These assumptions are based on facility safety documentation. The review team challenged them. A revised assignment, based on fission product yields, is provided in the response to Condition 3.
	Radionuclides reported as "Radium" are assumed to be ^{226}Ra.	Because SRS has processed uranium rather than thorium, "Radium" was assigned to ^{226}Ra, which is a component of the uranium decay chain, rather than ^{228}Ra, which is a component of the thorium decay chain.
Section 4.3 Transport Pathway Identification		
Page 4-24	Factors that limit release of tritium to the atmosphere are likewise expected to limit ^{14}C releases.	Transport of tritium and ^{14}C to the atmosphere is via advection and/or diffusion of vapor species. Thus, factors limiting these processes (e.g., solubility in vadose zone moisture) for tritium will also limit ^{14}C.
	Based on the above observations, it was not considered credible that any doses due to the atmospheric pathway could come within orders of magnitude of the 100 mrem/yr dose objective or the 30 mrem/yr dose constraint for the maximally exposed individual. Therefore, the atmospheric pathway was eliminated from further consideration, as indicated in Figure 4.3-1.	The "above considerations" show that it is not credible for the atmospheric pathway to contribute significantly to the dose calculated to the maximally exposed individual in the CA.
Section 4.4.3 Source Term Estimates		
Page 4-47	Existing solid waste sites were modeled for their actual time of operation. These were 1954 to 1972 for the OBG and 1972 to 1994 for the MWMF. Lysimeters were treated as separate sources within the MWMF. The MWMF and OBG were modeled without a closure cap. The F- and H-Area Seepage Basins were modeled as closed systems, including a closure cap, beginning in 1988.	To reduce conservatism, development of the OBG and MWMF source terms included consideration of their actual time of operation. Since both facilities have a detailed history of waste burials, the source term was distributed over the operational period rather than assuming it was emplaced at one point in time. However, because the final closure of the OBG and MWMF has not been determined, these facilities were conservatively modeled without a closure cap. The lysimeters, which are located within the MWMF, had a shorter operational period than the MWMF. Thus, they were modeled as separate sources within the MWMF. Since the F- and H-Area Seepage Basins have been closed, they were modeled in their closed state.

Page 4-49	Both high level waste tanks and solvent tanks were represented as concrete monoliths, based on the approved closure plans submitted to the State of South Carolina. Each HLW tank was modeled as containing the expected residual radionuclide inventory after waste removal and closure. Key assumptions were that the tanks remain intact for 300 years and that infiltration was reduced by the concrete.	Since the tanks are made of thick steel, it was judged that 300 years was a reasonably conservative life for the tanks. Experience with the SRS lysimeters and PA modeling show that concrete is an effective barrier to infiltrating water.
	Process buildings, F- and H-Area Canyons, the DWPF, the Sand Filters and the 772-F laboratories, were modeled as a concrete slab, with the footprint of the existing structure, contaminated with the assumed inventory. No cap was assumed for these facilities.	For this initial iteration of the CA, with decommissioning plans for such facilities not available, these simplifying assumptions were judged appropriate.
	The only spills of sufficient magnitude (total activity > 1 curie) to be considered in the CA were associated with the high level waste tanks (d'Entremont, 1988). The spill inventory was added to the residual inventory of the tank group within which the spill was located.	This assumption was made to facilitate calculation. In responding to Condition 3, the flux to the water table for each of the spills, independent of the residual inventory of the tank group, was determined.
Section 4.4.4 Excluded Source Terms		
Page 4-50	The source term criterion developed as part of the screening methodology is based on an all-pathways dose analysis. The criterion defines a magnitude of release to the water table, below which associated impacts of the source term are expected to be considerably less than 1 mrem/yr. In order to develop this criterion, it was assumed that releases to the water table were not diminished by sorption or radioactive decay during transport in the subsurface, such that a release to the water table eventually became a discharge to a stream. Thus, a 1 Ci/yr release to the water table was considered a 1 Ci/yr release to a stream.	Screening methodology should be demonstrably conservative. Since the performance objective for the CA is 100 mrem/year, it was felt that a screening criterion of 1% of that limit was appropriate. Further, to ensure conservatism and to facilitate the analysis, no credit was taken for natural processes (sorption, dispersion, radioactive decay) that would act to diminish the radionuclide concentration during transit from the source to the point of exposure.

Page 4-66	Initially, the hypothetical individual was assumed to obtain all drinking water (730 L/yr) and all dietary fish (19 kg/yr) from a location on the Savannah River just downstream of the Savannah River Site (near South Carolina Highway 301). The individual was also assumed to be involved in recreational activities (boating and swimming) on the Savannah River at this location throughout the year. Flow of the Savannah River at this location is assumed to be 4000 cfs, which is considerably lower than the average flow rate of 10,500 cfs at this location, and thus provides an additional degree of conservatism in the calculated doses since dilution is underestimated.	Screening methodology should be demonstrably conservative. Even though it is unrealistic to think that an individual would obtain his entire drinking water supply from the river, this assumption is demonstrably conservative. The assumption that the individual consumes the average amount of fish for this region of the country is reasonable. However, to provide conservatism in the screening methodology, it was assumed that all of the fish were obtained from the Savannah River. Similarly, a conservatively low average flow rate was assumed for the river.
Page 4-67	It is highly improbable, however, that an actual dose would approach 1 mrem/yr at this release rate, given the number of conservative assumptions incorporated in development of this criterion.	The conservative assumptions include using flow rates about a factor of two lower than average flows and using the radionuclide with the highest calculated dose per curie released to represent all radionuclides.
	The release criterion of 10^{-4} Ci/yr was applied in two ways. If the total release of all sources of a particular radionuclide to the water table was less than 10^{-4} Ci/yr during the 1000-yr assessment period (Table 4.4-5), then that radionuclide was neglected for all sources in subsequent transport and dose calculations. In some cases, however, release of a radionuclide with multiple sources was greater than 10^{-4} Ci/yr from a few sources, but much less than 10^{-4} Ci/yr from others. In those cases, only the sources characterized by releases of the radionuclide greater than 10^{-4} Ci/yr were addressed. The results are summarized in Table 4.4-6.	Since the screening criterion of 10^{-4} Ci/yr was developed on the basis that such a release could result in a dose of no more than 1 mrem/year (1 % of the dose limit), it is clear that, if the total release of a particular radionuclide from all sources is less than the criterion, then the radionuclide cannot contribute a significant fraction of the dose limit and should be neglected. In cases where the total release from all sources exceeds the criterion, but only a few sources cause the criterion to be exceeded, the other sources can be appropriately neglected.
Section 5.1 Performance Analysis, Hydrologic Model		
Page 5-4	Because these streams incise this unit, the remaining groundwater moves downward across the Gordon confining unit. Therefore, these streams provide natural boundary conditions for most of the UTR aquifer, and were prescribed as discharge regions in the groundwater model. On the west side of the unit, hydraulic head values from a contour map of measured groundwater elevations are prescribed in lieu of natural flow boundaries.	The western side of the model domain does not have a natural flow boundary (e.g., it is not incised by streams). Therefore, a constant-head boundary was imposed, using the observed values for hydraulic head in this region. The response to Condition 2 contains additional assessment of the model boundary conditions.

Section 5.3 Surface Water Concentrations		
Page 5-55	In order to calculate surface water concentrations of radionuclides, annual flux of radionuclides (Ci/yr) to the surface water body must be specified, as well as flow rates of the water body. Average concentrations at specified downstream locations are calculated. These concentrations do not account for radionuclide decay during transit from the point of discharge from groundwater, as this decay is accounted for in the exposure and dose calculations (Section 5.4).	Concentrations of radionuclides in surface water were calculated by simply diluting the annual flux of radionuclide from groundwater to the stream into the annual stream flow. Since the methodology for dose calculations from radionuclides in surface water incorporates radioactive decay during transit from the point of discharge, such decay was not accounted for in arriving at the surface water concentrations.
Section 5.4 Exposure Scenarios		
Page 5-64	Reduction of radionuclide concentrations as a result of sorption on sediment surfaces and subsequent deposition, or as a result of water treatment, are not accounted for in the LADTAP XL model. Reduction due to radioactive decay during transit time (t_w) between discharge of radionuclides to the streams and consumption of the water is accounted for, based on an assumed average transit time of 1.5 days.	The assumption of no reduction of radionuclide concentration as a result of sediment deposition or water treatment is appropriate for tritium and is conservative for other radionuclides.
Page 5-65	Aquatic food consumption rates are assumed to be a maximum of 19 kg/yr for a hypothetical individual, and 9 kg/yr for the average member of the population (Hamby 1991a). Average time between harvest and consumption of fish and invertebrates is assumed to be 2 days, during which radioactive decay may occur.	The assumed consumption rates and the time between fish harvesting and consumption are derived from surveys of the regional population.
Page 5-65	Exposure to contaminated shoreline sediments is addressed in the LADTAP XL spreadsheet model using the NRC Regulatory Guide 1.109 equations for this pathway. A factor describing deposition of radionuclides on sediment was derived from empirical data obtained from the Columbia River. A shore-width factor of 0.2 (NRC 1977), also derived from experimental data, is used to represent the fraction of exposure to an infinite plane source estimated for shoreline exposures. Unlike the Regulatory Guide 1.109, which assumes a buildup time of 15 years, the LADTAP XL spreadsheet assumes the shoreline sediments have been exposed to the calculated radionuclide concentrations for 40 years (t_b), corresponding to the approximate operating period of SRS facilities.	The calculations are performed per NRC guidance except where site-specific modification is appropriate (e.g., longer time for sediment deposition representative of SRS operational history).

Page 5-66	In the LADTAP XL spreadsheet, the hypothetical individuals and populations are assumed to participate in swimming and boating activities for periods of time (t_s) consistent with those reported by Hamby (1991b). The time spent by a hypothetical individual swimming and boating is assumed to be 1.0×10^{-3} yr (8.9 hr) and 2.4×10^{-3} yr (21 hr), respectively. The population is assumed to spend 18 person-yr swimming and 126 person-yr boating.	The exposure times were selected to be consistent with values obtained in surveys of the local populace.
Section 6.1 Sensitivity Analysis, Sensitivity to Point of Assessment		
Page 6-1	To understand the sensitivity of the results of this analysis to the point of assessment, doses associated with ingestion of water from Upper Three Runs (UTR) and Fourmile Branch (FMB) were calculated (Section 5.5). The calculated drinking water doses assume an ingestion rate of 730 L/yr, which corresponds to the rate for a maximally-exposed individual. These doses do not include recreational pathways (i.e., swimming, boating, shoreline) or the fish consumption pathway because recreation and fishing on these smaller streams are not considered realistic activities. Average flows of these streams at the GSA are approximately 6 m^3/s for UTR and 0.4 m^3/s for FMB. These low flows are not expected to support large enough populations of fish to constitute a significant fraction of the diet of any user of the streams.	The drinking water scenario, although unrealistic, was selected to provide a simple, conservative analysis that would illustrate the sensitivity to, and need for, land use controls.
Section 6.2 Sensitivity to Stream Flow		
Page 6-3	Doses calculated at the points of assessment in the mouths of UTR and FMB (Section 5.5.2) are based on the average flow of these streams. To assess the sensitivity of the results to changes on stream flow, doses were also calculated for the minimum and maximum average annual flows	Since doses are based on a year of exposure, it was judged that the maximum annual flow rate was most appropriate rather than the maximum flow rate over a shorter period (e.g., instantaneous, monthly).

Section 7.4.1 Interpretation of Results, ALARA Considerations, Population Doses		
Page 7-3	The population doses calculated for the ALARA process in this composite analysis consider the populations served by the City of Savannah Industrial and Domestic Water Supply Plant (formerly Cherokee Hill Water Treatment Plant), near Port Wentworth, Georgia (10,000 persons), by the Beaufort-Jasper Water Treatment Plant, near Beaufort, South Carolina (60,000 persons), and the population in a 80-km (50-mile) radius of the SRS which may participate in recreational and commercial usage of the Savannah River (620,000 persons). Exposure to radionuclides of populations served by treatment plants is assumed to take place as a result of drinking water at concentrations found at the location of the plants, which are approximately 160 km downstream of the SRS. Exposure of the population in the 80-km radius is assumed to occur as a result of harvest of aquatic fish and invertebrates, and as a result of shoreline activities, swimming, and boating. Ingestion of contaminated water by members of this population is assumed to be negligible. The concentration of radionuclides in river water for the 80-km radius population is assumed to be the concentration 20 km downstream of the SRS (at Highway 301) - the same location assumed for the maximally-exposed individual (Section 5.3).	The assumptions regarding river water usage for community drinking water supplies are reasonable because such use is currently taking place. The exposure of the 80-km population via a recreational scenario (harvest of aquatic fish and invertebrates, and as a result of shoreline activities, swimming, and boating) is reasonable, based on current activities of this population.
Page 7-3/4	The flow rate of the Savannah River at the location of these plants is assumed to be 13,000 cfs, which is the estimated average flow rate for this location (Hamby 1991b). A travel time of 4 days for radionuclides leaving the SRS before consumption is assumed, which includes transit down the Savannah River and residence in the water treatment system. Individuals in the population exposed are assumed to, on the average, consume water at a rate of 370 L/yr.	The 4-day transit time is based on studies of the travel time for conservative (i.e., non-sorbing) contaminants from SRS streams to the Savannah River estuary. The average water consumption rate is based on studies in the literature where dietary intake was determined by population surveys.

Section 7.4.2 ALARA Analysis		
Page 7-5	This maximum cost is calculated assuming dose is reduced to zero, at an upper-end cost of $10,000 per person-rem and assuming a dose integration time of one year. The many conservative assumptions that went into estimation of population dose further maximizes this cost.	For conservatism in the analysis (i.e., to maximize the cost benefit of actions potentially taken), it was assumed that the action would reduce the dose to zero, rather than a fraction of the base case dose (i.e., 25%). Similarly, the maximum dollar equivalent of collective dose, $10,000 per person-rem, recommended by USDOE was used to maximize the calculated benefit.

Notes:
Acronyms are generally not spelled out in the table due to space limitations. The Assumption column in the table may contain acronyms that are spelled out since this column represents direct quotations from the CA document. The following acronyms are used in the table.

ALARA	As Low As Reasonably Achievable
CA	Composite Analysis
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
D&D	Decontamination and Demolition
USDOE	U.S. Department of Energy
DQO	Data Quality Objectives
DWPF	Defense Waste Processing Facility
EAV	E-Area Vaults
EPA	U.S. Environmental Protection Agency
FMB	Fourmile Branch
GSA	General Separations Area
HLW	High-Level Waste
HQ	Headquarters
ILT	Intermediate-Level Trench
LAW	Low-Activity Waste
LFRG	Low-Level Waste Facilities Federal Review Group
LLW	Low-Level Waste
MCL	Maximum Contaminant Level
NRC	U.S. Nuclear Regulatory Commission
OBG	Old Burial Ground
PA	Performance Assessment
RCRA	Resource Conservation and Recovery Act
ROD	Record of Decision
SRL	Savannah River Laboratory
SRS	Savannah River Site
SRTC	Savannah River Technology Center
UTR	Upper Three Runs
WSRC	Westinghouse Savannah River Company

5.0 Condition 5

Disposition of all composite analysis review team comments (see attached enclosure Appendix G & H Review comments from Composite Analysis).

Appendices G and H from the Review Team Comments are not included with this SA. Table 5-1 is a compilation of the Review Team Comments taken from Appendix H of their report. The table lists each comment and the action that will be taken on that comment.

Table 5-1 Review Team Comment Disposition

INVENTORY AND SOURCE TERM DEVELOPMENT

Com. No.	Comment	Action
1	The purpose of the CA is to determine the affect from all potential sources of exposure to the offsite receptor from sources that are reasonably expected to have become commingled with those from LLW disposal operations. The identification of those sources which contribute to the inventory considered in the CA is not presented in a clear or logical manner. As a result, the exclusion of potential sources of radioactivity outside of the GSA which could interact with the wastes disposed of in E-Area and Z-Area is not justified. Subsequent to the site visit, additional material was provided (Letter from W. L. Noll to Jeff Perry 4/21/98) to identify the additional inventory in M-Area and Tim's Branch which could contribute to the potential future doses associated with the GSA. This additional material does not appear to include all of the potential sources in M-Area which could contribute to the potential future doses from the GSA. Most notable is the lack of mention of the numerous tanks of sludge and other radioactive materials in M-Area. Consequently, there is no basis to conclude the inventory has been rigorously estimated in the CA.	The inventory has been revised to include all significant sources in A and M Areas. See response to Condition 3.
2	In a number of cases, nuclides were incorrectly reported or activity was assigned to nuclides without sufficient justification. Examples include: (d'Entremont, 1988) - For the high level waste spills reported in this reference, all curies were attributed to Cs-137 and decayed using a 30 year half-life and subsequently screened out. This is not acceptable in light of the radionuclide distribution that is known for the various high level waste tanks. (Cole, 1996h) Table 1.2 - The unassigned beta-gamma activity was not accounted for in the Residual Radionuclide Summary for the spill at Tank 37. (Cole, 1996d) - The source term summary charts given in this reference do not correlate with the column in the Residual Radionuclide Inventory report that represents the source	The inventory has been revised to include all radionuclides in sources that had been assigned to only ^{90}Sr and ^{137}Cs. See response to Condition 3.

Com. No.	Comment	Action
	term for the Soil/Debris Consolidated Facility.	
3	The source term for the Old Burial Grounds is stated to be the COBRA database. While it is understood that the ER report titled "Source Term for the Old Radioactive Waste Burial Ground (ORWBG)-Savannah River Site WSRC-RP-97-0119 was issued in October 1997 - and hence was unavailable for the development of the CA, this should be used as it provides a much more in-depth analysis and justification for the source term used. In any future use of this data however, it should be explained how the Constituents of Interest (COI) were derived. The stated COIs are not the same as the radionuclides that the CA determines to be the principle contributors to dose. The differences need to be justified.	Per CA maintenance, will address this and other applicable estimates of OBG inventory, as well as revisions of other source inventories, during the next revision of the CA
4	Assumptions regarding the radionuclide distribution and its' completeness are stated with no justification in numerous places throughout the document. The lack of clearly stated assumptions and justifications severely undermines the credibility of the analysis. The use of assumptions is of special significance to the high-level waste tanks. The heel remaining in the tanks is likely to be a significant contributor to the overall radionuclide inventory for the GSA. The CA does not provide a justification to support the heel estimates in the CA as conservative estimates.	See response to Condition 4.
5	The CA includes a review of the inventory of radionuclides considered and not considered. The initial list of radionuclides to be considered is based on the existing records, which are associated with some uncertainty. The estimates included in the analysis range from well justified disposal records from recent disposals to best estimates from process knowledge or knowledgeable individuals. These latter estimates cannot be justified beyond being the best information available.	See response to Condition 2.
6	The estimates of inventories and radionuclides in the CA appeared to be derived from referenced documentation, but the documentation in Cole, Hsu, Lux, and Shappell is a compilation of notes and assumptions. This approach attributes more	Per CA maintenance, will address this and other questions related to estimates of inventory during the next revision of the CA

Com. No.	Comment	Action
	credibility to the references than is warranted. Much of the referenced inventory material should be presented in the CA as data summaries or appendices, rather than being regarded as referenceable documentation.	
7	The inventory information in the CA includes extrapolations from known data. The degree of justification to attributed to these extrapolations ranges from well justified to the best available estimates.	Per CA maintenance, will address this and other questions related to estimates of inventory during the next revision of the CA
8	The CA includes the effects of CERCLA in the CA, but includes those agreements which are prescribed by RODs, and those which are expected to be included in RODs. The speculative CERCLA actions included in the CA may not be part of the ultimate RODs. In discussions during the site visit, the potential for this to occur was acknowledged, and corrections were to be addressed as part of CA maintenance. The CA maintenance plan has not yet been developed. The inclusion of speculative outcomes of the CERCLA process results in the CA being a potentially non-conservative representation of the site. Similar assumptions were made with regard to D&D actions, where no binding agreements exist at this time, but expected outcomes were used for the CA. The use of assumptions is of special significance to the high level waste tanks. The heel remaining in the tanks and the inventory left in the HLW piping systems are likely to be significant contributors to the overall radionuclide inventory for the GSA.	The CA maintenance plan has now been developed. The plan requires, per USDOE Order, annual reviews of the CA. The annual reviews will capture changes in CERCLA, as well as other, actions from those assumed in the CA. See the attached maintenance plan.
9	The assignment of beta-gamma activity to radionuclides in numerous places has not been justified. The responses to comments provided a great deal of the justification for the problem areas noted. However, each source term needs to be reviewed to ensure that the document clearly provides the rationale behind the assignment of these isotopes. One example that still needs to be addressed is found on Page 4-4, *Old Solvent Tanks (S1 -S22),* the last sentence on this page indicates that the beta-gamma activity is assumed to be Cs-137. It is unclear why only Cs-137 is assumed to be present and not Sr-90. Both are beta emitting fission products commonly found together. (This comment	See response to Condition 3.

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	was raised during the site visit). The same comment applies to solvent tanks S23-S30, S32, and the new solvent tanks H33-H36.	
10	Pg. 4-42 - ^{137}Cs is screened from further consideration due to "All of these radionuclides, with the exception of ^{126}Sn and ^{90}Sr, are fairly short-lived and were excluded from further consideration in the Composite Analysis." ^{137}Cs is not a short-lived nuclide compared to the other nuclides in this list. The reason given verbally for excluding this nuclide is due to the Kd value of 100. It is not apparent to the reader that this is a conservative assumption since other nuclides with Kd values in this range do appear to be significant contributors to the dose in the surface water. Both the F-Area and the H-Area tank farms appear to be sufficiently close to surface water that it is not unrealistic to expect to see Cs contamination in the FMB over the course of the compliance period. Cs-137 has already been detected in the surface water of FMB from the F- and H-Area Seepage Basins and the OBG. This existing source has been screened out because it does not pose a significant dose today. The analysis should determine the dose for the next 1000 years not just over the short term.	See the response to Condition 4.
11	Comment resolutions provided, some. rationale for determining that the D&D source term was comprehensive. However, it is still unclear what facilities will undergo D&D in place and which facilities will be disposed in the E-Area Vaults. A complete description of the long term planning for each facility that will dispose of waste in the active LLW disposal facility needs to be included. The information needs to be presented in such a way that the reviewer can determine that the entire source term from a facility will be accounted for.	Per the CA maintenance plan, which is now developed, the annual CA review will require comparison of assumed D&D source terms with D&D actions or plans. If there is a significant revision, a special analysis will be required. See the attached maintenance plan.
12	(WSRC, 1996b) - The last sentence, 2nd paragraph states that curies from fission products increase curies, they do not significantly increase consequences. This source term was developed for the safety analysis to determine a bounding accident, however, this assumption is not conservative with respect to the CA. Provide an estimate of	This source term will be re-evaluated in the next annual review of the Composite Analysis.

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	the fission products that were not included in the source term for these facilities.	
13	The document referred to for the nuclide inventory and activity estimate of the S23-S30 tanks is a series of spreadsheets and does not provide explanatory text. In fact, many of the documents referenced as supporting source term development lack descriptions of the assumptions used. The lack of assumptions within the composite analysis and supporting documentation make it impossible for the reader to determine how the inventories are bounded and what degree of conservatism is built into the estimates.	See response to Condition 4.
14	A more accurate method of determining the residual inventory would be to use information from D&D activities that have taken place at SRS, such as BLDG 232F. Much of the building's debris was released for disposal in sanitary landfills. In addition, some of the waste streams at SRS have been characterized by process knowledge by using area contamination surveys to estimate the contamination of waste removed from those areas. It does not appear that any of the historical information was used in validating the inventory data that was used in the source term development.	The information from D&D of 232-F would only be pertinent to other tritium facilities. As the CA is maintained, refinement of significant source terms, including information from the waste characterization program will be done. See the attached maintenance plan.
15	Page 5-16, last para., Existing residual activity in the streams as a result of many years of operational releases was not considered in this analysis. Even though these operational releases will cease in the future, some of the radionuclides will remain in the sediment and biota and therefore contribute to exposures of offsite individuals. It is stated that it was not included because this source will not influence the waste management decision. This should be reconsidered if a decision is made by the LFRG regarding inclusion of all sources on site.	In response to Condition 1, the CA is now focussed on a single point of compliance at the UTR mouth. Except for releases to Tims Branch (which have been considered in response to Condition 3, and incorporated into the response to Condition 1), essentially no radionuclides have been released to date to UTR. See the response to Condition 3.

POINT OF ASSESSMENT AND PATHWAYS ANALYSIS

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16	This requirement has simply not been met. The document does not clearly identify the point of assessment. During questioning at the site visit, the exact location of the point of assessment was not clearly identified. At the end of the discussion, Elmer Wilhite explained how the point of assessment moved during the preparation of the CA. Wherever the point of assessment is, it is not justified. Criteria 6.2.1.1, 6.2.1.2, and 6.1.1.3 have also not been met. The point or points of public access reasonably expected for future members of the public for the time period of the assessment have not been defined in the existing CA. The point or points of assessment that have been selected are not supported by land use plans or reasonably conservative assumptions that are justified. In the CA, the less than conservative assumption is made that land use controls will persist in perpetuity, but documentation to support such an assumption is based on a "Future Use Report." Finally, any changes to the point of assessment as a function of time have not been discussed, identified, or justified. For any of the possible points of assessment, such as the A-Road bridge, the confluence of Upper Three Runs/Four Mile Branch and the Savannah River, Lower Three Runs, or the 301 bridge, there are inconsistencies in the analysis. For example, the effect of M-Area is not addressed in the discussion of Upper Three Runs, and the effect of the production reactors is not addressed in the discussion of the 301 bridge. The only scenario considered in the base case for the consumption of drinking water is with the point of assessment at the 301 bridge. These requirements suggest the point of assessment needs to be clearly presented and justified throughout the time period of assessment in the CA. The point of assessment is tied to the exposure scenarios considered in the CA. Consistency between the point of assessment and exposure scenarios needs to be maintained. Most importantly, the closest point of public access which is a point of	As determined by the LFRG, the point of assessment is the mouth of UTR. See the response to Condition 1.

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	assessment needs to consider the drinking water scenario. Postulating the closest exposure scenario as a base case which includes the consumption of drinking water at the 301 bridge without the consideration of contamination from Lower Three Runs, the SRS production reactors, and Vogtle Nuclear Power station is incomplete and inconsistent. Similarly, a point of assessment that is closer to the GSA that includes the consumption of water should be considered.	
17	The use of a point of assessment at bridge 301 does not seem to be conservative. The rationale for this point is that there is a gauging station at the bridge and hence an accurate flow. The verbal statements that no appreciable inflow into the river occurs between the SRS site boundary and the bridge has not been justified. With an annual rainfall of 124 cm/yr and considering normal runoff, the argument that there are no major streams flowing into the Savannah River between the SRS boundary and the 301 bridge does not provide adequate justification for the point of assessment.	See the response to Condition 1.
18	The supplemental information provided with regards to the sensitivity to the ground water divide seems to provide a good case for establishing an offsite point of assessment during the institutional control period. This information needs to be included in the CA. Alternate off-site points of assessment that should be considered are the confluence of Lower Three Runs with the Savannah River and the SRS boundary at Steel Creek.	See the response to Condition 1.
19	The guidance given for the preparation of the CA states that dose "to a potential future point of public access must be analyzed and the resulting dose to a hypothetical future member of the public determined." A residential scenario (including drinking water) at the mouth of FMB or UTR seems to be a more realistic scenario for the out years. In the near term, a residential scenario at the mouth of Steel Creek just south of the current SR boundary) seems to be defensible - this would allow for an analysis of the impact of the cumulative tritium dose.	See the response to Condition 1.
20	Section 2.3.7.2, page 2-24, paragraph 1, The	Hilton Head has not yet begun

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	Hilton Head population, which will soon be using Savannah River water, should be included in the dose calculations.	using river water. As water usage at Hilton Head changes, the impacts, if any, will be assessed in accordance with the maintenance plan (see the attached plan).
21	Section 2.4.1 Points of Assessment Although this discussion has no answers per se, I offer the following counter arguments to both the scenarios and locations that were selected and suggest that they are not only not conservative but not all that meaningful to the question that is being asked. If you put someone very far away and expose them in a limited way for a very short time than all sites look wonderful. The assumption that land use will be restricted perfectly for 1000 years is indeed optimistic at best. Particularly when the source that is referenced encourages as much recreational use as possible among other things. For example if parks etc are created then water from either UTR, FMB or even groundwater could be used for drinking. There could be community gardens etc. Another example residential use could indeed take place opposite the site at the mouths of UTR and FMB. This would increase not only the possible exposure routes but also the duration of these exposures. One is not trying to predict the exact future here but it is important to adequately bound the possibilities so that sound management decisions can be made. Placing the first all pathway location some 20 km downstream of a very large site might reflect the present worst case but by no possible means would it reflect the future worst case. Likewise assuming the nearest population dose will be 160 km away for the next 1000 years does not seem credible.	See the response to Condition 1.
22	Page 7-3, para. 7.4. 1, The future population of the 80 kilometer (km) area around SRS may be underestimated. Should the extrapolation of population, based on the 1990 U.S. census data, be extended to the period of time when the highest doses are cast? It is not clear from the CA guidance that this is acceptable or that additional uncertainty analyses should be performed.	See response to Condition 1.
23	This requirement is not fully addressed in the	As determined by the LFRG, the

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	CA. Reference to the comments relating to the point of assessment should be made with respect to this requirement. The scenarios described in Sect. 2.4.2 in the CA for the base case utilize average flow rates, and the only drinking water consumption is associated with the point of assessment at the 301 bridge. The discussion in Section 5 relating to the ingestion of surface water makes reference to the ingestion rate of 730 L/yr for a maximally exposed individual and 370 L/yr for an average adult. In the discussion that follows, the rate selected for the analysis is not identified. In Section 2.4.2, a recreational scenario is identified, which is supposed to be described in Sect. 5.4. This description is missing. As described in the site visit, the recreational scenario includes all pathways presented in Sect. 5.4 except the drinking water pathway. The PAs for E-Area and Z-Area considered other exposure scenarios that were much closer to the disposal facilities. In the CA, the PA exposure scenarios were not discussed, based on a future scenario that excluded individuals from the SRS throughout the time of assessment. The extended institutional control period was based on a "Future Use Project Report." This report was prepared for the USDOE with a listing of recommendations by stakeholders. The closure plans for the GSA, E-Area, or Z-Area were not provided. Land Use Plans for the SRS were not provided. The CA Maintenance Program was not provided. There were no CERCLA RODs identified that included an extended period of institutional control. The exposure scenarios addressed in the CA were not justified.	point of assessment is the mouth of UTR and the exposure scenario is the recreational fisher person. See the response to Condition 1.
24	The CA used a value of 23 hrs/yr of shoreline usage for that pathway. The reference document (Hamby, D. M. 1991b - pg. 26) refers to that figure as the exposure for the average individual. It seems to be more conservative to use the calculated maximum individual shoreline usage of 35 hrs/yr for calculating the dose to the maximally exposed individual.	Per CA maintenance, refinement of exposure parameters to best match the intent of the CA will be done. See the attached maintenance plan.

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25	(Hamby, D. M. 1991b - pg. 3, 2nd column, first full paragraph) - This paragraph excludes pork and chicken from the analysis on the basis of commercial feeding practices for these animals. It is common for individuals to let their hogs and chickens graze on a small farm. The exclusion of these two sources of potential uptake is not reasonable.	Per CA maintenance, refinement of exposure parameters to best match the intent of the CA will be done, including consideration of animals raised on a small farm. See the attached maintenance plan.
26	Hamby, D. M. 1991b - pg. 9. At some point during the CA maintenance period, it would be reasonable to do a scoping assessment of the radionuclide levels found in the American Shad.	Per CA maintenance, refinement of exposure parameters to best match the intent of the CA will be done, including radionuclide levels in various species such as the American Shad. See the attached maintenance plan.
27	To exclude a drinking water pathway is not reasonable. In establishing a point of assessment, a drinking water pathway must be assumed as part of a complete residential scenario.	Per the LFRG's determination, the CA point of assessment is the mouth of UTR where, due to the SRS land use plan, a residential scenario is not likely. See the response to Condition 1.
28	Pg. 6-1, Section 6.2, 2nd paragraph, last sentence - Since fish often feed at the mouths of streams, it is not apparent that this last statement is correct. It seems to be not unreasonable to assume that there is a large enough fish population to support a significant fraction of the diet of a user when considering the fish in the stream and those located at the mouth of a stream.	Per LFRG direction, the recreational fishing scenario, as defined in the CA, will be used. See the response to Condition 1.
29	The information describing the disposal site, its location on the USDOE site, and its proximity to other sources of radioactive material presented in the CA is derived from the PAs for E-Area and Z-Area. The sources of radioactive material and the methodology for assessing the migration of radionuclides are described with comments regarding those descriptions provided in previous comments. As noted in these comments, some of the potential sources of radioactivity, which could interact with the disposal facilities, were not described. The exposure scenarios following transport and the point of assessment also are discussed in previous comments. The scenarios selected for the CA are questionable.	Per LFRG direction, the recreational fishing scenario, as defined in the CA, will be used at the mouth of UTR. See the response to Condition 1.
30	Section 6.3 - The assumption that there will be no public use of the SR site for the next	Per USDOE guidance, the SRS Land Use Plan provides sufficient

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	1000 years does not seem credible. Provide a description of the types of controls to be established to ensure that there will be no public access to the SRS for 1000 years.	basis for the assumption of no public use.

SENSITIVITY AND UNCERTAINTY ANALYSIS

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31	The determination of the important parameters and assumptions which influence the conclusions of the CA was not presented in the CA. Several parameters and assumptions were discussed during the site visit which contribute to the conclusions of the CA, but the overall importance of these discussion topics, which are included in the minutes of the site visit, to the conclusions of the CA have not been established. Alternative land uses and remedial actions are not addressed in the uncertainty analysis. The CA provides a set of possible outcomes for CERCLA and RCRA and analyzes these remedial actions. Changes in the CERCLA or RCRA actions would be addressed as part of the CA maintenance plan.	The CA maintenance plan has now been developed. The plan requires, per USDOE Order, annual reviews of the CA. The annual reviews will capture changes in CERCLA, as well as other, actions from those assumed in the CA. See the attached maintenance plan.
32	The sensitivity and uncertainty of the results is presented in the CA, but in a manner which is not consistent with the requirement. Alternative land uses are not considered; however, the consumption of drinking water from FMB and UTR is considered. The sensitivity and uncertainty analysis considers changes in the streamflow from an average condition to a maximum or minimum condition. The uncertainties in the inventories for the disposal facility and other contributing sources are not analyzed, and doses are not calculated for ranges in the inventory estimates. Alternative remedial actions were not addressed in the analysis. Alternative closure plans were not considered. Alternative transport or site characteristics were not considered.	Alternative use of lands was considered in Section 6.3 of the CA. Uncertainty arising from inventory values has been assessed in response to Condition 3. See the response to Condition 3.
33	The major shortcoming to this section (Chapter 6) on sensitivity analysis is the lack of any work done related to the source term and the unsubstantiated statement that the source term is bounding and conservative. Further, there does not seem to be any work done in the release and fate and transport area either. The expected analyses would include attributes such as Kd values, release rates, infiltration rates, etc. Lastly, the sensitivity of the results to reasonable scenarios is not adequate. On the one hand, the land use	Per LFRG determination, the recreational fishing scenario is to be used at the mouth of UTR. See the response to Condition 1. Uncertainty arising from inventory values will be assessed in response to Condition 3. See the response to Condition 3. Uncertainty with respect to scenarios such as zucchini boat

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	document encourages more recreational use of the site, but on the other hand, the CA document indicates that recreational use is not realistic. The document needs to address more clearly what uses there may be and what doses may result. By encouraging use of the land, there will be additional public exposure. Recreational scenarios other than the traditional swimming and boating might need to be considered such as frog gigging and zucchini boat racing. The sensitivity area is especially important since there are so many unknowns and of course the future is unknown. The only way to better understand the potential areas of concern are with a thorough sensitivity and uncertainty analysis.	racing will be considered if such scenarios are defined.
34	In Section 6.5, an explicit sensitivity analysis of the results of the CA to the source term needs to be performed. Most of the data used for source term information have not been validated and hence it is not known whether this represents a reasonable representation of the source term. Lacking a validation of the source term, a sensitivity analysis must be conducted to show the reasonableness of the analysis.	Uncertainty arising from inventory values has been assessed in response to Condition 3. See the response to Condition 3.
35	In reviewing Section 7.3, one really cannot conclude much about the effect of sensitivities, since such a limited amount of sensitivity analysis was done. Also, it is not so much the point of assessment that is likely to be the most sensitive, but rather, it is how long a, period of time that the assessment must consider, and what the people are doing there during that time period. Lastly, the document once again cites the conservatism of the analysis but gives the reader absolutely no idea of the potential magnitude of such a statement. For instance, does the analysis overestimate the potential dose by a factor of 2, 10, 1000, 1,000,000 etc. This needs to be stated and justified.	USDOE guidance specifies the time of assessment as 1,000 years. Quantification of the degree of conservatism is not a requirement.
36	With reference to the section entitled "Sensitivity to Use of Land Not Permanently Controlled by USDOE" (discussion on Page 6-3), although future use plans do not call for release of the site for unrestricted use, and	Effects of remediation activities on hydrology in the GSA have been documented in (SRT-EST-98-154). These effects are minimal and would not influence

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	therefore, provide the opportunity for WSRC to conclude that such scenarios as the use of a drinking water well in the GSA are not realistic, the sensitivity analysis should reasonably assess the potential impact on the flow system, and thereby the doses projected in the CA, of plausible activities that could occur even though the present site restrictions continue. Rather than dismissing as unrealistic any foreseeable change in land use (e.g., on Page 6-4, "...large-scale irrigation is not practiced..."), and concluding that no further analysis is needed, it may be valuable to determine what magnitude of local, on-site land use changes would be necessary to alter the flow system, the hydrologic boundaries used in the models, and the assumptions regarding natural barriers. It should be noted that active remediation and disposal site capping, which potentially have significant impacts on the flow system, have only recently been implemented. Over the next several years, additional remediation, which may involve pump-treat-reinject (PTR) and capping, and other site activities that may involve substantial use of water and surface area in the GSA, could conceivably cause some of the changes in local hydrology that have been dismissed from further analysis in this section.	the CA results. See the response to Condition 2.
37	Reliance on recommendations included in a future land use plan is not an acceptable reason for not performing additional analyses of the potential impacts on the flow system of future land and water use changes.	Per USDOE guidance, the Land Use Plan is the basis for projections of future land use.
38	Page 4-15, para. 4.1.6, Recent events at SRS within the Environmental Restoration program have brought into question the disposal location of waste resulting from CERCLA actions. In particular, since disposal of seepage basin wastes may not be going into the E-Area soil trenches, should the analyses be changed or should additional sensitivity analyses be included?	Changes in remedial actions or planned actions must be assessed in the CA annual review, as mandated by the SRS CA Maintenance Plan, which is attached..
39	Section 6.4 - A general description is given of the effects of movement of the groundwater divide and is expounded upon in the supplemental information. The supplemental information states that it is not credible for the	See response to Condition 2.

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	groundwater divide to move significantly. Justification for this statement has not been provided. No discussion is given of the potential natural phenomenon that would cause a shift in the groundwater divide. This section needs to provide a discussion of the various mechanisms that would cause it to move and the likelihood of these scenarios.	
40	The sensitivity analysis (Section 6.0) should be re-worked and expanded, based at a minimum, on the re-analysis information provided by WSRC in the 4/21/98 memo. Table 6.1-1 indicates that the peak dose at the GSA for exposure to tritium through the drinking water pathway is 24 mrem/year, while the re-analysis estimates that it could be as high as 64 mrem/year. The entire sensitivity analysis should be carefully revised and re-analyzed to clearly place upper bounds on potential future doses. A major objective of the sensitivity analysis should be to set the direction for future studies and analyses that could most effectively reduce uncertainty in the overall CA. These studies and analyses, which may be conducted prior to completion of the technical review of the CA or could be more appropriately conducted as part of a CA maintenance program, should be viewed as part of the on-going effort to validate predictions of future physical conditions and future contaminant transport, and should be used to substitute actual data for assumptions. The value of the sensitivity analysis is the quantification of the various levels of uncertainty, which would provide direction on prioritizing future studies, so as to reduce uncertainty as much as possible, and thereby effectively improve the quality of the CA.	Per the LFRG, the CA has been refocused to a single point of assessment at the mouth of UTR, see the response to Condition 1. The SRS CA Maintenance Plan, which is attached, requires continuous improvement of the CA by test and research activities and special analyses.
41	While attempts to justify the assumptions in the CA have been made, previous comments are directed toward these justifications. In addition, the uncertainty analysis should quantitatively examine the assumptions associated with the inventory. Of particular note are the potential ranges in inventory which could be attributed to the various radionuclides. Additionally, Stewart was	See response to Condition 3.

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	identified at the site visit as the reference which was used to provide the activity distributions for the curies reported to be disposed of in the old burial ground, but Stewart is not listed in the references in the CA. So what was really used in the CA? While not challenging the distributions attributed to Stewart, the uncertainty analysis should examine the range in results associated with the range in the uncertainties in the radionuclide distributions derived from Stewart.	
42	Alternative land uses were not considered. Perpetual institutional control of the SRS was the only land use option considered in the CA. In discussions at the SRS, other land use-options were noted as possibilities to be considered as part of a CA maintenance plan. The CA maintenance plan was not provided. Variations in radionuclide inventories, site and facility characteristics, and transport parameters were not considered in the sensitivity and uncertainty analysis. Consequently, bounding estimates of the potential doses at the point of assessment for the time period of assessment were not provided in the CA. Alternative closure plans were not considered and alternative site and waste characteristics were not considered. Bounding analyses were not provided to provide some assurance that the dose constraint and dose limit would not be exceeded in the foreseeable future.	Alternative land uses were addressed in Section 6.3. The SRS CA Maintenance Plan, which has now been developed and implemented, requires an annual review of the CA versus changes in actions or plans with respect to such things as closure plans, etc. The maintenance plan is attached.
43	Page 6-3, para. 6.3, Since the guidance for a composite analysis requires that reasonable alternatives to land use be considered in the sensitivity and uncertainty analyses, it appears that at least one reasonable alternative has been excluded - a resident living on site. The SRS Future Land Use Plan has been approved locally and transmitted to HQ, but it is not clear if this plan will remain unchanged. It is also unclear how this plan will be implemented, (i.e. deed restrictions).	The SRS Land Use Plan, per USDOE guidance for the CA, provides the basis for not considering an on-site resident.
44	Page 6-6, *Sensitivity to Source Term,* states: "...the assessment of sources other than the two LLW disposal facilities used conservative, bounding assumptions to assess the maximum potential impact of these	See the responses to Conditions 3 and 4.

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	sources." The bounding assumptions used in the development and assessment of sources are not described in the document. As such, the conservatism in the development of source terms is not apparent. With the document lacking descriptions for the bounding assumptions and the existing information not being complete enough to determine a level of certainty, it is difficult at best, to determine what error factors or confidence intervals can be associated with the calculated maximum dose.	
45	An internal WSRC report entitled, "Impact of F- and H-Area Pump-Treat-Reinject Remediation Systems on the Old Radioactive Waste Burial Ground, (SRT-EST-98-154)", which was not used in the development of the CA since the CA pre-dated this report, is an analysis of the potential impact on the flow system in the upper (water table) aquifer of the active PTR systems in place at the F & H Areas and the cover recently installed at the Old Burial Ground (OBG). The report concludes, among other things, that these remediation activities will affect the flow system at F Area, E Area, and H Area, and that some impacts will occur in the short term (weeks and months), but other impacts will not be realized for years. This report was based strictly on a modeling analysis, which was designed to account for broad impacts on the flow system throughout a large area (i.e., the entire GSA), but also to account for relatively small scale impacts (i.e., impacts on water table elevations at each individual extraction or injection well). The additional information provided to the review team by WSRC on 4/21/98 reviews this report and concludes that there is "potential" for the ground water divide to change over time as a result of active remediation in the GSA, but that the magnitude of any such changes would be small. This conclusion (and presumably the decision not to explore this matter further) is not technically justified for the following reasons:	Per LFRG direction, as documented in response to Condition 1, the CA will consider a single point of assessment at the mouth of UTR. The bounding effect of all sources migrating to UTR is contained in the response to Condition 2.

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	a. The WSRC report did not specifically examine the impact of F & H Area remediation and capping of the OBG on the ground water divide. It analyzed the impact on the entire flow system at the GSA. Any conclusions drawn regarding the impact on the divide resulting from the nearby active remediation cannot rely solely on the results of the SRT-EST-98-154 report. b. Nothing in the WSRC report indicates that the magnitude of potential changes is either large or small. The modeling study did not vary the rate of pumping or reinjection at the F & H Area remediation sites, but used the design flow rates (200 and 150 gallons per minute for F & H Areas, respectively). c. Future undetermined active remedial activities (or other site operations in the GSA) will also have potential impacts on the local flow system, and need to be considered cumulatively, when they are in the planning stages. This WSRC report is an indication of the potential for disruptions in the flow system, upon which the effectiveness of natural hydrologic barriers rely. Actual data on the flow system in the GSA, and specific data on the location and dimensions of the ground water divide, are needed to quantify the response of the flow system to such perturbations in the future. The OBG cover has only recently been installed (1997). The PTR systems at the F & H Areas have not been operated at design capacity due to technical problems, and are currently scheduled to pump at design capacity by April 1, 1998 (H Area) and May 1, 1998 (F Area), according to a directive from the South Carolina Department of Health and Environmental Control (February 23, 1998 letter to A.B. Gould and J.V. Odum from Kim K. Hagan, Hazardous Waste Enforcement	

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	Section, Bureau of Land and Waste Management). Therefore, data needed to validate conclusions drawn in this modeling study are not currently available and probably will not be for some months or years. When such data (e.g., water table gradients and elevations in the immediate vicinity of the modeled location of the ground water divide) becomes available, a study should be performed to validate the results of the modeling analysis included in the WSRC report. Until such analyses are completed, it is premature and therefore, not technically supportable, to conclude that the magnitude of changes to the location of the ground water divide from local remediation activities will be small.	
46	The issue of uncertainty in the ground water divide should be treated more rigorously. Uncertainties in the cause of the ground water mound in H Area could impact flow directions and rates. Modeling the mound required reductions in horizontal conductivity and flow rates which may not be real. Lack of flux from the eastern edge of the model may also cause the model to underestimate flow rates. And the effect of the upward gradient in the three S and Z area wells has not been evaluated. Finally, there is a discrepancy between the tritium dose calculated for all contaminants reaching Four Mile Branch in the CA sensitivity analysis (29 mrem/year, page 6-5) and the Bounding Estimate of All GSA Contaminants Migrating to Either of the Streams provided to the review group via FAX on April 22, 1998 (64 mrem/year). These observations, taken together, indicate that the uncertainty in the model needs to be further evaluated.	Per LFRG direction, as documented in response to Condition 1, the CA will consider a single point of assessment at the mouth of UTR. The bounding effect of all sources migrating to UTR is documented in the response to Condition 2.

DATA QUALITY OBJECTIVES

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47	Section 3.2.3 and elsewhere - There is a statement in the last paragraph on page 3-3 that begins "All estimates and assumptions ..." Since the assumptions are critical to understanding the worthiness, if you will, of the estimates where are they documented and what sanity checks were made of them?	See the response to Condition 4.
48	Section 3.2.4 Spatial Boundaries - I question the adequacy of the domain. For example why were not sources on the other side of UTR considered? And if your point of compliance is at the 301 bridge why were not other on site sources considered?	See the response to Condition 1.
49	Page 3-6, Section 3.2.5: there needs to be more discussion provided on just what the "personnel knowledgeable" about the various waste streams provided and what they deemed representative. The concern is from a completeness standpoint. The nuclides of long term concern are seldom the ones that cause operational problems or show up on the near term radar screen. Typically the only way they are identified is by inference, scaling, derivation etc. The steps taken to ensure a complete inventory needs to be described.	See the response to Condition 3.
50	Section 4 in general - As the source term development is probably the most critical component of the composite it is most important that it be thorough, complete, defensible, credible and technically sound. There is not enough information provided to answer any of these questions. For example, two of the major potential sources, MWMF and OBG, just reference a COBRA database. No other information or discussion provided. Other sources just reference an "e-mail memorandum." Others like the Old Solvent Tanks just "assume" an activity with no explanation or justification. Then right on the heel of this assumption another is made which assigns entire groups of activity to one nuclide, again with no explanation.	See the responses to Conditions 2, 3, and 4.
51	It is understood that a good portion of the historic data regarding contributing source terms is limited, and in accordance with the April 30, 1996 document, *Guidance for a Composite Analysis if Interacting Source*	See the responses to Conditions 2 and 3.

Com. No.	Comment	Action
	Terms, the first, iteration of the composite analysis will use only the information at hand; no field samples will be collected for analysis. However, there is a need for discussion regarding the quality and level of certainty associated with the source term data collected and used in calculating the maximum dose. As an example, Page 1 - 1, Section 1.0 *Summary and Conclusions,* second paragraph states: "Two former LLW disposal facilities, the Mixed Waste Management Facility and the Old Burial grounds, are the major sources of these isotopes." Yet there is no discussion regarding the uncertainty associated with each source term developed and used in the composite analysis. In fact, the following statement is made in section 3.3.2, *Data Qualification:* "Ranking according to degree of certainty was not attempted because information with which to make these decisions is not complete." In order to understand the sensitivity of the calculated dose at the point of assessment with respect to the contributing source terms, some indication of the data quality and associated uncertainty must be established.	
52	Page 3-3, Section 3.2.2, *Step 2: Identify the Decision,* states: "The decision to be made in this application of the DQO Process is whether the resources available will provide a reasonably representative residual inventory upon which dose estimates for the Composite Analysis can be based. Unacceptable data quality or quantity will lead to unreliable estimates of doses." There is no discussion of the alternative actions that may result from the identified decision. In accordance with the EPA guidance document for data quality objectives, EPA QA/G-4, September 1994, possible alternative actions that may result from the decision question should be identified. In other words, since the decision is whether the resources available will provide a reasonable residual inventory from which dose estimates can be based, there should be some discussion on actions to be taken if available resources cannot provide for a	Revision of the application of the DQO Process to the CA will be considered as the CA is maintained. See the attached maintenance plan.

Com. No.	Comment	Action
	reasonable inventory.	
53	Page 3-3, Section 3.2.2, *Step 2: Identify Inputs to the Decision,* provides a discussion of the various sources that were used to create a residual radionuclide inventory for the composite analysis. However, there is no discussion regarding the establishment of a level of acceptability for the information being used for input into the decision. The EPA guidance document for data quality objectives, EPA QA/G-4, September 1994, indicates that when identifying inputs into the decision process, action levels should be established which define the basis for choosing between alternative actions. It would appear that some discussion is warranted in this section that describes a level of acceptability for the information where any information below the established level would be considered inadequate for providing a reasonable inventory estimate or at a minimum be used in assigning a level of certainty to the data.	Revision of the application of the DQO Process to the CA will be considered as the CA is maintained. See the attached maintenance plan.
54	Page 3-6, Section 3.2.5, *Step 5: Develop a Decision Rule,* states "The decision rule developed for this application of the DQO Process can be stated as: "If the radionuclide inventories identified for facilities and specific locations in the domain of interest are reviewed and deemed representative by personnel knowledgeable about waste streams and pertinent activities leading to residual radionuclides, then the inventories will be assumed to be appropriate for the Composite Analysis. If the information is unavailable or inadequate for a given facility, then the inventory will be considered incomplete and the composite analysis will not be considered comprehensive." A description of the level of acceptability for the information used for the radionuclide inventories should be included. Without a description of the level of acceptability or certainty as to what constitutes adequate versus inadequate data, a conclusion as to the sensitivity of the inventories to the estimated dose cannot be drawn. It does not appear from	Revision of the application of the DQO Process to the CA will be considered as the CA is maintained. See the attached maintenance plan.

Com. No.	Comment	Action
	the document that any of the data reviewed failed to meet the Decision Rule. Given the stated lack of source term information, it is surprising that none of the data reviewed failed the Decision Rule.	
55	Page 3-7, Section 3.2.6, *Step 6: Specify Limits on Decision Errors,* states: "There was no exclusion of data during the initial evaluation. Although a statistical analysis was not carried out, and confidence limits were not established, decision error was controlled through careful development, review and evaluation of data by qualified personnel." More discussion regarding controlling decision error is warranted. With the absence of alternative actions, levels of acceptability, and data confidence limits in the DQO process, the reviewer is lead to conclude that there was no mechanism for classifying any of the data as unacceptable, and no further evaluation of data will be conducted to establish levels of certainty.	Revision of the application of the DQO Process to the CA will be considered as the CA is maintained. See the attached maintenance plan.
56	Page 3-7, Section *3.2.6, Step 7: Optimize the Design,* states: "An alternative design would include field collection of soils at given facilities for radionuclide analyses. This would provide actual analytical data. However, the number of samples required in addition to the time and cost for sampling and analysis would be prohibitive for this initial characterization." This statement implies that additional characterization activities will occur, but there is no further discussion which describes what additional activities beyond the initial characterization are planned. This is especially relevant for the former LLW burial grounds that are major contributing source terms, but no level of certainty has been established.	Revision of the application of the DQO Process to the CA will be considered as the CA is maintained. See the attached maintenance plan.
57	Page 3-11, *Completeness,* in the context of data collection, completeness is used as a data quality indicator which is defined as the amount of collected data that is considered valid compared to the amount of data planned for. It appears from Chapter 3 that the data quality for each of the data sources was	Revision of the application of the DQO Process to the CA will be considered as the CA is maintained. See the attached maintenance plan.

Com. No.	Comment	Action
	designated, but no assessment of the needed data quality or quantity was made to determine if the data quality received was adequate.	
58	Page 3-11, Section 3.3.2, *Data Qualification*, this section states that the data sources were assigned numerical codes which classify the information according to type, but ranking according to degree of certainty was not attempted. However, the descriptions for each of the numerical codes used for data qualification on Page 3-12 all include statements as to whether the quantities and types of radionuclides are known or estimated. These descriptions appear to infer assigned levels of certainty based on the source of the information. Furthermore, page 3 -18 and Table 3.3 -3 indicate that 61 % of the radionuclide inventory and associated concentrations are considered known. Clarification is needed as to how 61% of the source term inventory can be assumed known if sufficient information is not available to ascertain any degree of certainty.	Revision of the application of the DQO Process to the CA will be considered as the CA is maintained. See the attached maintenance plan.
59	Page 3-11, para. *3.3.2,* although Data Qualification was discussed, no conclusions seem to have been drawn from this process, no justification that the data quality is acceptable and no recommendations for necessary future actions were made. The CA guide leads one to conclude that this DQO process may recommend future data/sample collection.	Revision of the application of the DQO Process to the CA will be considered as the CA is maintained. See the attached maintenance plan.

SUBSURFACE TRANSPORT

Com. No.	Comment	Action
60	Chapter 3, throughout this chapter there is a recognized need to identify and quantify radionuclides. Where was the physical and chemical form information captured? This information is integral to the transport mobility, release rate etc.	See the response to Condition 4.
61	The methodology given for estimating the release of radionuclides from the contributing sources is not complete. While the PAs contain a complete methodology, any degradation of waste forms is not included in the methodology for other sources. It is a simplified leach rate model from the waste form that does not include any consideration of the physical and chemical characteristics of the source materials and the site characteristics.	A simplified release model was judged adequate for this first iteration of the CA.
62	The modeling components selected for the analysis are reasonable and make use of the available data. The determination of the conservative nature of the methodology is difficult to assess. The scenarios considered for the CA are not apparently conservative.	No response needed.
63	The assumption is made that spills are added to the residual inventory of the tank group that they belong to. This is non-conservative because a source term that is already in the ground is being modeled as though it were encased in concrete with a 300 year tank surrounding it	In the response to Condition 3, the flux to the water table for each spill was assessed separately from the tank group.
64	The physical and chemical characteristics of the source materials and site characteristics are incorporated into the assignment of distribution coefficients to the radionuclides considered in detail in the CA. The CA includes all of the data as diskettes in Appendix B. The relationship between the input data files contained in the appendix and the understanding of the physical and chemical characteristics used in the CA is unclear. The relationship between the data in Appendix B and the release mechanisms is not clear.	See the response to Condition 4.
65	This requirement is addressed in the CA. As noted in many of the comments in this section, the justification or logic associated with many of the assumptions is debatable.	See the response to Condition 4.

Com. No.	Comment	Action
	However, there do not appear to be any significant changes to the conceptual model used in the CA as compared to the PAs for either E-Area or Z-Area.	
66	Criterion 6.3.3 is a similar requirement that is associated with this comment. However, this requirement speaks to the correctness of the conceptual model. The conceptual model used in the PA was developed for the close-in analysis of E-Area and Z-Area, where the point of compliance was about 100 meters away from the disposal unit. For the CA the conceptual model was extrapolated to include all of the SRS. As a result, the conceptual model does not include any additional potential mechanisms related to the areal extent of the confining units for the aquifers, and the potential mixing between aquifer layers away from the GSA.	Per LFRG direction, the single point of compliance for the CA is the mouth of UTR. The only sources outside the GSA that are to be considered are those in the A/M area. See the response to Condition 3.
67	Pg. 5-29, first paragraph - The first reason given for neglecting mechanical dispersion is that the time of assessment is 1000 years. Hence, "this amount of time is sufficient for arrival of the more concentrated portion of the plume at the location of concern,". With some nuclides of interest having high Kd values, it is not apparent that this statement is accurate. Justify this statement.	See the response to Condition 4.
68	This requirement is not clearly satisfied in the CA. As noted in other comments, the point of assessment is not well defined in space or time. Consequently, the conservative nature of the methodology cannot be assessed. There are indications from the omission of other potentially significant sources of contamination that the methodology used in the CA is not conservative. The transport of contamination is accomplished by the application of the PATHRAE, PORFLOW, and FACT models, which have extensive data inputs. The inputs to the models are provided in Appendix B, without a guide to the contents. Consequently, the files are mere compilations of numbers without meaning. Therefore, making a meaningful comment with respect to this criterion is not possible.	Per LFRG guidance, the CA point of compliance is the mouth of UTR. Only sources in the A/M-Areas need be added to those in the GSA. See the response to Condition 3.
69	The known physical and chemical	As the CA is maintained, re-

Com. No.	Comment	Action
	characteristics of the radioactive materials considered in the composite analysis are discussed in the CA. The effect these characteristics have on the source terms and the transport of radionuclides is also discussed in the CA. The correctness of the characteristics is difficult to establish because of the limited records available for old disposals, and the limited understanding of the behavior of the many different types of waste forms at SRS. The significant uncertainties associated with the physical and chemical characteristics of the radioactive materials considered in the CA should have been considered in the sensitivity and uncertainty analysis contained in the CA.	evaluation of the sensitivity analysis to include factors such as the characteristics of the waste will be considered. See the attached maintenance plan.

HYDROLOGY

Com. No.	Comment	Action
70	The mathematical groundwater flow model is based partly upon the assumption that there is an upward hydraulic gradient across the Crouch Branch confining unit. This gradient is assumed to naturally protect the aquifers beneath the Floridan aquifer system from contamination (Composite Analysis, p. 2-23). Using this assumption, the flow model was constructed for the Floridan aquifers above the Crouch Branch confining unit. However, no reference is provided for the above assumption. in the text or the accompanying Figure 2.3-5. Similarly, there are no supporting data in the Saltstone and E-Area Vaults Performance Assessment (PAs), which also rely on this assumption. Supporting data were provided during the review and should be referenced in the CA.	See the response to Condition 4.
71	No volumetric mass balance was performed on the amount of water flowing into the model compared to the amount exiting the model. This is a standard output for most models and its absence from the discussion in the CA, the two supporting PAs, and the reference documentation from Flach and Harris (1997) is troubling. Given precipitation, infiltration (and hence runoff), artificial recharge, discharge to the streams, and leakance through the Crouch Branch confining unit, a balance can be computed. It is unlikely that the model will balance in its present form because of the omission of flux through the northern and eastern model boundaries.	Presentation of mass balance information will be made in the next revision. See the attached maintenance plan.
72	The conservativeness of some model assumptions has not been verified or evaluated. One example is the assumption that the Crouch Branch confining unit has an upward gradient. Another example is the assumption of no-flow boundaries to the model. In both cases, if the assumptions are wrong, additional aquifers could become contaminated and travel times could be significantly altered. It is not clear whether these assumptions are conservative or not.	See the response to Condition 4.
73	The mathematical models utilized in the CA	No modeling exercise will ever

Com. No.	Comment	Action
	utilized the available site data. The PATHRAE and PORFLOW models were used in the PAs for Z-Area and E-Area. FACT was a model developed for the GSA that was used for the CA. As discussed in the site visit, the data to support the modeling of the entire GSA is incomplete. This lack of complete data to drive the three dimensional models used for the CA introduces additional uncertainty, which was not addressed in the uncertainty analysis.	have "complete" data. Data will be reviewed annually and incorporated in future revisions, per CA maintenance. See the attached maintenance plan.
74	The CA used essentially the same assumptions and justifications as those used in the PA's. The validity and adequacy of these assumptions is addressed in other comments.	Comment noted.
75	PORFLOW and PATHRAE are documented codes. LADTAP XL is referenced in the CA. FACT is documented in the appendix to the CA. All of these codes have been verified and validated to a reasonable extent.	Comment noted.
76	From the PATHRAE input files, it can be assumed that the precipitation runoff rate (40 cm/yr) plus the watershed infiltration rate (40 cm/yr) should equal the total precipitation. The total precipitation given in 124 cm/yr (pg.2-13). The remaining balance should be accounted for.	The balance is due to evapo-transpiration
77	The assumptions incorporated into the mathematical model used for the performance assessment were used in the composite analysis as well. These assumptions were identified in the PAs and CAs . Some assumptions are not well identified or justified. The other comments identify some of these examples. Additional examples are related to the site hydrology and are covered in Criterion 6.3.6.	See the response to Condition 4.
78	Calibration of the flow model indicates problems with the conceptual model and numerical model boundary Conditions. The model results as summarized in Figure 5.1-18 and Figure 5.1-19 show the effects of a large groundwater mound in H Area. This mound is not discussed in the CA but is thoroughly described in Flach and Harris (1997). Calibration of the model to incorporate the mound required significant changes in conductivity and in recharge. This included changes to the vertical and horizontal	See the response to Condition 2.

Com. No.	Comment	Action
	conductivity in the upper and lower aquifers and the tan clay as shown in Flach and Harris, Figures 26, 27, and 28 and Table 5. Increases to recharge are, on average, the equivalent of the annual natural recharge and range up to twice the natural recharge (Flach and Harris, Figure 22). From Figure 22 the artificial recharge can be estimated at 1.6 Mm^3 per year or 1300 acre-feet/year. The changes in conductivity are not supported by specific field data but are within the range of variability common in field permeability measurements. The increase in artificial recharge is poorly supported by anecdotal evidence of leakage in water and sewer systems (Flach and Harris, 1997, page 20). If leakages of over a million cubic meters per year are present, it should be possible to provide an accounting of known water production from water supply wells and discharge to water disposal systems to verify the model assumptions. The lack of such data calls the interpretation into question. An alternative to the model modifications of conductivity and artificial recharge is to account for the flux entering the eastern side of the model (see boundary Condition comment above). Treating this flux boundary as a no-flow boundary causes flow directions to track north along the eastern model boundary (see Figure 5.1-18) rather than westward to supply the groundwater mound. In addition, the use of phantom data points, or control data, in Figure 5.1-13 may be masking a true gradient that is more indicative of westward flow across the model boundary.	
79	The mathematical models used in the CA for analyzing transport are appropriate and provide calculated results which are representative of the results calculated in the PA	Comment noted.
80	This requirement is not clearly achieved in the CA. Assumptions have been used in the CA to formulate input data, but the justification and defensibility of the assumptions is not clearly presented. The relationship between the input data and the source of the input data by either	See the response to Condition 4.

Com. No.	Comment	Action
	field data, laboratory data, reference, or assumption is not presented in the CA. Input data such as the invariant infiltration rates and the distribution coefficients are not justified.	
81	The assumption of isolation of lower aquifers is at odds with site physical data. The CA states that the confining nature of the Crouch Branch confining unit in the GSA and the head-reversal phenomenon naturally protect the aquifers beneath the Floridan (sic) aquifer system from contamination? (CA, page 2-23). However, the CA Figure 2.3-5 and supporting data provided during the site review (Aadland and others, 1995, and Christensen and Gordon, 1983) show that the gradient from the Crouch Branch aquifer to the Gordon aquifer is thought to be downward immediately to the southeast of S and H Areas. The downward gradient can be seen by inspection of Figures 14 and 10 in Flach and Harris, 1997. This assumption is incorporated into the numerical model by virtue of the lower model boundary definition as a general head boundary with flux dependent upon head in the underlying Crouch Branch aquifer and overlying Gordon aquifer (Flach and Harris, 1997, page 11). The downward gradient present in the southeast comer of the model is a violation of the conceptual model assumption of no downward flow from the Gordon aquifer to the Upper Three Runs aquifer. Owing to the location of the sources in the General Separations Area and the probable flux into the model domain from the east, it is unlikely that contamination could reach the underlying Crouch Branch aquifer via this route. However, the protection of the Gordon aquifer is more complex than depicted in the model and is dependent upon accepting the heads, conductivities, and leakances as characterized in the model.	See the response to Condition 2.
82	Three wells in Z and S Areas are at odds with the conceptual model. Wells ZBG 1A, SCA 3A, and SCA 4A (Flach and Harris, 1997, Appendix C, pages 113 and 114) are completed 30 to 40 feet deeper than nearby companion wells in well clusters. In all three	This will be addressed as the CA is maintained. See the attached maintenance plan.

Com. No.	Comment	Action
	cases, the deeper well has a higher head than the shallower well, with the increase in head approximately equal to the difference in depth. This indicates a substantial upward hydraulic gradient in the water table aquifer of the Z and S Areas of approximately one-to-one. This phenomenon is not discussed nor accounted for in the conceptual and numerical models, even though it is at odds with the conceptual model of downward gradient in the Upper Three Runs aquifer. The impact on flow directions is hard to predict (with respect to conservatism) but the uncertainty associated with the model is increased.	
83	The flow model contains assumptions about boundary conditions which are not correct. a) The Gordon aquifer at Upper Three Runs Creek is defined as a no-flow boundary (Figure 5.1-1), when it appears that the Gordon aquifer continues to the northwest as part of the Steed Pond aquifer (see Aadland and others, 1995, Plate 3). The Gordon aquifer ceases only by definition because of the updip truncation of the Gordon confining unit. No data are presented on the hydraulic and hydrologic characteristics of the northwest continuation of the Gordon aquifer beyond Upper Three Runs Creek, so it is difficult to determine if this is a significant point. The model assumption is contradicted by the following statement from the CA: The Gordon aquifer is recharged both by precipitation within the GSA and by lateral flow from outside the GSA (page 6-6). b) The Upper Three Runs aquifer at Fourmile Branch is defined as a no-flow boundary (Figure 5. 1-1) when it appears that the lower unit, beneath the tan clay, continues to the southeast (see Aadland and others, 1995, Plate 3). Since leakage through the tan clay and discharge of the lower unit to Upper Three Runs aquifer are included in the model, this can be expected to be a flux boundary of unknown magnitude.	See the response to Condition 4.

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	c) The Upper Three Runs aquifer at the eastern boundary of the model between McQueen Branch and Fourmile Run is defined as a no-flow boundary (figure 5.1-1). The measured head map of Figure 5.1-13 shows that a westward flux into the model domain along this boundary is probable. Note also that the head map uses control data (invented data) to modify head contour lines in this area, potentially masking a larger gradient than shown.	
84	The measured head map of Figure 5. 1 -13 contains a sharp groundwater mound in Z Area (in the northeast part of the model domain) related to well ZBG 1A. The mound is not simulated by the model (see head map of Figure 5. 1 -11). The figures are from Flach and Harris (1997), Figures 11 and 36. Neither the CA nor Flach and Harris explain that the mound is the result of one data point, well ZBG 1 A, which was omitted from the model as an outlier after completion of the measured head map (Flach and Harris, Appendix E, page 137). In contrast, two other wells with anomalous head data in nearby S Area were omitted from the measured head map (wells SCA 3A and SCA 4A) The head data from all three wells should be treated the same.	See the response to Condition 4.
85	The CA does not provide intermediate calculations and results to demonstrate the CA calculations are representative of the site for similar situations. Comparisons between the PA results for E-Area and Z-Area, and the CA results are not provided. Concluding the PAs and CA are similar on the basis of the calculations has not been demonstrated.	In the next revision of the CA, consideration will be given to providing intermediate calculations and results.
86	The conceptual model used in the CA is consistent with the conceptual model in the PA. However, the additional components of a conceptual model for the SRS are not clearly introduced into the CA to ensure that regional subsurface phenomena and surface and groundwater interactions are properly considered in the CA. The material presented	Flux to the water table results are given in the CA to satisfy the intermediate results criteria. The results in Table 4.4-5 for the facilities labeled ILT, LAW and SLIT are for units in the EA PA. Entries under SALT are for the Saltstone facility. These data can

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	in the CA does not clearly address how the regional aquifer characteristics are included in the conceptual model.	be used to compare results.
87	The reliance on natural hydrologic barriers as effective mechanisms for preventing or controlling contaminant migration, are not adequately justified in this document, from a technical perspective. Additional data, and where appropriate, additional studies, must be provided to substantiate their effectiveness, or additional model uncertainty must be incorporated.	See the response to Condition 4.
88	This requirement speaks to the rigor included in the CA. While many of the assumptions in the CA related to the radionuclides have been examined, the examination has not been rigorous, as noted in previous comments. The source term evaluation similarly has questions concerning the rigor of analysis, as noted in earlier comments. The transport of radionuclides largely relies on the models used in the PAs for the two facilities that were extrapolated to the entire GSA, and the data driven FACT code, which has recently been developed and to some extent verified and validated. The lack of intermediate results, which are referenced to field or laboratory data, is a shortcoming in the CA that leaves many of the questions concerning the transport of radionuclides unanswered. This leads to uncertainties in results which have not been evaluated in the CA.	See the response to Conditions 3 and 4.
89	The assumption that anthropogenic changes will not alter the model results needs to be justified. To demonstrate that the CA is technically adequate, there must be more information provided on the assumptions that the hydrologic conditions that cause the natural hydrologic barriers will not change significantly over the time period of the analysis. As it stands, the only assurance that can be made is that institutional controls will prevent any on-site activity from disrupting flow conditions that would significantly impact the natural hydrologic barriers, and that off-site activities, such as large scale irrigation, are not likely. Since, therefore, no reasonable assurance can be given to justify the assumptions regarding flow conditions,	See the response to Condition 4.

Com. No.	Comment	Action
	there must be an analysis of the potential consequences of changes in the flow system. There should be a sensitivity analysis that determines the potential impact on the CA results of changes in hydrologic conditions that cause any of the three natural hydrologic barriers to fail to contain or retard migration. In addition to assurances on the future effectiveness of the natural hydrologic barriers, the document has not adequately demonstrated the current effectiveness of these barriers. The three natural hydrologic barriers - the ground water divide, the upward gradient in the Crouch Branch aquifer, and the incision of the upper ground water units by the three streams - are not well described in the CA, are susceptible to change as a result of local on-site and off-site activities, and are crucial factors in the CA results. References are provided to hydrogeologic studies (Aadland, et al, 1995 is the primary source) that provide the basic geologic and hydrostratigraphic data used in the CA. But what is missing is sufficient technical justification, through relevant studies and analyses, that support the assumption that these hydrologic conditions function effectively to contain contaminants or reduce their mobility, as described in the conceptual model of the GSA. There are no references provided in the document to studies or analyses that support the inferences drawn in the CA regarding the effectiveness of these natural barriers. If such studies or analyses exist, the document should include adequate discussion of their results and conclusions, and references should be provided. If relevant studies do not exist or if the conclusions do not support the assumptions made in the CA, there should be a plan to conduct the studies or analyses, accompanied by a commitment by USDOE-SRS to support such studies, before the technical adequacy of this document can be assured. To provide an example of the type of discussion that should be included in the CA, the SRS Ground Water Protection	

Com. No.	Comment	Action
	Management Program (GWPMP) document (WSRC-TR-96-0193), dated August 1996, provides a very brief discussion of one of the natural hydrologic barriers - the upward gradient. There is a discussion (Section 2.6) of the maintenance of natural head differences across the site, due to recognition of the value of the upward gradient in preventing downward migration of contaminants. There is a brief discussion of a long-standing site-wide policy of avoiding installation of high capacity production wells in areas where this natural upward gradient may be disturbed by pumping. The GWPMP indicates that this policy (put into effect in the 1980's) is still in effect, but there is no reference provided, nor is there any further detailed discussion of what actions this policy actually addresses. This entire issue is not discussed in the CA at all. There are no references to any section of the GWPMP. The CA should, at a minimum, investigate the specific provisions of this policy, discuss how well it has been implemented since its inception, and relate what is known about the process of maintenance of the upward gradient to the specific assumptions included in the conceptual model of the GSA that supports the analysis in the CA.	

INFORMATION MISSING FROM THE CA - COMPLETENESS CONCERNS

Com. No.	Comment	Action
90	The CA document dated September 1997 is not complete. The CA Review Team cannot reach a decision that ensures continued compliance with USDOE Order requirements at the two SRS disposal sites. In addition to further analyses and data collection described elsewhere in these comments, the analyses and data that are contained in this document are not complete. There are many statements that need to be better supported by references or by more complete analyses and explanations that clearly describe the analysts' logic.	No action required. The LFRG concluded that the CA provided sufficient information to support management decision for continued compliance with USDOE Order requirements at the two SRS disposal sites.
91	The document that the review team was asked to review is not complete. It basically presents statements of conclusion regarding the potential impact of the disposal facilities on the general public and on the environment, and statements that describe hydrologic conditions without adequate explanation, with few, or frequently no, reference to any detailed studies or other source documents, and with few new studies or analyses conducted to support the CA. The document, issued September 1997, contains some statements and conclusions that are unsupported (but not necessarily unsupportable) from a technical perspective. The document is incomplete because it does not enable the reviewer to understand the analyst's logic, or to reveal how the analysts used their data, their knowledge of the site, and their analytical tools to determine their results and to draw their conclusions. As a reviewer, I am left with the task of trying to piece together all of the technical work that was done on the CA to fully understand how the final results were derived. It became obvious to me during the initial site visit, when listening to presentations from various WSRC staff who had prepared the CA, that the technical work had been performed. The review team had numerous questions regarding the analyses described in the document, and most of these questions were answered satisfactorily by WSRC staff. It appears, though, that the information	No action required. The LFRG concluded that the CA provided sufficient information to support the management decision for continued operation of the SRS LLW disposal facilities.

Com. No.	Comment	Action
	presented was not documented. The document did not contain a complete and understandable description of what that work was, nor a mapping of the analysts' thought processes to allow the reviewer to trace the path from the basic data to the conclusions of the CA.	
92	Page 2-35, first sentence of the fifth paragraph states: "Concentrations of radioactive material at the mouths of the UTR and FMB will potentially include contributions from sources outside the GSA." However, the third sentence of this same paragraph states: "The composite analysis, however, has only considered the sources within the GSA because it is those sources that could influence decisions regarding operations of the LLW disposal facilities." The April 30, *1996 Guidance for a Composite Analysis of Interacting Source Terms* and the November 1, 1996 *Interim Review Process and Criteria for Composite Analysis* both indicate that the purpose of a composite analysis is to provide an analysis of the cumulative impacts of sources from LLW disposal facilities and all other sources that may interact with the LLW disposal facilities and contribute to the dose to a hypothetical future member of the public. It would appear that all source terms having the potential to interact at or before the point of assessment, must be considered and included in the composite analysis. This would be necessary to provide for a reasonably conservative estimate of the cumulative impacts of those source terms and their affects to the dose to future members of the public.	See the response to Condition 1.
93	The flow and transport models, as well as the conceptual model, of the ground water system at the GSA and the interrelationship of ground water and surface water needs further validation. Performing a water balance analysis of the GSA is one aspect of the needed validation. Designing and implementing an on-going monitoring strategy that will also function as a surveillance monitoring system is also needed	Comment noted. This will be addressed as R & D during the course of CA maintenance. See the attached maintenance plan.

Com. No.	Comment	Action
	for model validation.	
94	Sensitivity analysis (Section 6.0) is inadequate and needs to be rewritten. At a minimum, this section needs to be rewritten to account for the additional data provided by WSRC in the 4/21/98 memo from Bill Noll to Jeff Perry, and needs to consider the analysis of the effect of on-going remediation on the flow system, provided in "Impact of F- and H-Area Pump-Treat-Reinject Remediation Systems on the Old Radioactive Waste Burial Ground" (SRT-EST-98-154). Also, estimates of greatest uncertainty are needed to. provide direction and priorities for a CA maintenance program.	See the response to Condition 2.
95	The Savannah River CA, Section 6.3, Page 6-3; The first paragraph states "Plans for future use of the SRS (Appendix A) propose that release of the site to the general public for unrestricted use will not occur over the time period of this analysis." Appendix A; "Savannah River Site Future Use Project Report," is cited as the decision basis for future activities at the Savannah River Site. This project report does not reference or contain commitments made by the Department of Energy to its stakeholders regarding the future of the site. Composite analyses are conducted to demonstrate that management of all radioactive source terms; (past, present, and future) will not reasonably result in exceeding the dose limits set forth in USDOE Order 5400.5. Therefore, it would be prudent for the composite analysis to address all pertinent RODs, and other agreements made to the SRS stakeholders by the Department of Energy. No uncertainty analysis has been performed.	The SRS Future Use Plan has been transmitted to USDOE-HQ. This plan will be used as Appendix A in future CA revisions.
96	It is apparent that all of the potential interacting source terms have not been included in the analysis. The supplemental information provides a scoping analysis of the A and M-Areas, SRTC, and the SRL Seepage basins and their impacts on the UTR. It is not apparent from the document that B-Area, C-Area, D-Area, N-Area, or R-Area will not impact the analysis. The CA needs to include a comprehensive look at the SRS and specify	See the responses to Conditions 1 and 3.

Com. No.	Comment	Action
	what will and what will not impact the LLW disposal facility and provide justification for these exclusions.	
97	The Industrial Wastewater Closure Plan for F- and H-Area High-Level Waste Tank Systems needs to be incorporated into the CA. The stated CA requirement that most of the tanks be emptied with only 100 gallons of residual material is a requirement that must be communicated with the HLW Tank Closure project.	HLW Tank personnel are familiar with CA program. Updates in tank closure program will be reflected in CA maintenance activities. See the attached maintenance plan.
98	It is imperative that a good map of the SRS and GSA with all SRS facilities located on it be provided in the CA. It is difficult to understand the relative locations of the sources and LLW facilities with descriptive information only.	Comment noted and will be implemented in next revision of CA.
99	There is no discussion of the infiltration rates used in the analysis.	A table giving the infiltration rates used will be provided in the next CA revision.
100	There is no discussion of the corrosion rates used for the various waste forms. While leach rates are given for the concrete in the supplemental information provided, it is unknown whether the concrete is being considered to last for the entire 1000 year time of compliance. While the EAV and the Saltstone PAs provide justification for this assumption, the other concrete waste forms (i.e. the HLW tanks) have not been shown to meet this criteria. No corrosion data is given nor are the assumptions stated for the corrosion rates for the NR activated metals. Given the lack of information on this topic, the team is unable to assess whether the assumptions used are conservative or reasonable.	See the response to Condition 4.
101	The possible CERCLA and RCRA actions are included in the CA. There is no evidence provided that the representation of the possible future CERCLA actions is conservative, justified or supported by referenced documentation. Some of the representations of CERCLA actions presumed the outcome of the CERCLA process while other future CERCLA actions were not discussed. The site visit underscored the changing climate of RCRA and CERCLA actions at SRS, including the concept that	As CERCLA and RCRA actions are planned and completed they will be more accurately represented in CA revisions. See the attached CA maintenance plan.

Com. No.	Comment	Action
	RCRA actions being performed now will need to be addressed by CERCLA at some future point in time.	
102	In section 4.1.2 for building 235-F, it is stated that the residual radionuclide inventory was provided by Mr. Ray Lux. The reference document for this information is simply an E-mail message giving the source term. This does not adequately specify the source of the characterization information. It appears that the source term information came from the SAR for building 235-F. It is important (as a minimum in the reference documents) to state where the characterization information was obtained, to provide an indication of the accuracy of the information, and what assumptions were used.	See the response to Condition 4.
103	The effects of the ER cap (infiltration rates, impact on the ground water model) on the Old Burial Grounds is not given in the CA. While most of this information has been provided in the supplemental information provided, it needs to be incorporated into the CA.	See the response to Condition 2.
104	Incomplete Explanation of the Interrelationship of Ground Water Units and the Three Streams at the GSA - It appears that the full explanation of the relationship between the Upper Three Runs aquifer and the three surface water streams (Upper Three Runs, Four Mile Branch, and Tim's Branch) is not included in the CA document. It also appears that references to studies and documentation are not provided. The CA should, at a minimum, contain concise, but complete, explanations of critical hydrogeologic conditions. It is clear that the direction of ground water flow and the complex relationship of aquifers at various depths and locations throughout the GSA with surface water units, influenced by confining units of various thickness and continuity, are major determinants of contaminant levels and doses projected in the hydrologic modeling analyses, and that the existence of the natural hydrologic barriers (including the ground water divide and the incision of the upper aquifer by the three streams) is highly dependent upon flow conditions presented in this document. To provide SRS management	Comment noted. The next CA revision will attempt to provide a clearer description of the complex hydrologic Conditions at SRS.

Com. No.	Comment	Action
	with an analysis that supports proper disposal site operations for the long-term, more complete documentation and references are needed. The following are specific examples of the lack of complete explanation or the lack of adequate references that appears to exist throughout the document: a. Section 2.3.5 (Page 2-21) Ground Water Hydrology. There should be references to studies and discussion of their results to better substantiate the observation that the upward gradient in the Crouch Branch aquifer encompasses most of the GSA, and the basis for establishing the Crouch Branch confining unit as effectively preventing downward migration of contaminants into the Crouch Branch and lower aquifers. These hydrogeologic phenomena are cited as natural hydrologic barriers which protect lower aquifers from contamination. No references or detailed discussion of the technical data that is currently available to support these observations is included in this section. b. Section 2.3.5.2 (Page 2-25). The second paragraph refers to information on flow direction in the Gordon Aquifer being presented in Section 5.1.1. There is no Section 5.1.1 in the document. Section 5.1 (Hydrologic Model) presents a series of figures that contain hydraulic head data (modeled and measured) for purposes of demonstrating the relative agreement between model results and measurements. Section 5.1 refers back to Section 2.3.5.2 for discussion of ground water discharge to the three streams in the GSA. The only discussion in Section 2.3.5.2 is a very brief paragraph on Page 2-27, which merely states that the ground water discharges to these three streams, that the influence of these streams causes a ground water divide, and that the streams provide a natural flow boundary. None of these statements are referenced to a source of technical data, nor is there any further explanation of the technical, hydrogeologic basis for these conclusions. c. Section 5.1 (Page 5-4). In the second full paragraph (beginning "Hydraulic head measurements..."), there are numerous	

Com. No.	Comment	Action
	statements that are not referenced nor fully explained. This entire section is very crucial to understanding the conceptual model of the GSA and to quantifying the relationship of ground water units to surface water streams and the resulting modeling of contaminant transport. There should be a more complete discussion of the technical bases for these observations, there should be references provided, and there should be explanations of assumed boundary conditions and how they were quantified in the flow model. References in Section 5.1 to discussions in Section 2.3.5.2, as noted above, is an example of cross-referencing in this document to another equally incomplete discussion, rather than to a full discussion or to another referenceable source. d. Section 5.1 (Page 5-26). In the first full paragraph, the statement is made that "The hydrologic model was used to generate an average flow field for the GSA." This predicted flow field data - which is crucial to the accurate prediction of the movement of radionuclides in the subsurface and their control by natural hydrologic barriers - should be verified by performing a water balance analysis in the GSA. Using the conceptual model, water inputs to the Gordon and the Upper Three Runs aquifers and discharges to the three streams should be developed based on existing data on precipitation, subsurface flow and storage, withdrawals and reinjections (i.e., pump and treat at F & H Areas), and water table elevation measurements. Such a water balance would provide more credibility to the reliance on natural hydrologic barriers, if based on actual data accumulated over a sufficient period of time. The details of the data collection needed, and the development of the water balance are appropriate matters to determine in the context of the CA, and performed during CA maintenance.	

INTERPRETATION OF RESULTS/CONCLUSIONS

Com. No.	Comment	Action
105	Section 7.4 - It is silly to state that the only change that might increase the dose is a change in the land use. Obviously that could be a very big one but there are numerous others including the inventory that could increase the dose. This section really should list the assumptions and bases that are critical to the analyses and which are going to be compared during the periodic reviews.	See the response to Condition 4.
106	The calculated results do not clearly satisfy this requirement. The hydrology model does not provide convincing evidence that the regional aquifer system is well represented to the west of the GSA. For the individual PAs, this particular concern is not as relevant as the CA, where the potential release of contaminated groundwater to the soils and swamps near the Savannah River could introduce additional pathways for exposure. As discussed in the site visit, there was no data or verification step to ensure that mass was conserved in the hydrology model beyond the observation that the theory of the model supported the conservation of mass. The graphical results of the hydrology suggested that mass may not be conserved within the domain considered by the model. Additional graphical results indicated the zones of concern within the domain were associated with areas of low velocity. While the concern is less important, the additional results do not clearly indicate that mass is being conserved within the model domain. The importance of the groundwater divide is discussed in the CA and was discussed during the site visit. The movement of the water table was suggested to be +/- 5 feet from episodic events and the groundwater data suggested the divide did not shift that much from episodic events. Considering the significance of the groundwater divide in the transport of contamination, the low velocities of water near the divide, the concern over the conservation of mass, and the potential movement of the divide, the sensitivity analysis of the results should include the consideration of changes in the location of the	See the response to Condition 2. Future revisions of the CA will have a more detailed interpretation section.

Com. No.	Comment	Action
	groundwater divide. The results of this analysis should be addressed as an important consideration in the interpretation of results. The relationship between Fig. 4.4-11 and 5.2-15 is less than clear. The steady release of ^{99}Tc from the old burial ground in Fig 4.4-11 is not clearly represented in Fig. 5.2-15. In addition, the notion of a steady state release from the old burial grounds is questionable. The justification for a release of ^{233}U and ^{238}U from the old burial ground without a corresponding release of ^{234}U from the old burial ground is questionable, as shown in Figs. 4.4-12, 4.4-13, and 4.4-14. At the site visit, the long delay in the transport of "129 was attributed to the vadose zone thickness of 60 ft. This does not seem justified by other radionuclides with similar mobilities and other sources of the same radionuclide that do not have the similar sort of delay. Something is seriously wrong with Table 6.1-1. Figures 5.5-2 and 5.5-3 identify the dose from drinking water for FMB and UTR for ^{14}C and ^{3}H. The doses from these figures are not consistent with the table. The dose for one radionuclide could increase, as it has for ^{3}H, but the dose cannot decline for the other radionuclide. Perhaps there is an explanation, but none is provided.	
107	The CA provides an interpretation of the calculated results and the sensitivity and uncertainty results with respect to the dose constraint and the dose limit at the point of assessment and time period of assessment. The results are less than the dose constraint for all of the cases considered. As noted in other comments with respect to the CA, the logic, correctness, and rigor associated with these interpretations is not clearly presented or justified.	See the responses to Conditions 1 and 4.
108	The results of the CA indicate the maximum dose is 14 mrem/year, which is less than the dose constraint of 30 mrem/year. Consequently, and options analysis is not required and is not included in the CA. The	Comment noted.

Com. No.	Comment	Action
	dose of 14 mrem/year is the dose from the consumption of drinking water from FMB. This potential scenario is considered to be a sensitivity case and not a base case.	
109	The need for an ALARA assessment is presented in the CA for the results included in the CA. The presentation in the CA demonstrates there is no need for an ALARA assessment to identify any actions to further reduce the doses. Presuming the results of the analysis provide a complete, composite analysis of the SRS, this conclusion is justified.	Comment noted.
110	The CA does not provide a comparison to the PA to allow an evaluation of this requirement. The CA does not admit a resident scenario and the drinking water calculations in the CA are performed at a larger distance from the source than in the PA.	Results presented in Table 4.4-5 for the disposal units in the EAV and Saltstone facilities provide the comparison. Future revisions of the CA will provide a more explicit comparison.
111	The maximum projected dose over the period of assessment is presented, but without a clear and consistent definition of the point of assessment.	See the response to Condition 1.
112	The need for the ALARA assessment is presented and concludes an ALARA assessment is not warranted. The calculated population dose is 3 person-rem/year, allowing a cost of $30,000 per person-rem averted. The CA concludes the analysis of the options in the CA exceeds this maximum value.	Comment noted.
113	An options analysis was not performed for the CA because the resulting dose reported in the CA was less than the dose constraint.	Comment noted.
114	This particular requirement is associated with the rigor of the analysis presented in the CA. Numerous sources have been excluded without justification and the point of assessment is not well justified. The analysis does not provide bounding calculations for the many uncertain variables associated with the CA. As a result, the CA does not provide a clear case that the analysis is a reasonable representation of the existing site knowledge.	See the response to Condition 1.
115	Section 1.0 I don't believe the results of the CA clearly show there will be NO adverse health impact. The numbers presented are indeed less than the dose constraints and performance objectives but they are based on	Comment noted.

Com. No.	Comment	Action
	a less than robust or complete analysis. How this section will need to be reworded will be based on the resolution of the comments.	
116	The composite analysis does not include discussion or evaluation of potential off-site sources such as the Barnwell low level waste disposal facility, or a commercial nuclear facility located up river from the Savannah River Site.	Such discussion is not required if the dose constraint is met as is the case in the CA.
117	SRS CA Requirement; Page 7-2, section 7.4 first paragraph states "The maximum peak dose of 14 mrem/yr calculated for the GSA in this analysis is considerably lower..." The above referenced paragraph is inconsistent with the Supplemental information provided in "Bounding Estimate of All GSA Contaminants Migrating to Either of the Streams." This analysis shows at an estimated dose of ~30.8 mrem/year which is over the dose constraint of the CA.	Bounding (worst case) estimates are not appropriate for determining compliance with the CA dose constraint.
118	a. The ground water divide is a critical hydrologic factor in any analysis of the potential future impact on the environment of low-level waste disposal at the GSA. b. More careful, detailed analyses of the estimated impacts on drinking water and recreational exposures should be performed to better define the sensitivity of the CA doses to changes in this and other critical hydrologic factors. Such analyses should include estimated doses through the drinking water pathway at the mouth of the Upper Three Runs and Four Mile Branch streams, as well as at the Highway 301 Bridge. c. Studies designed to measure and quantify hydrogeologic factors, as well as the influence of site activities at the surface, should be designed and conducted to further quantify the hydrologic response of the ground water divide (as well as the other natural hydrologic barriers). Modeling studies are a first step, but longer term monitoring and aquifer stress tests are needed to quantify the likely response of the flow system to future conditions, all of which may impact the dimensions, as well as the existence and the effectiveness, of the ground water divide. d. Although the sensitivity analysis indicates that estimated doses are highest for tritium, there are other radionuclides with longer half-lives, that may be of greater concern. There should be a more detailed analysis of	See the responses to Conditions 1 and 2.

Com. No.	Comment	Action
	the potential impact of other "significant" radionuclides which consider both the drinking water and the recreational scenarios at the GSA, at the mouths of both streams, and at the Highway 301 Bridge. If a more thorough analysis indicates that potential doses reach or exceed 30 mrem/year, there will be the need for an options analysis for examining means for reducing potential doses further, by applying ALARA.	
119	The CA presents conclusions that the long-term performance of the disposal facility and other contributing sources is less than the dose constraint. The demonstration of these conclusions is the source of many comments included in this review. The logic correctness, and rigor of the conclusions reached in the CA warrant additional review prior to acceptance.	The conditions of approval given by LFRG have been met by publication o f this addendum.
120	The CA results are less than 30 mrem/year, the need for an ALARA assessment is presented, and the results show an ALARA assessment is not required. However, the need for preparing an options analysis is concluded using the results from the sensitivity analysis of the consumption of water from FMB, and not from the base case in CA that did not include the consumption of surface water. At this particular point of the CA, the conclusions are being drawn from the wrong results. This further underscores the many difficulties with the identification of the point of assessment throughout the CA.	See the response to Condition 1.
121	This requirement does not currently apply to the SRS CA.	Comment noted.
122	This requirement does not currently apply to the SRS CA.	Comment noted.
123	Section 7.3 of the CA concludes that potential doses are unlikely to exceed the dose constraint. Given the uncertainties in the conceptual and numerical groundwater flow models, it is not unreasonable to postulate conditions that would result in exceedance of the dose constraint. Acceptance of the CA should be conditional upon completion of a more thorough uncertainty analysis and any options analysis that may be required based upon those results.	See the response to Condition 2.

Com. No.	Comment	Action
124	In the Summary and Conclusions, Page 1-1, the statement is made in the first paragraph that the results of the CA clearly indicate that continued disposal will have no adverse impact on future members of the public. This conclusion is highly dependent upon the assumption that institutional control will effectively prevent human exposure to radiological contaminants and will prevent human activities that may disrupt the flow system characteristics that provide natural hydrologic barriers. It is misleading to state that the CA results are based on dose calculations that not only justify the statement that no adverse impact would occur, but justify not performing additional sensitivity analyses or options/ALARA analyses to reduce doses further. It is critical that this document state that the conclusions of the CA are based on the recommendations included in a future land use plan. No one. can predict the future, and even though many of us believe that the SRS, as it exists today, will continue to remain a restricted federal defense facility for a very long. time, there is a need for some assurances regarding maintaining the site's status. (Order USDOE 5400.5 requirements must be met before the site can be released, but there is no discussion of how or whether this requirement will be met, or what is in place to assure that the site will not be released.) Absent any other legally binding commitment to, maintaining restricted use of the existing site for a specific period of time or "in perpetuity", it is necessary to qualify all conclusions by stating the overall assumptions upon which they were based. To provide an illustration of the need for consistent use of qualifying statements when providing conclusions on the CA results, the additional information provided by WSRC in the 4/21/98 memo from W. Noll contains a re-analysis of the potential doses calculated by challenging the assumption that the ground water divide location will remain unchanged for the entire period of the analysis. The re-analysis indicates that the estimated dose	Comment noted.

Com. No.	Comment	Action
	from the drinking water pathway at Four Mile Branch at the GSA for tritium is 64 mrem/year, which is 16 times greater than the MCL. However, WSRC concludes that this level of exposure would never occur because overly conservative assumptions were used (all contaminants migrate to one stream rather than being partitioned to two streams due to the ground water divide, and no correction was made for the added decay of tritium in a longer migration pathway) and the calculated peak dose would occur at 62 years, which is well within the time period where exposure would be prevented by institutional controls, according to future land use plans. In this case, the results of the analysis exceed the MCL and the 30 mrem/year point where an options analysis would be needed. So the analysts provide qualifying statements that acknowledge the implications of the assumptions that were used. The same type of qualifications are needed when drawing conclusions that there will be no adverse impacts on the general public in the future.	
125	SRS CA Requirement, Supplemental "Assessment of Impact of A and M Area Sources on Composite Analysis Results." The sixth paragraph states "For each radionuclide, the concentration in Upper Three Runs from the GSA sources (i.e. that analyzed in the CA) is greater than that from the Tims Branch sources. The ratio of concentration the UTR to that in the Tims Branch ranges from 29 for ^{238}U to 29 million for tritium. Thus the Tims Branch watershed will make a negligible contribution of potential doses to the public calculated at the mouth of Upper Three Runs." Internal radiation exposure from multiple radionuclides is a cumulative effect not a singular event. All radionuclide sources and their respective dose contributions to the off-site receptor should be calculated and summed to determine if the off-site dose criteria has been met.	Comment noted.
126	SRS CA Requirement, Supplemental "Bounding Estimate of All GSA Contaminants Migrating to Either of the	See the response to Condition 3.

Com. No.	Comment	Action
	Streams." The included table *(no table number assigned)* under the column; "Estimated Dose form Recreation Scenario at FMB Mouth" indicates a current dose from C^{14} of 28.8 mrem/year. The indicated table does not include the dose contribution from the A and M areas, and it should be noted that the indicated dose of 28.8 mrem/yr is close to the 30 mrem/yr dose criteria for the CA. It should also be noted that the cumulative estimated dose at the mouth of FMB is ~30.8 mrem/yr. It is imperative that the CA source term be reevaluated to include the estimated dose from all radionuclides and that the effect on the down stream receptor site be determined. Additionally, there is no mention in the CA as to how future development on the opposite bank of the Savannah River will be guided.	
127	The conclusions of the CA are based on a limited interpretation of the results and the bases for the analysis presented in the CA. Since the results indicated that potential doses were less than the dose constraint, as long as access to the SRS was restricted in perpetuity, and that conservative assumptions were selected in preparing the CA, there was no apparent need to conduct a detailed examination of the assumptions in the CA and their effect on the results.	Comment noted.

Notes:
Acronyms are generally not spelled out in the table due to space limitations. The Comment column in the table may contain acronyms that are spelled out since this column represents direct quotations from the Comment document. The following acronyms are used in the table.

ALARA	As Low As Reasonably Achievable	LAW	Low-Activity Waste
CA	Composite Analysis	LFRG	Low-Level Waste Facilities Federal Review Group
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act	LLW	Low-Level Waste
D&D	Decontamination and Demolition	MCL	Maximum Contaminant Level
USDOE	U.S. Department of Energy	NRC	U.S. Nuclear Regulatory Commission
DQO	Data Quality Objectives	OBG	Old Burial Ground
DWPF	Defense Waste Processing Facility	PA	Performance Assessment
EAV	E-Area Vaults	RCRA	Resource Conservation and Recovery Act
EPA	U.S. Environmental Protection Agency		
FMB	Fourmile Branch	ROD	Record of Decision
GSA	General Separations Area	SRL	Savannah River Laboratory
HLW	High-Level Waste	SRS	Savannah River Site
HQ	Headquarters	SRTC	Savannah River Technology Center
ILT	Intermediate-Level Trench	UTR	Upper Three Runs
		WSRC	Westinghouse Savannah River Company

6.0 Condition 6

Discussion of the environmental monitoring program, inclusion of environmental monitoring data, and comparison with the expected results from the composite analysis.

6.1 Composite Analysis Monitoring Program

Per the requirements in the DAS issued for the E-Area LLWF and the SDF, a monitoring plan shall be written, approved, and implemented within 1 year of issuance of the DAS (9/28/00) and updated at least every 5 years.

An initial monitoring plan was developed in response to this requirement and issued July 31, 2000. The monitoring plan was designed to meet the requirements for monitoring LLW disposal facilities according to USDOE Order 435.1 and its associated implementation guidance with regard to actual performance versus projected performance based on the CA.

The CA monitoring plan uses existing monitoring currently performed via other monitoring programs whenever possible because it is not the intent of the plan to duplicate efforts undertaken elsewhere and/or to fulfill other requirements. Existing monitoring programs have been reviewed and are referenced in the CA monitoring plan as appropriate. Currently, all monitoring proposed in the monitoring plan is performed under existing programs/permits. No new or additional monitoring is proposed.

The monitoring plan also includes annual data review and evaluation intended to provide a means for ongoing assessment of the monitoring and modification of the monitoring plan as necessary in response to data evaluation. Results and recommendations from data evaluation will be reported and distributed in the annual review conducted through the PA/CA maintenance program.

The following subsections provide a summary description of the general monitoring programs from which the CA monitoring plan utilizes sampling data. A complete description of the CA monitoring program is provided in the most current CA monitoring plan.

6.1.1 Environmental Monitoring Program

SRS looks for, identifies, and quantifies its released contaminants through an extensive environmental monitoring program. This program's main components are effluent monitoring and environmental surveillance. Samples of air, water, and other media are collected and analyzed to determine the presence of contaminants from site operations. Results are used to monitor effects on natural resources and human health and also to demonstrate compliance with regulations. These results are published each year in the SRS Environmental Report which is made available to the public.

Much of the onsite monitoring is done by the Environmental Protection Department's Environmental Monitoring Section and by the Savannah River Technology Center. Groups outside the SRS also monitor the site. These include the South Carolina Department of Health and Environmental Control and the Georgia Department of Natural Resources.

6.1.2 Effluent Monitoring

Effluent monitoring is the collection of samples at the point where materials are released from the facilities and their subsequent analysis. Two types of effluent monitoring are done at SRS.

Radiological effluent monitoring looks for radionuclides that are released from the facilities. More than 4,400 radiological samples were collected and analyzed during 1996. Nonradiological effluent monitoring looks for nonradioactive materials that are released from the facility.

6.1.3 Environmental Surveillance

Environmental surveillance covers more that 31,000 square miles and extends up to 100 miles from the site. With results of this surveillance, scientists attempt to assess contaminants that may have spread into the environment. Like effluent monitoring, environmental surveillance can be both radiological and nonradiological.

6.1.4 Radiological Releases

Radionuclides released from the site can travel through the environment, potentially causing exposure to the offsite public. Routes that contaminants may follow through the environment are called pathways. Airborne release pathways include (1) inhalation and (2) the consumption of locally produced foods and milk, contaminated by deposition of the airborne contaminants; liquid release pathways include the consumption of (1) fish, (2) shellfish from downriver in the Savannah River estuary, and (3) Savannah River water. Monitoring groundwater migration from contaminated areas on the site is important in determining liquid releases.

6.1.5 Radiological Surveillance

Routine surveillance is performed on the atmosphere (air and rainwater), surface water (site streams and the Savannah River), drinking water, food products (terrestrial and aquatic), wildlife, soil, sediment, vegetation, and groundwater. Monitoring of gamma radiation in the environment is conducted on site, at the site boundary, and in surrounding communities.

6.2 Comparison of Environmental Monitoring Data with Composite Analysis Results

Data from the last two annual monitoring reports are compared with CA results in Table 6.2-1. The monitoring reports give annual average radionuclide concentrations in SRS streams. These concentrations were used to calculate radiological dose by assuming consumption of 2 liters of stream or river water per day for a year. These doses are presented along with the doses calculated in the CA as a "reality check" on the CA results. The numbers are in good agreement, with those for the Savannah River being closest and those for UTR being farthest apart.

Results and recommendations from data evaluation will be reported and distributed in the annual review conducted through the PA/CA maintenance program.

6.3 Probabilistic Uncertainty Analysis

WSRC has begun a program to develop a methodology for performing probabilistic uncertainty analyses. As this program matures, the understanding of the disposal systems and the environment will increase and monitoring locations, equipment, and analytes can be modified as needed to reflect the improved knowledge.

Table 6.2-1 Monitoring Data and Composite Analysis Results Comparison

Stream	From 1996 Monitoring (mrem/yr)	From 1997 Monitoring (mrem/yr)	From Composite Analysis (mrem/yr)
Upper Three Runs	0.11[a]	0.15[c]	2.4[e]
Fourmile Branch	9.7[a]	9.9[c]	24.[e]
Savannah River	0.05[b]	0.05[d]	0.08[f]

Notes:

[a] Based on concentration given in Savannah River Site Environmental Report for 1996, WSRC-TR-97-0171, Table 6-4, page 83.

[b] Based on concentration given in Savannah River Site Environmental Report for 1996, WSRC-TR-97-0171, Table 6-5, page 85.

[c] Based on concentration given in Savannah River Site Environmental Report for 1997, WSRC-TR-97-00322, Table 6-3, page 91.

[d] Based on concentration given in Savannah River Site Environmental Report for 1996, WSRC-TR-97-00322, Table 6-4, page 94.

[e] From Composite Analysis, WSRC RP-97-311, Table 6.1-1, page 6-2.

[f] From Composite Analysis, WSRC RP-97-311, Table 5.5-2, page 5-73.

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7.0 Condition 7

Inclusion of the information that Savannah River Site committed to be incorporated in the composite analysis maintenance plan over the course of the composite analysis review.

During preparation of this addendum, the authors discussed this condition with J. N. Perry, the Review Team leader. Mr. Perry indicated his understanding that the commitments had been captured in the minutes of the Review Team meetings. The authors then discussed the Review Team minutes with R. U. Curl. The LFRG Technical Secretary. Mr. Curl indicated that no commitments for incorporation of information in the SRS Maintenance Plan are noted in the review team minutes.

The authors believe that all of the items discussed with the Review Team regarding what would be in the SRS Maintenance Plan have, in fact, been incorporated into the plan (Attachment 1).

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Attachment 1

Performance Assessment and Composite Analysis Maintenance Program

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Attachment 2

Excerpt from Long Range Comprehensive Plan

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