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PRELIMINARY SAFETY EVALUATION

FOR

241-C-106 WASTE RETRIEVAL

PROJECT W-320

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**PRELIMINARY SAFETY EVALUATION, 241-C-106 WASTE
RETRIEVAL SLUICING SYSTEM, PROJECT W-320**

1.0 INTRODUCTION AND SUMMARY

This document provides the preliminary safety evaluation (PSE) for Project W-320, Tank 241-C-106 (C-106) Waste Retrieval Sluicing System (WRSS). *The National Defense Authorization Act for Fiscal Year 1991*, Public Law 101-510, Section 3137 (Wyden Amendment), "Safety Measures for Waste Tanks at Hanford Nuclear Reservation," mandates that the U.S. Department of Energy (DOE) develop plans for response to safety issues associated with the waste storage tanks at the Hanford Site, and to report the progress of implementing those plans to Congress. The *Plan to Implement Remediation of Waste Tank Safety Issues at the Hanford Site*, WHC-SP-0697 (Wilson and Reep 1991), has identified C-106 as the fourth Priority 1 safety issue.

The *Hanford Federal Facility Agreement and Consent Order* (Ecology et al. 1994), also known as the Tri-Party Agreement, includes milestones to demonstrate single-shell tank (SST) retrieval. Milestone M-45-03, "Initiate full-scale demonstration of waste retrieval technology," by October 1997, requires a demonstration of waste retrieval from Hanford Site SSTs as the initial phase of SST closure. Tank C-106 has been identified as the M-45-03 retrieval demonstration tank by the DOE. The Washington State Department of Ecology has concurred in this selection.

1.1 BACKGROUND

Tank C-106 is a 2,000,000 L (530,000-gal) capacity SST. It is located in the C Tank Farm at the 200 East Area on the Hanford Site. This tank has been used for radioactive waste storage since 1947.

Between mid-1963 and mid-1969, C-106 received about 500,000 L (132,000 gal) of high-heat waste, including neutralized plutonium-uranium extraction high-level waste and strontium-bearing solids from the strontium and cesium recovery program. In 1971, temperatures exceeding 99 °C (210 °F) were measured in C-106. As a result of these high temperatures, a ventilation system was installed to cool C-106. About 23,000 L (6,000 gal) of cooling water are added to C-106 each month to prevent the sludge from drying out and C-106 from overheating. The current radiogenic heat rate has been calculated at 116 MJ/h (110,000 Btu/h) (Bander 1993a and 1993b). Tank C-106 was withdrawn from active service in 1979 and is categorized as sound (i.e., not known to be leaking).

If the 23,000 L/month (6,000 gal/month) cooling water additions are stopped, waste in C-106 will dry out, sludge temperatures will rise exceeding established limits, and structural damage to the tank may result. Should C-106 start to leak, continuation of the water additions required to cool the waste may increase the amount of leakage to the soil column.

An analysis of core samples from C-106 shows that more than 90% of the heat is produced by strontium-90 (^{90}Sr). For this reason, the heat generation rate of the sludge is expected to decline with time in correspondence to the 28.1 year half-life of ^{90}Sr .

The waste contained in C-106 is reported (Hanlon 1994) to consist of 746,000 L (197,000 gal) of sludge. This waste is stratified into two layers. The top layer consists of 655,000 L (173,000 gal) of sludge containing a sufficient amount of ^{90}Sr to generate excessive heat (high-heat layer). The bottom layer consists of 91,000 L (24,000 gal) of hardened material which, based on tank fill records, is a coating waste from dissolution of aluminum fuel cladding. Of the 182,000 L (48,000 gal) of liquids reported for C-106 (Sederburg 1994), 61,000 L (16,000 gal) are interstitial and 121,000 L (32,000 gal) are supernatant.

A water cover is maintained over the solids to facilitate evaporative cooling and to keep the sludge from drying out. During the last 10 years, about 3,000,000 L (800,000 gal) of water have been added to C-106 for cooling purposes. Evaporation of the cooling water additions has resulted in deposition of a mineral layer as explained in WHC-SD-RE-TI-205, *Data Transmittal Package for 241-C-106, Waste Tank Characterization* (Weiss 1989). The mineral layer correlates to surface crust noted during core sampling in 1986. The postulated safety hazard for the C-106 high-heat tank involves a potential release of high-level nuclear waste caused by overheating and degradation/failure of the tank reinforced concrete structure.

An engineering study (Pauly and Torgerson 1987) examined several cooling options to support in-situ stabilization. That study could find no viable option for cooling C-106 contents if the sludge were allowed to dry completely. The study concluded that parallel development of cooling and retrieval alternatives should be pursued.

A subsequent report (Ferlan 1993) was commissioned to examine only the waste retrieval option. That report forms the basis for Project W-320, Tank 241-C-106 WRSS.

The chemical compatibility report (Sederburg 1994) evaluates the chemical compatibility of waste currently stored in C-106, AY-101, and AY-102. Specifically, this report describes the acceptability to transfer these wastes in the sequence required by the WRSS. Further, the chemical models presented in Appendix C of the chemical compatibility report demonstrate that no chemical reactions identified will result in future waste incompatibility. Thus, no chemical safety issues will be exacerbated in any way by retrieving the contents from C-106 and transferring them to AY-102. In fact, a safer overall tank farm configuration is expected.

The objectives of Project W-320 are to design, fabricate, develop, test, and operate a new retrieval system capable of removing a minimum of about 75% of the high-heat waste contained in C-106. Encased and shielded transfer lines will be used to transfer the high-heat waste to AY-102. It is anticipated that sluicing operations can remove enough waste from C-106 to reduce the remaining radiogenic heat load to 42 MJ/h (4 E+4 Btu/h) or less. Reducing the heat load to this level will resolve the high-heat safety issue as well as allow closure of the tank safety issue.

Without remediation, this SST will require cooling water additions until the year 2045 (Pauly and Torgerson 1987). Tank-to-tank sluicing, an existing technology, will provide the earliest possible closure of this safety issue. Removal of the heat-generating sludge allows cooling water additions to cease. Cessation of cooling water additions will allow operations to place C-106 in a safe, interim stabilized state. Additional retrieval operations can proceed using alternate technology, as required, to achieve a 99% waste retrieval according to terms of Tri-Party Agreement Milestone M-45-05 (Ecology et al. 1994). Final remediation will take place as part of site closure actions.

1.2 GENERAL DESCRIPTION

A summary of WRSS sluicing operations is provided in WHC-SD-W320-TI-002, *Project W-320 Tank 241-C-106 Sluicing Process Flowsheet* (Estey 1994). Figure 1-1 depicts the Hanford Site area map which includes the C-106 waste retrieval identified as project area. Figure 1-2 depicts the facility general arrangement of the Hanford Tank Farm Facilities for 200 East Area that includes tanks C-106 and AY-102. Schematic diagrams for WRSS and the C-106 HVAC system are shown in Figures 1-3 and 1-4 respectively.

Before starting the WRSS, the tank content level in AY-102 is monitored to ensure that it contains enough space for transfer of most of the solids from C-106 plus the anticipated liquid additions for line flushes, scrubber bleed streams, etc. The C-106 slurry pump will be flushed and will be valved to recirculate solution to the sluicers in C-106 until the slurry is at a pumpable concentration.

The system will be revalved and the variable-speed slurry pump in C-106 will be restarted for transfer of slurry from C-106 to AY-102. The sluice pump in AY-102 will be simultaneously started to transfer liquid from AY-102 to C-106.

Sluicer nozzle direction will be controlled from a control room. The intent is to create channels to direct liquid from the sluicer into the intake path of the slurry pump. Sluicing will be switched between two sluicers to maintain slurry flow and to achieve the highest possible solids recovery rate. An in-tank imaging system will enable operators to determine the liquid level, locate the waste, and aim the sluicer to retrieve the waste.

The sluicing technique will remove sludge from C-106 while minimizing the potential for leakage. As currently planned, waste solids located against the tank walls will not be sluiced until near the end of the sluicing operations. This will minimize the potential for the sluicing stream to cause a leak by impinging on a weak point in the C-106 tank wall or by opening a potentially preexisting corrosion or sludge-plugged leak site.

Figure 1-1. Hanford Site Area Map.

Figure 1-2. Facility General Arrangement.

Hanford Tank Farm Facilities

200 East

Note: All single-shell tanks were removed from service (not allowed to receive waste) on or before November 21, 1980

This Figure has been excerpted from WHC-EP-0182

Figure 1-3. WRSS Schematic Diagram.

(This figure has been excerpted from Drawing H-2-818536)

Figure 1-4. C-106 HVAC System Schematic Diagram.

(This figure has been excerpted from Drawing H-2-818470)

The C-106 ventilation system will perform the following functions: maintain a negative pressure; remove heat from the tank and dehumidify recirculating air; remove aerosols; exhaust a filtered ventilation stream to the atmosphere through an exhaust stack and monitor emissions; and provide an air inlet via a breather filter when the ventilation system is inoperative.

Current characterization data from C-106 indicate that the waste contains fissionable material. This fissionable material will be distributed as evenly as possible by a slurry distributor designed to maximize solids settling in AY-102.

1.3 HAZARDS CLASSIFICATION

The hazards classification for the WRSS has been evaluated in WHC-SD-WM-HC-007, *Facility Hazard Classification 241-C-106 Waste Retrieval* (Conner 1993). The focus of the hazards analysis is the risk associated with the energy sources and the material at risk during postulated accident events such as waste transfer line breaks and waste leaks from a jumper or connector. Based on a comparison of the calculated maximum dose consequences to the criteria in WHC-CM-4-46, *Nonreactor Facility Safety Analysis Manual*, the WRSS has been classified as a "high" category facility.

1.4 SUMMARY AND CONCLUSIONS

A preliminary hazards analysis was conducted for the systems that are being installed under this project to identify potential hazards for the WRSS. This analysis identified the potential worst-case accidents, which are discussed in Chapter 4.0 and in Appendix A of this document. The analysis in Appendix A is based on the radioactive inventory data presented in WHC-SD-WM-TI-543, *Radionuclide and Chemical Inventories for the Double Shell Tanks* (Van Vleet 1993a), and WHC-SD-WM-TI-565, *Radionuclide and Chemical Inventories for Single Shell Tanks*, (Van Vleet 1993b). As noted in Bander 1993a and 1993b, the radiogenic heat load is 32.2 Kw (110,000 Btu/h). The calculated heat load due to radioactive elements in C-106 based on data from Van Vleet 1993b is dominated by decay of ^{90}Sr and ^{137}Cs with negligible contribution from other radionuclides, and is equivalent to 13.4 kW; 12 kW from ^{90}Sr and 1.4 kW from ^{137}Cs . The radioactive inventory data in Van Vleet 1993b are suspect values according to Bander 1993a and 1993b. This introduces a scaling factor of 2.4 due to the discrepancy between radiogenic heat load in the Bander reports and the estimated heat load from the current radioactive inventory in C-106. Characterization of C-106 is scheduled to be completed in the near future. Until such time as the new characterization data for C-106 is available, the 2.4 scaling factor has to be applied to the estimated EDE values in Appendix A. According to the guidelines in WHC-CM-4-46, the EDE values for postulated accidents are used for determination of safety classification of WRSS and radiological risk acceptance. The risk is based on the frequency of occurrence of accidents and the consequence. The results and conclusions are as follows:

- One of the event hazardous conditions estimated to be worst case is an environmental unmitigated release to the atmosphere. This condition is the result of the waste retrieval transfer system continuing to operate with a break of the waste transfer supply and return lines caused by a seismic event (Scenario 1).

Scenario 1: Waste Transfer Line Break

The following postulated accident scenarios were evaluated as part of Scenario 1: Scenarios 1a and 1b--waste transfer supply line and return line breaks (2-hour release time), and Scenarios 1c and 1d--waste transfer supply line and return line breaks (10-second release time) taking credit for a seismic switch that shuts down transfer pumps.

Scenarios 1a and 1b:

The dose to the maximally exposed offsite individual has been calculated to be $8.5 \text{ E-1 mrem} \times 2.4 = 2.0 \text{ mrem}$, effective dose equivalent (EDE). The maximally exposed onsite individual has been calculated to receive a dose of $8.1 \text{ E-1 rem} \times 2.4 = 1.9 \text{ rem}$, EDE. Even though these values are within the radiological risk acceptance guidelines for EDE doses (Table 4-4 of this document and Section 7.0 of WHC-CM-4-46), an environmental hazard safety classification (EHSC) calculation revealed that the calculated EHSC value of $1.1\text{E}+6$ according to Criterion 10 of Table 1, Section 9.0, WHC-CM-4-46. A seismic switch will be installed to reduce the impact on the personnel safety and health hazards caused by dose consequences. This feature also reduces soil contamination by greatly reducing the large quantity of waste spill to the ground during a postulated break of transfer lines. See the following summary and conclusions for Scenarios 1c and 1d for the line breaks with a seismic shutoff feature.

Scenarios 1c and 1d

The calculated offsite EDE value for a break in the supply and return lines, Scenarios 1c and 1d, is $4.2 \text{ E-2 mrem} \times 2.4 = 1.0\text{E-1 mrem}$. The calculated onsite EDE value is $8.0 \text{ E-2 rem} \times 2.4 = 1.9\text{E-1 rem}$. An EHSC calculation was also performed with a resultant value of $2 \text{ E}+4$ which is far less than the value of $5 \text{ E}+5$ according to Criterion 13 of Table 1, Section 9.0, WHC-CM-4-46. The EDE values are within the radiological risk acceptance guidelines in Table 4-4 of this document, and in Section 7.0 of WHC-CM-4-46.

- Another accident consequence estimated to be worst case is an environmental release to the atmosphere. The release is the result of the waste retrieval transfer system continuing to operate with a breached or leaking jumper connector that sprays aerosol waste out of a pit. The results of this scenario are noted as follows.

Scenario 3: Spray Leak from Jumper or Connector

The following postulated accident scenarios were evaluated as part of Scenario 3: Scenario 3a--jumper or connector break inside the pit with cover block off, with a release time of 2 hours; Scenario 3b--jumper or connector break inside the pit with cover block on and all openings closed, with a release time of 8 hours; and Scenario 3c--jumper or connector break inside the pit with cover block on and 5.1-cm (2-in.) valve handle hole open, with a release time of 8 hours.

Scenario 3a:

As noted in this scenario, the dose to the maximally exposed offsite individual has been calculated to be $3.1 \text{ E}+3 \text{ mrem} \times 2.4 = 7.4\text{E}+3 \text{ mrem}$, EDE. The maximally exposed onsite individual has been calculated to receive a dose of $2.9 \text{ E}+3 \text{ rem} \times 2.4 = 7.0\text{E}+3 \text{ rem}$, EDE. The offsite and onsite calculated EDE doses exceed the radiological risk acceptance guidelines in Table 4-4 of this document and in Section 7.0 of WHC-CM-4-46.

The dose consequence to environmental release with a breached or leaking jumper connector that sprays aerosol waste out of an uncovered pit is significantly reduced with the cover block on, as analyzed in Section 4.2 for Scenarios 3b and 3c. According to Section 3.6.1 of WHC-SD-WM-OSR-005 (Dougherty 1993), pit covers are required to be installed and remain in place while performing the transfer operation. The radiological risks are within the acceptance guidelines of Table 4-4 with the cover block on.

- Concerns about tank C-106 temperature anomalies and potential steam releases are being investigated. Tank temperatures and waste levels will be closely monitored until the tank returns to a steady state condition.
- All other accidents with significant consequences (frequencies $>1 \text{ E}-6$) for WRSS will be analyzed during the preparation of the safety assessment (SA).
- Preliminary hazards analyses for existing systems noted in the AWF interim safety equipment list (ISEL) (Kidder 1993a) and the SST ISEL (Kidder 1993b) have been completed and form the basis for existing systems. The analyses results of these systems are included in the respective ISELs.

The WRSS uses existing AWF and SST facilities and similar tank farm facilities. The proposed design for the WRSS also uses buried and bermed piping, pits with cover blocks on, and seismic switch shutdown features. A review of the design basis accidents (DBA) and design basis earthquake (DBE) for the WRSS demonstrates that the WRSS can be implemented without exceeding applicable risk acceptance criteria.

2.0 DESIGN CRITERIA

This section discusses current safety-related design criteria and requirements used in the design and construction of the C-106 WRSS (Project W-320).

The WRSS will be designed and constructed to meet the requirements in DOE Orders 6430.1A, *General Design Requirements* (DOE 1989); 4700.1, *Project Management System* (DOE 1987); 4330.4A, *Maintenance Management Programs* (DOE 1990); 5480.21, *Unreviewed Safety Questions* (DOE 1991b); 5480.22, *Technical Safety Requirements* (DOE 1992b); and 5480.23, *Nuclear Safety Analysis Reports* (DOE 1992c).

The C-106 WRSS elements not fixed by the location of the transfer and receiving tank are placed in accordance with guidelines in DOE Order 6430.1A, *General Design Criteria* (DOE 1989) and DOE-RL Order 4320.2C, *Site Selection* (DOE-RL 1990a). Included in this section is a preliminary list of safety class systems and equipment. Design information is taken from WHC-SD-W320-FDC-001, *Functional Design Criteria for Tank 241-C-106 Waste Retrieval*, Project W-320 (Bailey 1994).

Design and construction of the WRSS (a nonreactor nuclear facility) will comply with the following codes and standards:

- American Concrete Institute (ACI)
- American National Standards Institute (ANSI)
- American Society of Mechanical Engineers (ASME)
- American Water Works Association (AWWA)
- Hanford Plant Standards (HPS)
- International Conference of Building Officials (ICBO)
- Institute of Electrical and Electronics Engineers (IEEE)
- Instrument Society of America (ISA)
- National Fire Protection Association (NFPA).

All applicable DOE Orders and standards in effect at the time design is started will be used. The project will be in compliance with all applicable federal, state, and local laws and regulations. The latest edition of all industry codes, standards, and manuals in effect, or authorized by the DOE, will be used.

The design for material confinement is based on current, accepted methods of construction for hazardous waste transfer systems and ventilation systems. These accepted and applied designs have proven their ability to confine the hazardous material in normal, abnormal, and accident conditions.

The WRSS is capable of mobilizing and transferring the sludge from C-106 to AY-102. The receiver tank system modifications or upgrades necessary to support retrieval of C-106 waste are included in this project.

The sluicing systems will be of modular design to the greatest extent practical. This design will incorporate features that facilitate and minimize

fabrication, installation, removal, and disposal within the tank farm area. Modifications installed by Project W-320 will not significantly add to the difficulty of, or preclude accomplishing, post-retrieval of C-106.

2.1 SAFETY CLASS SYSTEMS

A preliminary assessment of the major systems and structures that constitute the WRSS was conducted using available data and facility design information from WHC-SD-W320-FDC-001 (Bailey 1994) to identify safety class systems. The results of this analysis are shown in Table 2-1.

Table 1 of WHC-CM-1-3, *Management Requirements and Procedures*, MRP 5.46, "Safety Classification of Systems, Components and Structures," provides the criteria. *Nonreactor Facility Safety Analysis Manual*, WHC-CM-4-46, Table 1, Section 9 "Safety Classification", implements the requirements for safety class systems as follows.

- A Safety Class 1 (SC-1) system is a system whose failure could adversely affect the offsite environment or the health and safety of the public.
- A Safety Class 2 (SC-2) system is a system which, if it failed, could adversely affect the health and safety of onsite workers or the onsite environment.
- A Safety Class 3 (SC-3) system is a system whose failure could adversely affect the health and safety of facility workers or result in releases of radioactive or toxic material to the environment.
- A Nonsafety Class 4 (NSC-4) system is a system whose failure has no significant importance to health, safety, or environmental protection.

Using the GENII code from PNL-6584, *The Hanford Environmental Radiation Dosimetry Software System* (Napier et al. 1988), and the bounding waste solution inventory from Tables 3-1, 3-2, and 3-3 of this document as input data, a bounding calculation was made to assess the maximum onsite and offsite individual doses as noted in Section 4.0 of this document. Also, large waste inventories which could be released to the ground were evaluated. Results are shown in Appendix A of this report. An additional analysis to quantify offsite and onsite consequences of all risk-significant events will be performed when more definitive design information and operating data are available.

Because an unmitigated release was found to result in an unacceptable consequence to the public, workers, or environment (as noted in Section 4.2), Safety Class 1 and 2 systems and components are deemed necessary for the project (see Table 2-1). As data become available and more definitive calculations are possible, some system safety classifications may be changed. Detailed accident analyses will be presented in the SA.

The existing systems' safety classifications for C-106 and AY-102, as noted in Tables 2-2 and 2-3, respectively, were analyzed as part of the AWF and SST ISELs (Kidder 1993a and 1993b). Further analyses for these existing systems, if required to support WRSS operation, will be presented in the SA.

Table 2-1. Waste Retrieval Sluicing System Safety Classification.

New systems	Safety Class (SC)	SC justifications	
		Section of SEL (b)	Table 1 (a)
Safety systems			
Sluicing waste transfer system	SC-1	4.1	1, 10, 15
• Seismic switch	SC-1	4.1.1	15
• Primary transfer pipes from and to tanks	SC-2	4.1.2	2
• Secondary transfer pipes from and to tanks (c)	SC-3	4.1.3	3
• Waste transfer pits	SC-1	4.1.4	1
• Slurry distributor	SC-3	4.1.5	3
• Syphon breaker	SC-3	4.1.6	3
• Waste transfer equipment (pumps, valves, pipes flanges, etc.) in pits	SC-3	4.1.7	3
• Raw water supply and transfer line flushing system	SC-3	4.1.8	3
• Compressed air system	SC-3	4.1.9	3
Ventilation system	SC-3	4.2	3
Ventilation exhaust radiation monitoring system (continuous air monitors and record samplers)	SC-3	4.3	3
In-tank imaging system	SC-3	4.4	3
Instrumentation control system	SC-3	4.6	3
AC power	SC-3	4.5	14
Equipment removal system	SC-3	4.7	3
• Water washing, CO ₂ decontamination equipment	SC-3	4.7	3
• Flexible receiver system	SC-3	4.7	3
• Strongback system	SC-3	4.7	3
• Equipment storage container	SC-3	4.7	3
• Container transport trailer	SC-3	4.7	3

(a) Safety classifications are from Table 1, Section 9, WHC-CM-4-46.

(b) From WHC-SD-WM-SEL-033 (Conner 1994).

(c) Secondary transfer pipe (encasement or outer pipe) will meet applicable Safety Class 2 quality assurance and structural/seismic requirements, similar to primary transfer pipe.

AC = Alternating current
 EP = Engineering Practice
 SEL = Safety equipment list
 SC = Safety Class

Table 2-2. Tank 241-C-106 Systems List and Interim Safety Classifications.

Existing system descriptions	Safety Class (SC)	SC justifications	
		Section of SST ISEL (b)	Table 1 (a)
Safety systems			
Primary dome and tank	SC-1	4.1.1, 4.2.1	1, 2
Active ventilation system	SC-3	4.1.2	3
• Exhaust duct headers	SC-3	4.1.2	3
• Exhaust fan, HEPA filters	SC-3	4.1.2	3
• Steam or electric preheater	SC-3	4.1.2	3
• Process condensate drain with seal pot	SC-3	4.1.2	3
• Air inlet filters	SC-3	4.1.2	3
Active ventilation exhaust radiation monitoring system	SC-2	4.3.4, 4.7	19
• Continuous air monitors	SC-2	4.3.4, 4.7	2
• Record samplers	SC-2	4.3.4, 4.7	2
Tank temperature instrumentation	[IOSR]*	4.3.1	--
• Thermocouples	[IOSR]*	4.3.1	--
Tank liquid-level instrumentation	[IOSR]*	4.3.3	--
• Food Instrument Corporation	[IOSR]*	4.3.3	--
• Manual tapes	SC-3	4.3.3	3
Tank leak detection instrumentation	SC-3	4.3.2	3
• Dry wells	SC-3	4.3.2	3
Support systems			
AC power (c)	[IOSR]*	4.4.3	--
• Active ventilation system	SC-3	4.4.3	3
• Temperature monitoring instrumentation	[IOSR]*	4.4.3	--
• Liquid level monitoring instrumentation	[IOSR]*	4.4.3	--
• Leak detection monitoring instrumentation	SC-3	4.4.3	3
Compressed air	SC-3	4.4.4	3
• Instrument air supply to monitoring instrumentation	SC-3	4.4.4	3
Raw water supply	SC-3	4.4.5	3
Area radiation monitors (d)	SC-3	4.4.1	3
Fire protection system	SC-3	4.5	22

(a) Safety classifications are from Table 1, Section 9, WHC-CM-4-46.

(b) From WHC-SD-WM-SEL-027 (Kidder 1993b).

(c) The AC power is classified according to its safety support function and not to its source or distribution. Classifications provided are only applicable to the tank farm fence line. Classification of offsite sources and distribution networks is beyond the scope of this document.

(d) To be removed in the future.

AC = Alternating current

EP = Engineering Practice

HEPA = High-efficiency particulate air

SC = Safety Class

[IOSR]*: The notation [IOSR]* has been designated for those systems, components, and structures important to health, safety, and environmental protection, but, for which current analyses are not sufficient to establish the Safety Classification. Such items shall be treated as Safety Class 3, with additional designation as operational safety requirements (OSR) related. The additional OSR designation will ensure the priority and funding for continued operation and maintenance of these items consistent with OSR requirements (IOSR where applicable).

The [IOSR]* safety classification will be replaced to be consistent with the requirements of WHC-CM-1-3, MRP 5.46, "Safety Classification of Systems, Components, and Structures," as analyses are completed to establish definite safety classifications. In the interim, approval requirements shall include Nuclear Safety and Quality Assurance, as a minimum, and Environmental Assurance where deemed appropriate.

Table 2-3. Tank 241-AY-102 Systems List and Interim Safety Classifications.
(sheet 1 of 2)

Existing system descriptions	Safety Class (SC)	SC Justifications	
		Section of AWF ISEL (b)	Table 1 (a)
Tanks (241-AY-102)	SC-1	--	10
• Primary tank	SC-1	4.2.1	10
• Secondary tank, refractory support pad, concrete structure	SC-1	4.2.2	23
• Secondary tank leak detection system	SC-3	4.3.6	--
Waste Transfer System	SC-1	4.2.3	
• Primary tank and secondary inter tank transfer pipes	SC-1	4.2.3.2	10
• Waste transfer system pits/boxes	SC-1	4.2.3.3	1, 12
• Waste transfer system pits/boxes equipment (pumps, valves, pipes, flanges, leak detectors, etc.)	SC-3	4.2.3.3	3
Primary tank ventilation system (A-702 Building) (c)	SC-2	4.1.3	2
• Ventilation duct work	SC-2	4.1.3	2
• Seal loops	SC-2	4.1.3	2
• Deentrainers, condensers, heaters	SC-2	4.1.3	2
• Exhaust fans	SC-2	4.1.3	2
Backup ventilation system (A-702 Building) (c)	SC-2	4.1.4	2
• 112-m ³ /s (4,000 ft ³ /min) exhauster	SC-2	4.1.4	2
• Ventilation duct work	SC-2	4.1.4	2
Annulus ventilation system (tank 241-AY-102)	SC-2	4.1.5, 4.8	24
Instrumentation (for various systems)			
• Primary tank instrumentation system			
- Temperature measurement system	[IOSR]*	4.3.2	--
- Liquid-level measuring system	[IOSR]*	4.3.7	--
- Pressure/vacuum measuring system	[IOSR]*	4.3.1	--
• Primary tank leak detection systems			
- Liquid-level detector element	SC-2	4.3.5, 4.8	2
- Continuous air monitors	SC-2	4.3.5, 4.8	2
• Ventilation system radiation monitors			
- Primary ventilation exhaust radiation monitoring system	SC-2	4.3.8, 4.8	19
- Annulus ventilation exhaust radiation monitoring system	SC-3	4.3.9	14
• Area radiation monitor system (d)	SC-3	4.3.10	3
• Secondary tank instrumentation			
- Tank concrete structure thermocouples	SC-3	4.3.3	3
- Pressure/vacuum measuring system	SC-2	4.3.4, 4.8	24
Compressed air system (A-701 Building)	SC-2	4.4.7, 4.8	24
• Process air	SC-2	4.4.7, 4.8	24
• Instrument air	SC-2	4.4.7, 4.8	24
• Balance of plant	SC-3	4.4.7, 4.8	--
AC power system (AWF boundaries)	[IOSR]*		
• Normal AC power for primary tank instrumentation	[IOSR]*	4.4.3	--
• Caterpillar diesel generator (e)	[IOSR]*	4.4.3.1	--
• Normal AC power for ventilation	SC-2	4.4.5, 4.4.6	24
• Detroit diesel generator (f)	SC-2	4.4.5.1	24

Table 2-3. Tank 241-AY-102 Systems List and Interim Safety Classifications.
(sheet 2 of 2)

Existing system descriptions	Safety Class (SC)	SC Justifications	
		Section of AWF ISEL (b)	Table 1 (a)
Raw water system (AWF boundaries)	SC-2	4.4.9, 4.8	24
Raw water system (service pits and boxes)	SC-3	4.4.9	3
Emergency cooling water system (c)	SC-2	4.4.10, 4.8	24
Steam supply system (AWF boundaries) (c)	SC-2	4.4.2, 4.8	24
Steam supply system (balance of plant)	SC-3	--	3
Fire protection system	SC-3	4.5	22
Air-lift circulators	[IOSR]*	4.4.1	--
• Compressed air system--process air		4.4.1	--
• Raw water system		4.4.1	--

(a) Safety classifications are from Table 1, Section 9, WHC-CM-4-46.

(b) From WHC-SD-WM-SEL-020 (Kidder 1993a).

(c) May be replaced by Project W-030 ventilation system.

(d) To be removed in the future.

(e) Caterpillar is a trademark of the Caterpillar Corporation.

(f) Detroit is a trademark of the Detroit Corporation.

AC = Alternating current

AWF = Aging Waste Facility

ISEL = Interim safety equipment list

SC = Safety class

[IOSR]*: The notation [IOSR]* has been designated for those systems, components, and structures important to health, safety, and environmental protection, but, for which current analyses are not sufficient to establish the Safety Classification. Such items shall be treated as Safety Class 3, with additional designation as operational safety requirements (OSR) related. The additional OSR designation will ensure the priority and funding for continued operation and maintenance of these items consistent with OSR requirements (IOSR where applicable).

The [IOSR]* safety classification will be replaced to be consistent with the requirements of WHC-CM-1-3, MRP 5.46, "Safety Classification of Systems, Components, and Structures," as analyses are completed to establish definite safety classifications. In the interim, approval requirements shall include Nuclear Safety and Quality Assurance, as a minimum, and Environmental Assurance where deemed appropriate.

2.2 SPECIAL FACILITY DESIGN CRITERIA

Design criteria incorporate requirements from many inputs: the function of the facility, operation of the facility, offsite and onsite safety, and protection of the environment. A comprehensive listing of the parent documents for requirements can be found in DOE Order 6430.1A, Sections 0106 and 0109 (DOE 1989). Comprehensive lists of requirements are also found in DOE Order 5480.4, *Environmental Protection, Safety, and Health Protection Standards* (DOE 1984a); DOE-RL 5480.4B, *Environmental Protection, Safety, and Health Protection Standards for RL* (DOE-RL 1989a); and DOE 5480.5, *Safety of Nuclear Facilities* (DOE 1986). These lists will not be duplicated here. In addition, the requirements noted in WHC-CM-7-5, *Environmental Compliance Manual*, and in Section 6.0 of WHC-SD-W320-FDC-001 (Bailey 1994) are applicable.

This document is confined to addressing the requirement for Special Facilities as used in DOE Order 6430.1A, Division 13, and all subdivisions ending in a designation of "-99" (see Table 2-4), and the requirements in WHC-CM-4-9, *Radiological Design* (see Table 2-5).

Table 2-4. Special Design Criteria.*

Nonreactor Nuclear Facilities - General	
0110-99.0	General Requirements
0111-99.0	Structural Design Requirements
0200-99.0	Site Development
0273-99.0	Water Pollution Controls
0275-99.0	Industrial Waste Water Treatment
0285-99.0	Solid Waste Systems
0900-99.0	Finishes
1530-99.0	Fire Protection
1540-99.0	Plumbing/Service Piping
1550-99.0	Heating, Ventilating, and Air Conditioning Systems
1589-99.0	Air Pollution Control
1660-99.0	Electrical--Special Systems
Special Facilities	
1300	General Requirements
1323	Radioactive Liquid Waste Facilities

*From DOE Order 6430.1A, *General Design Criteria*, U.S. Department of Energy, Washington, D.C. (DOE 1989).

Table 2-5. Allowable Personnel Dose Rates for Design.*

Zone category	Access time allowed	Initial design level (mrem/h)	Maximum (mrem/h)
Uncontrolled area			
	Full time	0.05	0.1
Controlled area			
1	Full time	0.2	0.5
2	Less than 1 h/d	2	5
3	Less than 1 h/wk	10	20
4	Less than 10 h/yr	50	100
5	No normal access permitted	>100	

*Source: WHC-CM-4-9, *Radiological Design*, Section 8.0, Table 1.

2.2.1 Applicable Sections of DOE Orders

The DOE Orders were reviewed to identify special safety-related design criteria pertinent to potentially hazardous facilities. The results of this review are shown in Table 2-4 for the DOE Order 6430.1A, Special Facility Requirements, and in Table 2-6 for the other principal DOE Orders containing safety-related criteria.

Table 2-6. Safety-Related Design Criteria in Other U.S. Department of Energy Orders.

DOE Order 5480.4	<i>Environmental Protection, Safety, and Health Protection Standards</i> (DOE 1984)
DOE-RL Order 5480.4B	<i>Environmental Protection, Safety, and Health Protection Standards for RL</i> (DOE-RL 1989a)

2.2.1.1 Architectural. Preliminary and detailed design for new supporting structures, housing modules, and services of the WRSS will be in accordance with DOE Order 6430.1A (DOE 1989) and Section 2.4 of this document.

2.2.1.2 HVAC. An HVAC system(s) will be provided to ensure safe operation of the WRSS. The HVAC system will be designed to maintain airflow from the WRSS facilities that form part of the confinement barrier. Airflow within the WRSS moves from noncontaminated to progressively more contaminated areas. Consideration will be given to providing separate HVAC supply systems for contaminated and noncontaminated areas. The HVAC system will meet applicable requirements in DOE Orders 6430.1A; 5480.11, *Radiation Protection for Occupational Workers* (DOE 1988); and pertinent HPS.

2.2.1.3 Utilities. Utility requirements for raw water, sanitary water, and electrical power will be assessed in terms of available capacity and tie-in points. Water suppression-type fire protection systems, if used, will be supplied with raw water because existing sanitary water capacities are committed.

The underground water pipe installation complies with HPS-SDC-3.2, "Minimum Depth of Underground Water Lines" (DOE-RL 1973).

Backflow prevention will be built into water supply systems that could possibly become contaminated. Selection and installation will be in accordance with the following criteria:

- DOE Order 6430.1A, Section 1530-9, "Water Storage and Distribution"
- DOE-RL Order 5480.10, *Industrial Hygiene Program*, Part D, "Cross-Connection Control Standard, Hanford Water System" (DOE-RL 1990b)

- WAC 248-54, Paragraph 285, "Public Water Systems"
- *Cross Connection Control Manual: Accepted Procedure and Practices* (AWWA 1990).

2.2.1.4 Lighting. Design criteria for new lighting are described in the following documents:

- ANSI/Illumination Engineering Society (IES) RP7-90, *Practice for Industrial Lighting* (ANSI/IES 1990)
- ANSI/IES RP1-82, *Office Lighting* (ANSI/IES 1982)
- NFPA 70-1990, *National Electrical Code* (NFPA 1990a) and NFPA 101-1994, *Life Safety Code* (NFPA 1994)
- DOE-RL Order 5480.1A, *Environment, Safety, and Health Program for U.S. Department of Energy Operations for Richland Operations*, Appendix A, "DOE Energy Management Directives" (DOE-RL 1988)
- 29 CFR 1910, "Occupational Safety Health Standards and Environmental Control," (General Industry).

2.2.1.5 Electrical. Any new electrical designs will comply with applicable requirements of the following codes and standards:

- NFPA 70-1990, *National Electrical Code* (NFPA 1990a)
- *National Fire Codes* (NFPA 1990b)
- ANSI C2, *National Electrical Safety Code* (ANSI 1990)
- *IES Lighting Handbook* (IES 1993)
- IEEE standards, recommended practices, and guides
- Electrical power systems, components, and structures--portable or fixed in place depending on the intended use and cost
- Safety Class 1 and 2 electrical power systems, components, and structures--designed, fabricated, tested, and installed in accordance with the requirements of DOE Order 6430.1A (DOE 1989).

2.2.1.6 Structural. At a minimum, the following design criteria will be met for new structures.

- All structural systems, components, and structures will be designed, fabricated, inspected, and installed in accordance with the requirements of DOE Order 6430.1A; and HPS-SDC 4.1, "Standard Architectural Civil-Design Criteria: Design Loads for Facilities" (DOE-RL 1989b).

- All structural steel design will meet the minimum requirements of the American Institute of Steel Construction (AISC) *Manual of Steel Construction* (AISC 1989), unless otherwise required by safety classification.
- All Safety Class 1 and 2 concrete designs will meet the minimum requirements of ACI 349-1990, *Code Requirements for Nuclear Safety Related Concrete Structures* (ACI 1990).
- All Safety Class 1 and 2 steel structures will meet the requirements of AISC/ANSI N690-1984, *Nuclear Facilities-Steel Safety Related Structures for Design, Fabrication and Erection* (ANSI/AISC 1984).

2.2.1.7 Energy Conservation. General design guidelines for energy conservation will comply with the following sections of DOE Order 6430.1A:

- 0110-12, "Energy Conservation"
- 1595-10, "Energy Management Systems"
- 1595-11, "Interior Electrical Power and Lighting Systems."

The design must include an energy conservation report in accordance with Section 0110-12.8 of DOE Order 6430.1A. The DOE Order 4330.2D, *In-House Energy Management* (DOE 1992a), contains additional policies and requirements.

2.2.1.8 Maintenance. This section discusses facility and equipment maintenance, and materials selection for maintenance.

Facility--The WRSS facilities and support systems will be designed, arranged, and constructed for efficient, cost-effective operation and maintenance, with as low as reasonably achievable (ALARA) personnel radiation exposure. Cost versus exposure decisions are based on the expected radiation exposure to employees, the planned duration of this exposure, and the nature of required maintenance activities. Remote maintenance capabilities will be provided wherever required and justified. System designs will provide for the safety of maintenance personnel.

Equipment--Provisions have been made for radioactive and nonradioactive equipment maintenance to support WRSS operations, and to minimize equipment downtime. The WRSS will be designed to preclude the need for routine, hands-on maintenance in contaminated or high-dose rate areas. The retrieval equipment design will include features that will allow controlled, intact removal of the equipment in the event of failure in the SST.

Provisions will be made for remote replacement of system equipment that operates in a high-radiation area. The capability to perform a preliminary decontamination of equipment will be provided, where practical, before repair or removal. Services and utilities required to support operation of the WRSS will be provided as part of the project.

All mechanical equipment, windows, cameras, television monitors, and other equipment in areas of potential radioactivity are sealed or otherwise protected, where practical, from penetration by contamination and

decontamination solutions. All electrical and instrumentation equipment will be properly grounded for safety of personnel and protection of equipment.

Materials--Materials will be compatible with the exposed environment. Wherever possible, radiation-resistant fabrication materials will be specified for equipment to be located in radiation fields. Use of materials such as plastics, organic compounds, lubricants, and insulation that degrade in radiation environments is minimized. These materials will be used only when no substitutions are acceptable or when costs preclude the use of other materials.

Installed materials will be resistant to radiation, process solutions, acid and caustic vapors and solutions, and known decontamination agents. The materials will be nonabsorbent and easily removable if not contamination resistant.

2.2.2 Applicable Sections of WHC-CM-4-9, *Radiological Design*

Radiological Design, WHC-CM-4-9, was reviewed to determine the sections with which this project must comply. The results are shown in Section 2.4.2 of this document.

2.2.3 Applicable Sections of WHC-CM-6-1, *Standard Engineering Practices*

The project will comply with WHC-CM-6-1, *Standard Engineering Practices*, during definitive design.

2.2.4 Applicable Sections of Other WHC Manuals

A thorough review of these manuals will be made during the definitive design, and incorporated into the SA. Section 8.0 specifies that a thorough review will be ensured during definitive design, and incorporated into the SA.

2.3 WRSS PROCESS DESIGN CRITERIA

The C-106 WRSS represents a method of removing tank waste that requires a new storage location. The method of retrieval is capable of mobilizing and transferring the sludge from C-106 to the receiver (AY-102). Receiver tank system modifications or upgrades necessary to support retrieval of C-106 waste are included in this project.

The WRSS design will consider human factors for maintenance and operations activities. This includes control panels, equipment and valve location, secondary waste handling, and facility arrangement. The design will comply with DOE Order 6430.1A, Section 1300-12 (DOE 1989).

2.3.1 Waste Retrieval and Storage

The C-106 WRSS will be designed to remove the radioactive and chemical sludge from C-106 and transport that material to AY-102 via a new, temporary, shielded, encased transfer line. The system includes hardware specific to retrieval of waste tank contents; all supporting structures; facilities (including control room, lunchroom, and sanitary facilities); waste tank modifications; receiver tank system modifications necessary to retrieve the C-106 waste; the transfer line to and from the DST system; and all associated support equipment.

The solution used to sluice out the solid contents of C-106 will be the liquid waste that is transferred into AY-102 from AY-101 or an alternate fluid as stated in ES-290 (Sederberg 1994). This will result in the C-106 and AY-102 solutions having the same basic classification (i.e., dilute complexed, transuranic liquid waste).

Although it is likely the contents of AY-101 will be replaced with different liquid before sluicing operations begin, its contents will be of the same dilute complexed waste type at that time. The current chemical and radionuclide composition of AY-101 liquid is considered to be a bounding case for future waste receipts in the timeframe in question. The current AY-101 characterization is therefore used in the airstream source term calculation.

The C-106 WRSS includes the following functions.

- Remove existing in-tank equipment.
- Deploy the retrieval equipment and systems.
- Install removal equipment.
- Mobilize and retrieve waste in C-106.
- Convey waste out of C-106.
- Transfer waste to AY-102 at a controllable rate.
- Confine and filter airborne hazardous vapors and radioactive particles at C-106.
- Remove heat as required to maintain acceptable temperature levels in C-106.
- Avoid any tank dome or riser loading that exceeds allowable limits.
- Supply electrical power, telecommunications, and all other support utilities.
- Decontaminate and transfer failed equipment.
- Provide replacement tank pit cover block assemblies.

- Monitor and control operations.
- Provide shielding and maintenance actions.
- Provide in-tank imaging system for monitoring sluicing operations and waste retrieval progress.
- Remotely monitor surface level of C-106 waste with a level-sensing device.
- Provide required HVAC for C-106 and retrieval support facilities.
- Recirculate clarified supernate from AY-102 to C-106.
- Recirculate slurry from the sluice pump to the sluicers at C-106.
- Minimize operating liquid content in C-106.

Any equipment that might be necessary to facilitate transportation of the WRSS or any of its components to or from the retrieval site (before or after waste tank retrieval, or for the purposes of reuse, refurbishment, or disposal) will be provided as part of the WRSS.

Additional information on pipe erosion, supernatant/slurry viscosity, and sludge particle size distribution in C-106 is presented in the following sections and in Tables 2-7 and 2-8.

Slurry Erosion--It is expected that sluicing operations, at the maximum practical slurry concentration and stated system design specifications, will result in Project W-320 process piping system's erosion of no more than 30 mil/yr at the greatest flow obstructions.

Table 2-7. Supernatant/Slurry Viscosity.

Solids locating (%)	Viscosity (cP) at 20 °C (68 °F)
0 (clear)	4
5	8
10	13
15	26
20	40
25	73
30	105
35	155
40	230
45	315
50	440
The effect of temperature on viscosity from 20 to 58 °C (68 to 138 °F) is expressed as:	
$\text{viscosity at } T = \text{viscosity at } 68 \text{ }^{\circ}\text{F} (1 - 0.005[T - 68 \text{ }^{\circ}\text{F}]), \text{ where } T \text{ is } ^{\circ}\text{F}.$	

Table 2-8. Tank 241-C-106 Sludge Particle Size Distribution.

Particle size range (μm)	Weight %
5 - 10	35
10 - 15	11
15 - 20	5
20 - 25	12
25 - 30	4
30 - 35	1
35 - 40	8
40 - 45	11
45 - 50	7
50 - 55	6

NOTE: It is expected that small amounts of hardened sludge particle agglomerations exceeding 0.26 cm (0.25 in.) diameter exist in C-106. The slurry pump suction inlet should be screened to prevent entry of large sludge particle agglomerations of this size.

2.3.2 Design Life

All WRSS process components and supporting equipment will have a minimum design life of 2 years, including 1 year for retrieval operations. Materials that have a standard design life of more than 2 years will not be degraded to meet these minimum requirements. Design life is exclusive of vendor testing and any shipping and/or storage. Therefore, equipment, components, and systems will not require extended design or operational life considerations.

2.3.3 General Mechanical Process

Equipment will be designed so that no single failure of mechanical or electrical equipment will cause a loss of control resulting in an unrecoverable condition. Functions will be provided that will facilitate placement of the equipment in a safe configuration during a faulted condition.

Design of the equipment will provide fail-safe operation in the event of loss of the control system power supply.

Equipment will be positioned away from potential leaks of corrosive liquids, or will be provided with shielding. Rotating equipment, electrical components, and other high-maintenance items will be located outside radiation areas to the extent practical.

Commercially available equipment and parts will be used to the greatest extent practical. Equipment will be designed for recovery from in-process failures. This criterion applies particularly to remotely operated in-tank equipment. Loads expected throughout the entire product life cycle, including fabrication, storage, installation, operation, and decommissioning, will be considered in the design.

2.3.4 Capacity

The WRSS is capable of transferring a minimum of 22 L/s (350 gal/min) of abrasive fluids with specific gravities (SpG) up to 1.25 (equivalent to 50% solids loading) in the C-106 recirculation loop and 1.20 (equivalent to 30% solids loading) in the balance of the process piping systems. Minimum pipeline transfer velocities are 1.8 m/s (6 ft/s). The maximum allowable design velocities will be determined during the definitive design.

Based on rheological studies performed for plutonium extraction neutralized current acid waste (NCAW), which is similar to C-106 sludge in physical and chemical composition, the maximum pumpable concentration of C-106 sludge in water is expected to be about 30% by weight of settled solids. Laboratory tests indicate that sludge can be resuspended when waste particles are small (in the range of 7 to 15 μm). Larger particles (greater than 15 μm) can also be slurried. The viscosity of the sludge was not measured in WHC-SD-W139-ES-001, *Interim Stabilization of Tank 241-C-106 Engineering Study* (Cornwell and Esvelt 1990), but is anticipated to be in the range of 3 E+5 centipoise (Cp).

2.3.5 Tank Environmental Conditions

All parts of the retrieval system that will be exposed to contaminated or hazardous waste are corrosion resistant. Equipment employed within C-106 and AY-102 will be capable of functioning in the environmental ranges listed in Table 2-9. Ambient conditions are provided for exposed portions of equipment.

2.3.6 Operating Temperature

The WRSS is designed to withstand normal waste temperatures in the range of 21 to 82 °C (70 to 180 °F).

2.3.7 HVAC Control System

The HVAC control system will be designed and operated to maintain emissions of radionuclides ALARA (WAC 173-480, "Ambient Air Quality Standards and Emission Limits for Radionuclides"). Additionally, the control system will use, as a minimum, best available control technology (WAC 173-400, "General Regulations for Air Pollution Sources"), best available control technology for toxic materials (WAC 173-460, "Controls for New Sources of Toxic Air Pollutants"), and best available radionuclide control technology (WAC 246-247, "Radiation Protection--Air Emissions").

Table 2-9. Operating Environmental Conditions.

Equipment/tank no.	Max. design temperature*	Min. design temperature*	Normal operating temperature range*
Slurry pump - 106-C	82 °C (180 °F)	5 °C (40 °F)	16 to 49 °C (60 to 120 °F)
Heel pump - 106-C	82 °C (180 °F)	5 °C (40 °F)	16 to 49 °C (60 to 120 °F)
Sluicer #1 - 106-C	82 °C (180 °F)	5 °C (40 °F)	16 to 49 °C (60 to 120 °F)
Sluicer #2 - 106-C	82 °C (180 °F)	5 °C (40 °F)	16 to 49 °C (60 to 120 °F)
Imaging system - 106-C	49 °C (120 °F)	5 °C (40 °F)	10 to 35 °C (50 to 95 °F)
Miscellaneous probes - 106-C	82 °C (180 °F)	5 °C (40 °F)	16 to 49 °C (60 to 120 °F)
Ventilation system - 106-C	82 °C (180 °F)	10 °C (50 °F)	10 to 35 °C (50 to 95 °F)
Jumper assemblies - 106-C	82 °C (180 °F)	5 °C (40 °F)	16 to 49 °C (60 to 120 °F)
Valve actuators - 106-C	66 °C (150 °F)	5 °C (40 °F)	16 to 49 °C (60 to 120 °F)
Supernate pump - 102-AY	49 °C (120 °F)	5 °C (40 °F)	10 to 35 °C (50 to 95 °F)
Distributor - 102-AY	49 °C (120 °F)	5 °C (40 °F)	10 to 35 °C (50 to 95 °F)
Valve actuators - 102-AY	66 °C (150 °F)	5 °C (40 °F)	16 to 49 °C (60 to 120 °F)
Miscellaneous probes - 102-AY	49 °C (120 °F)	5 °C (40 °F)	10 to 35 °C (50 to 95 °F)
Fluid transfer lines	82 °C (180 °F)	5 °C (40 °F)	16 to 49 °C (60 to 120 °F)
Miscellaneous control wiring	Same as parent equipment temperature requirements.		

NOTE: The following are common site and environmental operating conditions for waste retrieval sluicing system equipment:

- Ambient temperature range - -29 °C (-20 °F) to 46 °C (115 °F)
- Ambient humidity range - uncontrolled
- In-tank radiation dose rate - 540 rad/h
- Total integrated radiation dose - 1×10^7 rad
- In-tank humidity range - 0 to 100%
- Chemical Ph range (wetted components) - 7 to 13.

*Minimum and maximum design temperatures are intended for purposes of structural design.

The purpose of the HVAC system is to maintain confinement of particulates within the C-106 vapor space during sluicing operations. At a minimum, the HVAC system controller will perform the following functions:

- Differential pressure indication and alarm for process related high-efficiency particulate air (HEPA) and absorption filtration units
- Isokinetic monitoring station at HVAC exhaust stack
- Tank vacuum monitoring and control system capable of throttling adjustments and/or HVAC shutdown on tank ventilation system failure
- Emissions monitoring at HVAC exhaust stack to meet 40 CFR 61, "National Emission Standards for Hazardous Air Pollutants," Subpart H, and referenced requirements.

The HVAC system will provide cooling to maintain the tank within operating temperature specifications of OSD-T-151-00013, *Operating Specifications for Single-Shell Tanks*, paragraph 13.2.1E (Boyles 1992). No requirement exists for maintaining a particular negative pressure during transient construction and maintenance operations.

A minimal ventilation flow from C-105 will be maintained and will be cascaded into C-106 (see Figure 1-4). Leaving the cascade line in place would avoid the personnel exposure associated with isolating this line.

2.3.7.1 HVAC System Diagnostics Capability. The HVAC system controller will provide diagnostic information relating the system performance and status. Should the operations be automatically terminated or fail to restart upon demand, the cause of the shutdown or failure will be clearly identified.

2.3.7.2 In-Tank Imaging System. An in-tank imaging system will be provided for operations personnel in the control room so they can view the inside of the tank. The imaging system will facilitate monitoring the minimum possible liquid level in the tank and aiming the sluicer at the remaining waste. This system will enable operators to determine liquid level and location of remaining waste while minimizing the liquid level. The imaging system will be capable of providing images with a resolution of 5 cm (2 in.) at maximum in-tank range of 21.3 m (70 ft).

A portable imaging system will be provided to periodically monitor the integrity of the interior of the process pits through the 5-cm (2-in.) valve handle holes in the cover blocks. The system will be used for determining pit and equipment conditions for leak verification before maintenance and to check for leak verifications and equipment conditions.

2.3.7.3 Miscellaneous. Instrumentation and monitoring equipment systems will be specified and designed to facilitate troubleshooting and replacement.

2.3.8 Electrical and Fluid/Pneumatic Systems

Transfer lines carrying radioactive liquids, dangerous wastes, or hazardous chemicals outside the waste tank (between treatment, storage, and disposal facilities) have secondary (pipe-in-pipe) containment with leak detection and alarm capabilities.

Piping and valve arrangements will be designed to preclude misrouting of liquids outside their confinement boundaries due to incorrect valve settings.

Vessel and piping design will accommodate accidental overflows and pressurization by diverting the excess fluids to C-106 or AY-102.

Equipment such as dip tubes, sample lines, transfer lines, and drains will be designed to preclude the possibility of syphoning or backflow.

Alternating current (AC) power from the Hanford Site power grid will be used to supply electrical equipment during sluicing. This will be accomplished via a skid-mounted transformer, switchgear, and a motor control.

center. All electrical equipment and components are selected or protected to resist adverse affects of the environment in which they are installed. Electrical enclosures and junction boxes of the proper National Electrical Manufacturers Association rating and material are used to resist and protect internal components from the corrosive effects of moisture and chemicals.

Sensitive electronic devices will be located outside high-radiation areas, if possible. The equipment will be designed to withstand the demands of the environment and meet the design life requirements previously identified.

2.3.9 Piping

Piping will be in accordance with the requirements of DOE Order 6430.1A, Section 1300-3.2, based on the safety class designation of the system.

All piping will be arranged for gravity flow wherever possible and will be sloped to drain. The transfer lines will include provisions for flushing. Process piping sizes will be specified to minimize solids from accumulating. All piping will be designed to preclude radioactive wastes from being syphoned or backflowed into nonradioactive lines.

The sluicers to be provided by this project may be similar to, or identical in form and function to, that shown in drawing H-2-70030, "Sluicer Assembly" (AEC 1976).

Cathodic protection will be considered for the buried transfer piping. Provisions will be made to ensure that required interfaces with existing 200 Area cathodic protection schemes are made.

Temporary, below-grade, shielded transfer lines will be used between tanks C-106 and AY-102. Valve and actuation design will minimize the accumulation of solids in crevices or water hammer caused by fast-closing valves.

Adequate protection will be provided for waste transfer lines with consideration given to radiological exposure to personnel and damage by vehicles.

A slurry distributor will be provided to distribute the solids as evenly as possible and maximize solids settling in AY-102.

2.3.10 Instrumentation and Control

The instrumentation and control systems will monitor and control retrieval and transfer operations, HVAC, and support systems to perform all functions necessary to retrieve the contents of C-106 and transport the material to AY-102.

The monitoring and controls required for the Safety Class 2 functions will be provided for operator interface at a standalone Safety Class 2 skid.

The sluicing control room incorporates control panels with monitoring and control instruments associated with the WRSS, and a graphic display depicting the sluicing and waste transfer between C-106 and AY-102, sluicing controls, an in-tank remote imaging system monitor and controls, and a surveillance system monitor and associated controls. The control room provides instrumentation for monitoring the HVAC system, electrical system, and water and air supplies for the WRSS.

The sluicing control and associated instrumentation system will include engineered features that allow either of the sluicers to be aimed at the waste contained in the tank. Control features also will be provided to permit safe sluicing within restricted tank interior areas. No sluicing will be permitted within prohibited areas (e.g., lead flashing at dome/wall interface).

New and existing tank farm and sluicing system interlock functions required to maintain safe operations will be provided. Alarms will annunciate in the sluicing control room and will be audible/visible within the tank farm proper, as required.

The control panel for the WRSS is a graphic panel minimizing the number of software-based controllers used in the control system. Any software developed as a part of the control system design will comply with WHC-CM-3-10, *Software Practices*. The human-factor engineered graphical display includes pump status (on-off), valve position (open-closed), and flowing status (flow-no flow).

The leak detection system and radiation monitoring system associated with the transfer lines and pits are integrated with the misrouting prevention system and will be alarmed in the control room. The misrouting prevention system interlock will abort either pumping system for an unplanned or controlled addition of water for flushing. The location of leak detection system failure or misrouting will be alarmed on the graphic display by the misrouting prevention system.

Instrumentation is designed for "fail safe operation" in the event of an air or electrical outage. These events will activate alarms upon signal failure. Any required radiation monitoring and leak detection instruments will have a "fail-safe" alarm feature on the loss of ability to detect radiation or leakage. Instrument and power cable runs are housed in separate raceways within the tank farm. Any outside alarms in the tank farm that are required to be visual will have high-intensity strobe indicators. Radiation monitoring capabilities will be provided on water lines that come into contact with a radioactive environment or that have the potential for contamination.

A retractable temperature measurement device will be provided to measure tank waste and vapor space temperatures.

Instrumentation will be designed for its intended service and qualified for the environmental conditions in which it is required to function.

2.3.11 Support Services

Raw water, sanitary water, instrument air, process air, telecommunications, hoisting hardware, and a step-off pad will be provided. The use of existing sanitary fields will be considered. A sanitary catch tank that is sized for 1 week of operation will be provided. An access provision will be made for chemical additions and for tank sampling.

Flushing will be accomplished with a combination of clarified supernatant from AY-102 and/or raw water. Flushing of transfer lines from C-106 to the receipt DST will be provided. The system will be capable of flushing all process lines, jumpers, pumps, and sluicers with treated water. Flushing facilities will be provided at C-106.

2.3.12 Operating Requirements

The WRSS facility will be designed to be unstaffed during nonoperational periods or after emergency shutdown.

Existing instrumentation for tank liquid level (Food Instrument Corporation and manual tapes), temperature, pressure/vacuum, flow meters, and leak detection systems will be used, as required, to monitor operating conditions in C-106 and AY-102 during the WRSS operation. The FIC is currently being phased out and replaced with the Enraf series 854 level gauge. The Enraf gauge is a high precision, low maintenance device that determines waste level by detecting variations in the weight of a displacer suspended in the tank waste. Flow meters and a running mass balance will be able to detect a leak when about 30,000 L (8,000 gal) are removed from the closed system. These monitoring systems will be extended to the WRSS control room for operator convenience. Safety classifications for these systems are included in the respective ISELs for AWF and SST (Kidder 1993a and 1993b); also see Tables 2-2 and 2-3.

The waste level of C-106 will be monitored and will not exceed 201 cm (79 in.) above the bottom of the tank. The vapor space pressure of this tank will be maintained between -0.8 and -15.0 cm (-0.3 to -5.9 in.) water gauge relative to atmospheric pressure during active sluicing operations.

Selected new alarms from radiation monitors, leak detectors, tank containment pressures, and HVAC system shutdown will alarm to the tank monitor and control system (TMACS). The selected alarms from the radiation monitors, leak detectors, and tank containment pressures will be those of abnormal conditions (high radiation, leak, high vacuum, low vacuum) and system failure. The alarms required for TMACS operator interface in the 2750-E Building will use available spare points on the input/output board located in the terminal box on tank C-106.

The retrieval system will be designed to minimize the generation of new liquid waste created as a result of the retrieval operations.

The retrieval equipment will be remotely operable from a control room located outside the tank farm boundary. Systems operation will be in accordance with ALARA principles. Only limited hands-on activities will be

required in the tank farms to facilitate operation of the WRSS. Process monitoring by the imaging system will be provided to facilitate maintaining the minimum possible liquid level and to aim the sluicer at the remaining waste. The imaging system is intended to enable operators to determine the level. Human factors engineering will be performed in accordance with DOE Order 6430.1A (DOE 1989) to enhance assembly, operation, maintenance, and disassembly of the retrieval system, and to minimize potential for operator error.

The project design assumes that the C-106 WRSS will be operated by a crew divided into three operating shifts that operate 24 hours/day, 7 days/week. It has been projected that the crew will consist of the following staff:

- Eighteen operators (three on each A, B, C, D, and E shift, and three on day shift)
- Six health physics technicians (one on each A, B, C, D, and E shift, plus one on day shift)
- Six supervisors (one on each A, B, C, D, and E shift, plus one on day shift)
- Six engineers (one on each A, B, C, D, and E shift, plus one on day shift).

2.3.13 Thermal Considerations

The currently applicable tank farm operating temperature specifications (Boyles 1992) provide the following operating temperature limitations: a maximum change in temperature of the undisturbed waste will be less than 11.1 °C (20 °F)/day, and a maximum waste temperature will be 149 °C (300 °F) with a maximum allowable tank dome temperature of 121 °C (250 °F).

2.3.14 Waste Tank Loading

The WRSS design will ensure that the operating specification document (Boyles 1992) requirements will not be exceeded by the equipment placement nor by operation of the retrieval system. This will prevent damage to the structure, which might result in the loss of tank contents into the environment.

Project W-320 design will ensure that operation of the retrieval system and equipment placement will not induce loading in the tank structure, which exceeds the tank's allowables, or cause damage to the structure, which could result in the loss of the tank contents into the environment. Allowable tank loading during the retrieval of its contents will be in accordance with OSD-T-151-00013, page 6, "Dome Loading" for C-106 (Boyles 1992) and OSD-T-151-00017, page 10, "Dome Loading" for AY-102 (Bergmann 1989).

2.4 COMPLIANCE WITH SAFETY-RELATED DESIGN CRITERIA

The special facility safety-related design criteria sections identified in Tables 2-4 and 2-6 were compared with available information on the WRSS design to assess compliance. The assessment will need to be reviewed with each design iteration to incorporate new information and to reassess conclusions. The following paragraphs draw attention to criteria already identified as requiring clarification.

2.4.1 DOE Order 6430.1A, *General Design Criteria*

During definitive design, a more comprehensive comparison to the DOE requirements will be developed and incorporated into the SA. During this time, a DOE Order 6430.1A comparison checklist will be completed by WHC to ensure compliance to the applicable sections of the DOE Order. It is noted in Section 8.0 that any outstanding design issues will be resolved before hot operations.

2.4.2 WHC-CM-4-9, *Radiological Design*

The retrieval system is designed to confine contaminants within the combination of boundaries defined by the existing tank, the confinement surfaces of the retrieval system, and any other barriers that may be required. A system is employed to control contamination caused by failure of the primary confinement or other off-normal situation.

Section 3.4.2 of WHC-CM-4-9 lists 29 criterion, most of which cannot be checked because of the limited design definition. These criteria will be incorporated during the detailed design, and verified by the SA.

Section 11.4 of WHC-CM-4-9 lists numerous criterion that will be checked during the detailed definitive design development. Section 11.4.3, Criterion 12, calls for the outer pipe of a containment system to be designed to system pressure. These requirements will be incorporated during the detailed design, and verified by the SA.

Guidelines for radiological design are provided in WHC-CM-4-9. The shielding design criteria specified in Section 8.0 of WHC-CM-4-9 was used to determine the shielding requirements for different areas in the facility to meet ALARA guidelines according to DOE Order 5480.11, *Radiation Protection for Occupational Workers* (DOE 1988); WHC-CM-1-6, *Radiological Control Manual*; and WHC-CM-4-11, *ALARA Program Manual*. The annual exposure to an individual from all sources should be ALARA and less than 500 mrem, summed over all controlled access areas, unless approved by management in accordance with the requirements of WHC-CM-1-6. The source term used for shielding design is the maximum expected during normal operation. Consideration will be given to normal operation, abnormal events (e.g., jumper or pipe leaks/breaks), and maintenance activities. Where there is a potential for airborne contamination, allowance will be made for internal deposition of radionuclides in determining the total dose. Allowable airborne contamination levels are

provided in WHC-CM-1-6. Source terms which may develop in operating areas, such as in process and valve pits, will be considered.

Another area that must be verified at the final design stage and in the SA are the requirements for the necessary shielding in an uncontrolled area. Section 8.0, Table 1, of WHC-CM-4-9 (see Table 2-5 of this document) gives upper and lower bounds of 0.1 and 0.05 mrem/h, respectively, for the shielding design. These criteria will be incorporated into the detailed design.

A last item for future verification is the potential for release of contamination from the pits at the time of lid removal. One proposed solution is the use of sampling and washdown capabilities. This and other solutions will be evaluated during detailed design to resolve the issue. Verification is needed in the final detailed design and the SA.

In accordance with Section 3.6.1 of WHC-SD-WM-OSR-005 (Dougherty 1993), pit covers are required to be installed and to remain in place while the transfer operation is performed.

2.4.3 WHC-CM-7-5, *Environmental Compliance Manual*

The environmental requirements for new and modified facilities, as defined in Section 9.0 of WHC-CM-7-5, *Environmental Compliance Manual*, will be applicable to the design and operation of the WRSS. In addition, all applicable *National Environmental Policy Act of 1969* requirements will be addressed.

An environmental requirements checklist will be prepared identifying all applicable environmental and permitting requirements for the project. This checklist will serve as the basis for development of the project permitting plan. The permitting plan will identify the permitting alternatives and strategy for the project. The following permits and approvals are required:

- *Resource Conservation and Recovery Act of 1976*, Part A permit application modification for the SSTs
- Clean air approvals required according to the *Clean Air Act of 1977* and Amendments of 1990:
 - Approval of Modification according to 41 CFR 61, Subpart H., "National Emission Standards for Hazardous Air Pollutants; Radionuclides"
 - Approval of Construction according to WAC 246-247, "Radiation Protection--Air Emissions"
 - Approval of Modification according to WAC 173-460, "Controls for New Sources of Toxic Air Pollutants"
- A *National Environmental Policy Act of 1969* Environmental Assessment.

This project will be compatible with ongoing environmental protection and compliance projects and include any necessary upgrades to the 241-C Tank Farm required to facilitate the retrieval of C-106 wastes.

2.5 DECONTAMINATION AND DECOMMISSIONING

The design of the WRSS facilitates decontamination so that the facility can be decommissioned at a future date. Guidance for retrieval equipment design to facilitate eventual decommissioning will be obtained from DOE Order 4700.1, *Project Management System*, (DOE 1987); ANSI N300-1975, *Design Criteria for Decommissioning of Nuclear Fuel Reprocessing Plants* (ANSI 1975); and DOE Order 6430.1A, Sections 1300-11.2 and 1321-7 (DOE 1989).

The following principles are employed to the extent practicable:

- Use of modular, separable confinements for radioactive and other hazardous materials to preclude contamination of fixed portions of the structure
- Location of exhaust filtration components of the ventilation systems at or near individual enclosures to minimize long runs of internally contaminated ductwork
- Areas subject to contamination designed to facilitate decontamination. Liners and coatings selected to withstand decontaminating agents and radiation degradation throughout the life of the facility. Surfaces free of crevices, ledges, and/or protrusions which could collect radioactive material.
- Penetrations designed for closing and sealing at the time of decommissioning
- Penetrations waterproofed to facilitate decontamination efforts
- Fixtures and outlets sealed
- Drains and similar piping have physical provisions for cleaning
- Piping systems sloped and free of traps except as required for retrieval process isolation
- Adequate clearance provided for transfer of equipment
- Use of lifting lugs on large tanks or equipment
- Use of modular radiation shielding in lieu of, or in addition to, monolithic shielding walls.

3.0 HAZARDOUS MATERIAL INVENTORIES

This section describes the radiological and nonradiological hazardous material inventories expected for the liquid waste to be transferred from existing C-106 facilities to AY-102 for storage and future processing. As noted in Section 8.0, contents of the tanks will be satisfactorily represented by the hazardous material description used in the SA before operation.

3.1 DESCRIPTION OF RADIOLOGICAL MATERIALS

The waste in C-106 contains radionuclides with greater than 100 nanoCuries per gram (Nci/g) transuranic content and qualifies the sludge in C-106 as both high level and transuranic waste. The chemical composition of the sludge also qualifies the contents as "listed waste" in accordance with WAC-173-303, "Dangerous Waste Regulations."

A conservative estimate of the radionuclide inventories for the C-106, AY-102, and AY-101 tanks used in the WRSS is shown in Tables 3-1, 3-2, and 3-3, respectively.

An analysis was performed (see Appendix C of WHC-SD-WM-ISC-001, Leach and Stahl 1993) for unit liter respirable releases of selected Hanford Site high-level waste tanks which included C-106 and AY-102 (AY-101 supernatant or an alternate fluid as stated in Sederberg 1994 will be transferred to AY-102; see Section 2.3.1). The unit liter release calculations were made using the GENII dosimetry code (Napier et al. 1988) for both laboratory sample data for C-106 solids and AY-101 supernatant. These EDE unit liter dose rates have been used in dose consequence calculations in Appendix A of this document, and are summarized in Table 3-4. This table represents maximum doses to the offsite individual.

Because all the radionuclides of concern have long half-lives, doses to the receptors at other locations are proportional to these maximum doses calculated for the offsite individual. The atmospheric dispersion coefficients used to derive these conversions are provided in WHC-SD-WM-ISC-001, Appendix C, Table C-5. To convert from the offsite individual maximum dose to doses for other receptors, the receptor location conversion factors given in Table 3-5 (excerpted from WHC-SD-WM-ISC-001, Appendix C, Table C-10) are used. For the purpose of this evaluation and as recommended in WHC-SD-WM-ISC-001, Appendix C, the most conservative multiplication factors (CF) with or without plume meander (PM) have been used and are as follows: CF=1 for maximum doses to offsite individual without PM; CF=0.64 for maximum doses to offsite individual with PM; CF=610 for maximum doses to onsite individual with PMs; and CF=1,900 for maximum doses to onsite individual without PM. The PM is applied for postulated accident releases to occur for 1 hour or greater.

The unit liter release dose rate calculations for C-106 are listed in Table 3-4 for offsite individual without/with PM, Table 3-6 for onsite individual without PM, and Table 3-7 for onsite individual with PM. It is

assumed that the solids/sludge within AY-102 are not disturbed during sluicing from C-106 to AY-102.

Table 3-1. Tank 241-C-106 Radionuclide Inventory Data. (a)

Sample type	Drainable interstitial liquid			Sludge/slurry solids			Tank inventory
Sample date	05/19/86			05/19/86			05/19/86
Density (g/ml)	--			1.43			--
Volume (kgal)	16			197			213
Volume (L)	6.06E+04			7.46E+05			8.06E+05
UNITS	Ci/L	Bq/L	Bq	Ci/g	Bq/L	Bq	Bq
C ¹⁴	3.51E-01	1.30E+04	7.87E+08	2.32E-04	1.23E+04	9.12E+09	9.94E+09
Co ⁶⁰	<1.27E+01	4.70E+05	2.85E+10	8.88E-01	4.50E+07	3.51E+13	3.51E+13
Sr ^{89/90}	1.65E+03	6.11E+07	3.70E+12	1.98E+03	1.05E+11	7.82E+16	7.82E+16
Tc ⁹⁹	1.41E+02	5.22E+06	3.16E+11	2.19E-01	1.16E+07	8.64E+12	8.96E+12
I ¹²⁹	1.40E-02	5.18E+02	3.14E+07	8.10E-05	4.29E+03	3.20E+09	3.23E+09
Cs ¹³⁷	2.78E+04	1.03E+09	6.23E+13	3.30E+02	1.75E+10	1.30E+16	1.30E+16
U-Nat	9.58E-01 g/l	2.50E+01	1.52E+06	4.09E+02 g/g	1.53E+04	1.14E+10	1.14E+10
Pu ^{239/240}	9.78E+02	3.62E+07	2.20E+12	2.69E+00	1.42E+08	1.06E+14	1.08E+14
Am ²⁴¹	<1.39E+01	5.14E+05	3.12E+10	1.05E+00	5.56E+07	4.15E+13	4.15E+13
Total α	--	--	--	--	--	--	--
Total β	--	--	--	--	--	--	--
Total γ	2.78E+04	1.03E+09	6.23E+13	3.63E+02	1.92E+10	1.43E+16	1.44E+16

(a) Source: WHC-SD-WM-TI-565 (Van Vleet 1993b).

Table 3-2. Tank 241-AY-102 Radioactive Nuclide Laboratory Data. (a)

Sample	--	--	Core	--	--
Sample date	November 1984		January 1988	February 1990	
Reference #	65453-84-348		PNL log	PNL report	
Sample #	R-3214	R-3215	PNL 12988	Supernate	Washed solids
Density (g/ml)	--	--	--	≈1.0	≈1.4
C ¹⁴	--	--	--	NA	<1.40E+00
Co ⁶⁰	--	--	--	ND	2.02E+03
Sr ^{89/90}	2.32E+04	IS	--	6.58E+03	4.13E+07
Y ⁹⁰	--	--	--	--	--
Tc ⁹⁹	--	--	--	NA	3.50E+01
I ¹²⁹	--	--	--	NA	<1.68E+00
Cs ¹³⁴	--	--	--	4.95E+00	ND
Cs ¹³⁷	5.925E+05	7.525E-01	--	4.32E+03	1.60E+03
Pm ¹⁴⁷	--	--	--	--	--
Eu ¹⁵⁴	--	--	--	ND	7.20E+04
U-Nat.	--	--	--	5.71E-01	1.46E+01
Np ²³⁷	--	--	5.29E+01	--	--
Pu ²³⁸	--	--	1.59E+01	ND	1.60E+03
Pu ^{239/240}	--	--	3.18E+01	2.34E-01	5.05E+03
Pu ²⁴¹	--	--	1.90E+02	--	--
Am ²⁴¹	--	--	2.91E+02	1.50E-03	2.55E+04
Am ^{242M}	--	--	--	--	--
Am ²⁴³	--	--	--	--	--
Cm ²⁴²	--	--	--	ND	8.83E+02
Cm ²⁴⁴	--	--	7.41	--	--
Total α	--	--	--	--	--
Total β	--	--	--	--	--

(a) Source: WHC-SD-WM-TI-543 (Van Vleet 1993a).

NOTE: All units are in Ci/L unless otherwise indicated.

IS = Insufficient sample
 NA = Not analyzed
 ND = Not detected

Table 3-3. Tank 241-AY-101 Radionuclide Laboratory Data. (a)

Sample	CC Super	CC Super	--	Complexant concentrate solids	Supernate	Slurry
Sample date	February 1985		March 1987	March 1985	July 1988	
Reference #	65453-85-041		65453-87-096	65453-85-053	12221-PCL88-169	
Sample #	R-3641 By AI	R-3641 By APDU	TD062387 CPLX	R-3642	R-8371	
Co ⁶⁰	--	--	--	NA	--	--
Sr ^{89/90}	2.00E+05	--	1.44E+05	1.1E+04	9.87E+06 (b)	--
Tc ⁹⁹	--	--	19.6	--	--	--
I ¹²⁹	--	--	<0.15	--	--	--
Cs ¹³⁷	2.20E+06	--	1.15E+05	5.2E+02	1.09E+05	1.54E+04
U-Nat.	7.83E-03	--	--	2.26	--	--
Pu ^{239/240}	1.21E+01	4.2	4.03E-03	6.77E+01	--	--
Pu ²⁴¹	--	1.60E+01	--	--	--	--
Am ²⁴¹	3.03+01	1.60E+01	NR	3.01E+02	--	--
Total α	--	--	--	--	4.80E+02	6.51E+03
Total β	--	--	--	--	--	--

(a) Source: WHC-SD-WM-TI-565 (Van Vleet 1993b).

(b) Assume all Sr²⁺ of 0.07 g/L is Sr-90.

NOTE: All units are in Ci/L unless otherwise indicated.

CC = Complexant concentrate

NA = Not analyzed

NR = Not reported

Table 3-4. Unit Liter Release Dose Data for Offsite Individual without/with Plume Meander.

Offsite	EDE (rem/L) solids	EDE (rem/L) liquid
Inhalation without PM (CF=1)	2.30E-2	1.80E-5
Inhalation with PM (CF=0.64)	1.47E-2	1.15E-5

CF = Conservative multiplication factor

EDE = Effective dose equivalent

PM = Plume meander

Table 3-5. Receptor Location Conversion Factors. (a)

To convert from data for offsite individual, multiply by	To obtain
1	99.5% maximum offsite individual
0.64	99.5% maximum offsite individual with PM
1,900	99.5% maximum onsite individual
610	99.5% maximum 100-m onsite individual with PM
150	50% overall 100-m onsite individual
120	50% overall 100-m onsite individual with PM
0.044	50% overall offsite individual
0.042	50% overall offsite individual with PM
2,500	99.5% maximum 50-mile population
0.036	99.5% average 50-mile individual
30	50% overall site total 50-mile population
0.000083	50% overall site average 50-mile individual

(a) Source: WHC-SD-WM-ISB-001 (Leach and Stahl 1993).

PM = Plume meander

Table 3-6. Unit Liter Release Dose Calculation for Onsite Individual without Plume Meander.

Onsite	EDE (rem/L) (solids) (CF=1,900)	EDE (rem/L) (liquid) (CF=1,900)
Inhalation	4.37E+1	3.42E-2

CF = Conservative multiplication factor

EDE = Effective dose equivalent

Table 3-7. Unit Liter Release Dose Calculation for Onsite Individual with Plume Meander.

Onsite	EDE (rem/L) (solids) (CF=610)	EDE (rem/L) (liquid) (CF=610)
Inhalation	1.40E+1	1.10E-2

CF = Conservative multiplication factor

EDE = Effective dose equivalent

Table 3-8. Tank 241-C-106 Chemical Inventory Data. (a)

Sample type	Sludge			Supernate			Tank inventory (b)
Sample date	4/1/90 (c)			--			--
Sample identification	RAT-C106-4A			--			--
Density (g/ml)	1.75			--			--
Volume (kgal)	181			32			213
Volume (L)	6.85E+05			1.21E+05			8.06E+5
Units	ug/g	g/L	g	--	g/L	g	g
Ag	2.10E+03	3.68E+00	2.52E+06	--	--	--	2.58E+06
Al	1.15E+05	2.01E+02	1.38E+08	--	--	--	1.38E+08
As	3.90E+02	6.83E-01	4.68E+05	--	--	--	4.68E+05
B	3.50E+02	6.13E-01	4.20E+05	--	--	--	4.20E+05
Ba	3.50E+02	6.13E-01	4.20E+05	--	--	--	4.20E+05
Ca	1.90E+03	3.33E+00	2.28E+06	--	--	--	2.28E+06
Ce	4.90E+02	8.58E-01	5.88E+05	--	--	--	5.88E+05
Cr	1.75E+03	3.06E+00	2.10E+06	--	--	--	2.10E+06
Cu	3.00E+02	5.25E-01	3.60E+05	--	--	--	3.60E+05
Fe	8.24E+04	1.44E+02	9.88E+07	--	--	--	9.88E+07
Mg	4.20E+02	7.35E-01	5.04E+05	--	--	--	5.04E+05
Mn	4.50E+03	7.88E+00	5.40E+06	--	--	--	5.40E+06
Na	1.56E+05	2.73E+02	1.87E+08	--	--	--	1.87E+08
Nd	5.60E+02	9.80E-01	6.71E+05	--	--	--	6.71E+05
P	7.30E+03	1.28E+01	8.75E+06	--	--	--	8.75E+06
Pb	2.50E+03	4.38E+00	3.00E+06	--	--	--	3.00E+06
Sb	2.90E+03	5.08E+00	3.48E+06	--	--	--	3.48E+06
Se	1.40E+03	2.45E+00	1.68E+06	--	--	--	1.68E+06
Si	2.60E+04	4.55E+01	3.12E+07	--	--	--	3.12E+07
Sr	1.10E+02	1.93E-01	1.32E+05	--	--	--	1.32E+05
Ti	3.30E+02	5.78E-01	3.96E+05	--	--	--	3.96E+05
Tl	1.20E+03	2.10E+00	1.44E+06	--	--	--	1.44E+06
Zr	6.00E+02	1.05E+00	7.19E+05	--	--	--	7.19E+05

- (a) Source: WHC-SD-WM-TI-565 (Van Vleet 1993b).
 (b) Denotes sludge volume only.
 (c) Date of document.

Table 3-9. Tank 241-AY-101 Chemical Inventory Data. (a)

Sample type	Supernatant			Slurry/sludge			Tank inventory (b)
Sample date	07/20/88			07/20/88			07/20/88
Sample identification	R-8371			--			--
Reference	WHC I.M. 12221-PCL88-169			--			--
Density (g/Ml)	1.09			--			--
Volume (kgal)	853			83			853
Volume (L)	3.23E+06			3.14E+05			3.54E+06
Units	--	g/L	g	--	g/L	g	g
Al(OH)4-	--	4.64E+00	1.50E+07	--	--	--	1.50E+07
CO3--	--	1.84E+01	5.93E+07	--	--	--	5.93E+07
Cl-	--	4.40E-01	1.42E+06	--	--	--	1.42E+06
F-	--	7.60E-01	2.45E+06	--	--	--	2.45E+06
K	--	3.10E-01	1.00E+06	--	--	--	1.00E+06
NH3	--	2.79E+00	9.01E+06	--	--	--	9.01E+06
NO2-	--	1.36E+01	4.38E+07	--	--	--	4.38E+07
NO3-	--	3.84E+01	1.24E+08	--	--	--	1.24E+08
Na	--	4.48E+01	1.45E+08	--	--	--	1.48E+08
OH-	--	8.06E+00	2.60E+07	--	--	--	2.60E+07
PO4---	--	7.70E-01	2.49E+06	--	--	--	2.49E+06
SO4--	--	4.83E+00	1.56E+07	--	--	--	1.56E+07
TOC	--	4.47E+00	1.44E+07	--	--	--	1.77E+07

(a) Source: WHC-SD-WM-TI-543 (Van Vleet 1993a).

(b) Supernate only.

Table 3-10. Tank 241-AY-102 Chemical Inventory Data. (a)
(Sheet 1 of 2)

Sample type	Supernatant			Comp solids/sludge			Tank inventory
Sample date	2nd Qtr 1988			2nd Qtr 1988			2nd Qtr 1988
Sample identification	102-AY Seg 1			102-AY Comp			--
Reference	PNL Letter Rpt 9000855 (b) (2/12/90)			PNL Letter Rpt 9000855 ^b (2/12/90)			--
Density (g/ml)	1.0			1.4			--
Volume (kgal)	818 (June 1988)			32 (June 1988)			850
Volume (L)	3.10E+06			1.21E+05			3.22E+06
Units	M	g/L	g	mmol/g	g/L	g	g
Ag	6.40E-05	6.90E-03	2.14E+04	6.70E-02	1.01E+01	1.23E+06	1.25E+06
Al	5.90E-05	1.59E-03	4.93E+03	1.60E+00	6.04E+01	7.32E+06	7.33E+06
B	7.70E-04	8.33E-03	2.58E+04	4.40E-01	6.67E+00	8.07E+05	8.33E+05
Ba	2.90E-06	3.98E-04	1.23E+03	1.50E-02	2.88E+00	3.49E+05	3.51E+05
Ca	1.10E-04	4.41E-03	1.37E+04	4.20E-01	2.36E+01	2.85E+06	2.87E+06
Cd	<1.0 0E-06	<1.12E-04	<3.48E+02	4.00E-03	6.29E-01	7.63E+04	7.66E+04
Ce	<4.00E-06	<5.61E-04	<1.74E+03	8.60E-03	1.69E+00	2.04E+05	2.06E+05
Cl-	1.70E-02	6.03E-01	1.87E+06	2.40E-01	1.19E+01	1.44E+06	3.31E+06
Cr	2.50E-04	1.30E-02	4.03E+04	7.20E-02	5.24E+00	6.35E+05	6.75E+05
Dy	<1.00E-07	<1.63E-05	<5.03E+01	--	--	--	<5.0 3E+01
F-	9.30E-03	1.77E-01	5.47E+05	1.80E-01	4.79E+00	5.80E+05	1.13E+06
Fe	7.20E-06	4.02E-04	1.25E+03	1.60E+00	1.25E+02	1.52E+07	1.52E+07
Hg	--	--	--	3.80E-04	1.07E-01	1.29E+04	1.29E+04
K	1.10E-03	4.30E-02	1.33E+05	5.90E-02	3.23E+00	3.91E+05	5.24E+05
La	<4.00E-07	<5.56E-05	<1.72E+02	3.00E-02	5.83E+00	7.07E+05	7.07E+05
Li	<1.00E-05	<6.94E-05	<2.15E+02	--	--	--	<2.15E+02
Mg	2.30E-05	5.59E-04	1.73E+03	3.00E-01	1.02E+01	1.24E+06	1.24E+06
Mn	<2.00E-07	<1.10E-05	<3.40E+01	1.70E-01	1.31E+01	1.58E+06	1.58E+06
Mo	4.30E-06	4.13E-04	1.28E+03	2.00E-03	2.69E-01	3.25E+04	3.38E+04
NO2-	2.50E-02	1.15E+00	3.56E+06	1.10E-01	7.09E+00	8.58E+05	4.42E+06
NO3-	4.40E-03	2.73E-01	8.45E+05	1.70E-02	1.48E+00	1.79E+05	1.02E+06
Na	9.60E-02	2.21E+00	6.86E+06	1.80E+00	5.79E+01	7.02E+06	1.39E+07
Nd	<7.00E-07	<1.01E-04	<3.13E+02	1.90E-02	3.84E+00	4.65E+05	4.65E+05
Ni	<2.00E-06	<1.17E-04	<3.64E+02	5.60E-02	4.60E+00	5.58E+05	5.58E+05

Table 3-10. Tank 241-AY-102 Chemical Inventory Data. (a)
(Sheet 2 of 2)

Sample type	Supernatant			Comp solids/sludge			Tank inventory
Reference	PNL Letter Rpt 9000855 (b) (2/12/90)			PNL Letter Rpt 9000855 ^a (2/12/90)			--
Density (g/ml)	1.0			1.4			--
Volume (kgal)	818 (June 1988)			32 (June 1988)			850
Volume (L)	3.10E+06			1.21E+05			3.22E+06
Units	M	g/L	g	mmol/g	g/L	g	g
P	1.00E-04	3.10E-03	9.59E+03	2.30E-01	9.97E+00	1.21E+06	1.22E+06
P04---	5.50E-04	5.22E-02	1.62E+05	2.40E-02	3.19E+00	3.87E+05	5.48E+05
S04--	1.10E-03	1.06E-01	3.27E+05	8.10E-03	1.09E+00	1.32E+05	4.59E+05
Si	5.90E-03	1.66E-01	5.13E+05	4.30E-01	1.69E+01	2.05E-06	2.56E-06
Sr	2.60E-06	2.28E-04	7.05E+02	9.00E-03	1.10E+00	1.34E+05	1.34E+05
TOC	2.40E-02	2.88E-01	8.92E+05	3.82E+00	6.42E+01	7.78E+06	8.67E+06
Te	<2.00E-06	<2.55E-04	<7.90E+02	4.00E-03	7.15E-01	8.66E+04	8.74E+04
Ti	<4.00E-07	<1.92E-05	<5.93E+01	7.10E-03	4.76E-01	5.77E+04	5.77E+04
U	3.50E-03	8.33E-01	2.58E+06	6.50E-02	2.17E+01	2.62E+06	5.20E+06
Zn	5.00E-06	3.27E-04	1.01E+03	8.10E-03	7.41E-01	8.98E+04	9.08E+04
Zr	7.60E-07	6.93E-05	2.15E+02	6.30E-03	8.05E-01	9.75E+04	9.77E+04

(a) Source: WHC-SD-WM-TI-543 (Van Vleet 1993a).

(b) Source: PNL Letter Report 9000855 (Scheele et al. 1990).

3.2 DESCRIPTION OF NONRADIOLOGICAL HAZARDOUS MATERIALS

In addition to the radiological hazards, the waste solutions may be corrosive. The design criteria are 7 to 13 Ph. This level of caustic presents a moderate nonradiological hazard. Waste solutions out of the design range for Ph levels present a hazard to the stainless steel or carbon steel materials used for construction of the waste transfer lines and storage tanks. Controls in place for radiological concerns will ensure reasonable safety with respect to the caustic solutions. The tanks also have nonradioactive elements present. Controls in place for radiological concerns ensure reasonable safety with respect to mixed waste. See Tables 3-1, 3-2, and 3-3 for the radiological elements, and Tables 3-8, 3-9, and 3-10 for the nonradiological elements contained in C-106, AY-101, and AY-102, respectively.

3.3 DESCRIPTION OF AIRSTREAM CHARACTERIZATION

Tank C-106 airstream characterization is provided in Appendix D of Bailey 1994. Confinement is achieved by ventilation control and by directing air from uncontaminated areas toward areas of higher contamination. The confinement ventilation system will incorporate those components required to mitigate unacceptable environmental releases.

3.4 DESCRIPTION OF TOXIC MATERIALS

The hazards or the safety issues caused by tank toxic gas emissions are not fully characterized as noted in WHC-SD-WM-ISB-001 (Leach and Stahl 1993). As noted further in WHC-SD-WM-ISB-001, significant concern over the release of toxic gas from the tanks (other than C-103 and SY-101) has been shown only recently. Guidance provided in Section 2.4 and the topical report for toxilogical consequences of WHC-SD-WM-ISB-001, will be used to determine the risks from toxic materials (see Section 4.2.3 of this report).

4.0 ACCIDENT EVALUATION

The principal hazards of concern in operation of the C-106 WRSS (Project W-320) are associated with radioactive material releases. The hazards associated with electrical, mechanical, or thermal energy are similar to those for other normal hazardous industrial operations (Figure 4-1).

4.1 PRELIMINARY HAZARDS EVALUATION

A preliminary hazards assessment was conducted by an assessment team to identify potential hazards for the WRSS (see Table 4-1). The events were further evaluated by establishing a potential accident and sequence, establishing potential target/consequences, mitigating features (engineered and administrative), and qualitatively estimating the probability of occurrence. The hazardous event evaluation provides a basis for establishing barriers for mitigating any hazardous material releases to the environment. The hazards assessment established the highest consequence level for each event. Worst-case levels were evaluated with respect to probability and consequence level. This preliminary safety evaluation considers releases of hazardous material during (1) normal operations, (2) off-normal but anticipated conditions, and (3) accident conditions discussed in this section. In addition, concerns about temperature anomalies in C-106 and potential steam releases to the environment are also investigated. As discussed in Section 8.0, these hazards will be evaluated during the preparation of the SA.

During normal transfer operations, no releases above limits are expected. During jumper change operations, small residual amounts of radioactive material are expected to be available for release. Significant leaks within the pits will be detected and returned to the tank through pit drains. Small leaks that result in an accumulation of solution on the floor of the pit will be detected when a sufficient amount is collected. The pit will use lids and maintain a low negative pressure to prevent release of any airborne radioactivity from the pit to the atmosphere during operation. Administrative controls will require all covers for pits to be in place, in addition to compliance with lock and tag procedures, before any transfer. Spray and washdown systems are incorporated into the design to reduce any contamination in the pit before opening them for any maintenance activities.

Leaks in the primary piping system of the transfer lines will be controlled by the secondary containment (encasement pipe). The secondary containment system is designed to collect released waste at a common point for detection and removal.

Unplanned releases can occur if accidents or natural forces cause a breach of the primary and secondary confinement. The accidents that are considered involve mechanical failures or human error. The natural forces that are considered include earthquake, high wind, and freezing weather.

Figure 4-1. Accident Evaluations.

Table 4-1. Preliminary Hazards Analysis: Accidents and Consequences. (Sheet 1 of 5)

Event	Potential accident and sequence	Potential consequences	Mitigative features		Probability of occurrence (Ref. provided or Estimated) (a)
			Engineered	Administrative	
1.0 GENERAL					
Nearby tank release	Locally unacceptable plume of toxic and radiological gases	Personnel contamination/exposure	Existing tank ventilation system	Work procedures, area radiation monitoring, continuous air monitoring, personnel protective equipment	E
Excavation of site for modification of pits, riser, etc.	Riser is struck	Breaches primary containment	HVAC system in operation, radiation monitors	Work procedures, hand excavation is required	E
Tank dome failure	Construction equipment and permanently installed equipment overloads tank dome; structural collapse	Loss of primary and secondary confinements, large liquid and airborne releases	Analysis of acceptable loading	Control of construction activities over dome (b)	I
High winds damage	Support structures and utilities subjected to high winds and associated debris and support utilities affected	Loss of tank support systems and ventilation confinement	Support structures and utilities are designed to HPS-SDC 4.1 (c)	The peak gust wind measured at Hanford Site was 80 mi/h	U
Thunderstorm damage	Lightning strikes and causes loss of electrical power	Loss of tank ventilation and system shutdown	System goes into passive state; shut down HVAC system; backup generator and/or UPS automatically power to ventilation system being evaluated	Following restoration of power, sluicing system can be started with necessary precautions per work procedures	E

Table 4-1. Preliminary Hazards Analysis: Accidents and Consequences. (Sheet 2 of 5)

Event	Potential accident and sequence	Potential consequences	Mitigative features		Probability of occurrence (Ref. provided or Estimated) (a)
			Engineered	Administrative	
Freezing damage	Dormant piping system may freeze and shut down system operation	Pipe may rupture, and radioactive material may be released to soil and environment	Design for freeze, proper insulation provided for piping including ventilation duct	Mild winters at Hanford Site; selected equipment protected; emergency procedures in place during winter	E
Corrosion of waste transfer system	Chemicals corrode primary transfer piping, ventilation duct, and leak occurs	Leak to confinements; premature loss of transfer piping and ventilation duct; damage to confinement barriers	Tank contents are characterized and monitored before transfer and during system operation; materials of construction being evaluated	Administrative control; tank contents are monitored through out the life of system operation	E
Excess of water addition to the tank	Excessive flush sequence	Tank high level, airborne release, liquid release	Flow and level instrumentation	Controls over water addition per work procedures	E (excessive water addition)
	External flood				I (external flooding)
Nuclear criticality	Accumulation of critical mass material	Environmental release, personnel radiation exposure, loss of system/tank usage	Criticality analysis performed	Administrative controls and proper transfer procedures	<1.0E-06 (d)
2.0 WASTE TRANSFER SYSTEM					
Waste transfer line (or recirculation line) breaks	Accident is due to an earthquake	Radioactive material released to soil, loss of primary piping, release to atmosphere	Piping is buried 0.3 m (1 ft) below grade. A berm of 0.9 m (3 ft) over the pipe is provided.	Health physics technician will post the areas appropriately	7.0E-04 (d)

Table 4-1. Preliminary Hazards Analysis: Accidents and Consequences. (Sheet 3 of 5)

Event	Potential accident and sequence	Potential consequences	Mitigative features		Probability of occurrence (Ref. provided or Estimated) (a)
			Engineered	Administrative	
Waste transfer line (or recirculation line) leaks	Pipe ruptures due to corrosion, blockage, etc., and leaks into encasement pipe	Radioactive material released to secondary confinement, loss of primary piping	Secondary containment	Administrative controls, leak detection	E
Waste transfer line secondary confinement leaks	Both primary and secondary confinement ruptured due to corrosion, etc., and leaks to the environment	Radioactive material released to soil	Leak detection instrumentation is provided at the secondary confinement end point	Work procedures and operating procedures	I
Misrouting or jumper misalignment	Waste is not received by the receiving tank	Loss of tank use	Verification of liquid level in the receiver tank	Administrative controls	E
Load Drop	a. Cover block drop	Piping/riser damaged, penetration into the tank, tank waste inventory increased if equipment is dropped in the tank, potential liquid and/or airborne release to the environment	Certified lifting equipment, each lift is reviewed; analysis for tank leaks has been completed	Rigging and work procedures will be used	E
	b. Equipment drop				E
	c. Tools/material drop				E
Valve misalignment	Waste is misrouted and possibly dilutant pumped forward to transfer line	Loss of proper tank usage; waste inventory increased	Valve alignment procedures in place before startup	Administrative and operating procedures	E
Failure of nozzle angle adjustment arm remote control	Installation or operator error, nozzle angle adjustment arm remote control not in proper position	Damage to tank wall by jet impingement; loss of tank usage; potential liquid release to the environment	Analysis for tank leaks has been completed (e)	Operating and work procedures in place	E

Table 4-1. Preliminary Hazards Analysis: Accidents and Consequences. (Sheet 4 of 5)

Event	Potential accident and sequence	Potential consequences	Mitigative features		Probability of occurrence (Ref. provided or Estimated) (a)
			Engineered	Administrative	
3.0 VENTILATION SYSTEM					
Ventilation duct breach	a. Due to vehicular accident b. Natural causes like earthquake c. Due to flammable gas	Loss of ventilation, tank contents and tank heat up due to loss of cooling, C-106 will return to the original condition that existed before WRSS implementation (i.e., evaporative cooling); potential airborne release to the environment	Operating and plant procedures; add cooling water to tank to maintain temperature within the tank; area radiation monitors, if available, will provide indication of radioactive release	Work procedures, health physics technician available, personnel protection, surveillance procedure	1.0E+00 (estimated) 7.0E-04 (e) 3.0E-05 (e)
Filter skid failure	a. Filter fails due to high Dp and failure to transfer to backup filter b. Chemical attack bonding or seal around filter causing leaks	Environmental release	Pressure across filter monitored and filter is changed before high Dp occurs	Monitoring of stack exhaust would indicate HEPA filter leak. Pressure gauges/DOP checks across each HEPA filter, and also change the affected filter	U
HEME plugged	HEME allows moisture to pass and HEPA filters become moisture laden	HEPA filter failure followed by environmental release	Dp for each HEPA filter is monitored; HEME is flushed	HEPA efficiency reduced; monitoring of stack will ensure releases are within allowable limit; administrative controls	U
Fan motor failure	Bearing seized, and fan stopped	Loss of ventilation	Fan motor is monitored for drive trouble	HVAC and sluicing operation will be shut down; fan will be replaced	A

Table 4-1. Preliminary Hazards Analysis: Accidents and Consequences. (Sheet 5 of 5)

Event	Potential accident and sequence	Potential consequences	Mitigative features		Probability of occurrence (Ref. provided or Estimated) (a)
			Engineered	Administrative	
Fan fire	Electrical short ignites motor, motor windings short, and fan stops	Loss of ventilation	Fan motor is monitored for drive trouble	HVAC and sluicing operation will be shutdown; fan will be replaced	A

- (a) Probabilities are taken from WHC-CM-4-46, Section 7.0: A = Anticipated; U = Unlikely; E = Extremely unlikely; I = Incredible (see Table 4-2).
- (b) WHC-SD-WM-ES-244 (Estey 1993).
- (c) HPS-SDC-4.1, "Standard Architectural Civil-Design Criteria: Design Loads for Facilities" (DOE-RL 1989b).
- (d) WHC-SD-WM-SEL-027 (Kidder 1993b).
- (e) WHC-SD-WM-ES-218 (Lowe 1993).

DOP = Dioctyl phthalate
 Dp = Differential pressure
 HEME = High-efficiency mist eliminator
 HEPA = High-efficiency particulate air
 HVAC = Heating, ventilating, and air conditioning
 UPS = Uninterruptible power supply

Table 4-2. Estimated Probability of Occurrence.

Probability category	Category description	Frequency
Anticipated	An abnormal event that is expected to occur once or more during the lifetime of the facility (e.g., small radioactive materials spills, small fires).	1E-2 to 1
Unlikely	Individually, the condition is not expected to occur during plant lifetime, but collectively, events in this category may occur several times.	1E-4 to 1E-2
Extremely unlikely	Extreme low-probability conditions that are not expected during the plant lifetime but that represent extreme or limiting cases of faults identified as possible. This category includes design basis accidents (i.e., design basis earthquakes).	1E-6 to 1E-4
Incredible	Accident for which no credible scenario can be identified.	<1E-6

4.2 DISCUSSION OF ACCIDENTS

This section includes discussions on the following classes of accidents:

- Physical
- Chemical

These discussions provide a clear statement of the class of accident and the phenomena that comprises the accident being evaluated to include radiological, chemical, and toxicological consequences.

4.2.1 Radiological Consequences

Accidents that were evaluated involve loss of radioactive materials confinement during waste transfer, and loss of confinement of radioactive and hazardous materials during storage operations. The only important sources of accidental energy are pumping energy and gravity. The volume of material evaluated in the transfer system is small compared to the large volume of material in the storage tank system.

Four physical accident scenarios are evaluated in detail (see Appendix A): a large-volume spill caused by a waste transfer line rupture, an unfiltered release to the environment via a recirculation HVAC duct, a release to the environment because of jumper or connector failure (spray leak) with the cover block not in place, and a release to the environment because of a jumper or connector failure (spray leak) with the cover block in place. The annual probability (frequency) of a DBE at the Hanford Site is 7E-4/yr (Boyles 1989). For the purposes of the accident analysis, seismic event or vehicular accident is considered for the worst-case accident scenario. In addition, the analysis in Appendix A is based on the radioactive inventory data presented in WHC-SD-WM-TI-543 (Van Vleet 1993a), and WHC-SD-WM-TI-565 (Van Vleet 1993b). As noted in Bander 1993a and 1993b, the radiogenic heat load is 32.2 kW

(110,000 Btu/h). The calculated heat load due to radioactive elements in C-106 based on data from Van Vleet 1993b is dominated by decay of ^{90}Sr and ^{137}Cs with negligible contribution from other radionuclides, and is equivalent to 13.4 kW; 12 kW from ^{90}Sr and 1.4 kW from ^{137}Cs . The radioactive inventory data in Van Vleet 1993b are suspect values according to Bander 1993a and 1993b. This introduces a scaling factor of 2.4 due to the discrepancy between radiogenic heat load in the Bander reports and the estimated heat load from the current radioactive inventory in C-106. Characterization of C-106 is scheduled to be completed in the near future. Until such time as the new characterization data for C-106 is available, the 2.4 scaling factor has to be applied to the estimated EDE values in Appendix A. According to the guidelines in WHC-CM-4-46, the EDE values for postulated accidents are used for determination of safety classification of WRSS and radiological risk acceptance. The risk is based on the frequency of occurrence of accidents and the consequence.

4.2.1.1 Waste Transfer Line Break. It is postulated that waste transfer lines could breach during transfer operation because of an earthquake since both transfer lines are buried in the same trench and bermed. It is also assumed that the seismically induced guillotine break identified at the location to the entrance to the sluice pit 241-C-106 is bounding, i.e., this is the lowest elevation at which both transfer lines will be fully loaded with waste material. The transfer lines are buried 0.3 m (1 ft) below grade with a berm. A minimum 0.9 m (3 ft) cover over the pipe will be provided for freeze protection and shielding. Based on the failure probability of $1\text{E-}9$ hr/ft (Garza 1992), 610 m (2000 ft) of pipe, and an operating period of 24 hours per day for one year, the probability is estimated to be $1.7\text{E-}2/\text{yr}$. The probability of failure of both lines is estimated to be $3.0\text{E-}4/\text{yr}$. The annual probability (frequency) of occurrence of a DBE, equivalent to safe shutdown earthquake (SSE), of 0.2 g horizontal ground acceleration at the Hanford Site is $7.0\text{E-}4$ (Kidder 1993b). Because of the environmental impact as noted in Appendix A, Scenarios 1a through 1d, a safety class system is required to protect the environment. The buried waste transfer lines with mitigative features, such as use of a berm and barricades, will potentially eliminate the possibility of vehicular (backhoe excavating into the line) accident and line breach occurrence.

Radioactive leak from a transfer line break with 0.9 m (3 ft) cover, will reach the above grade level surface and will form a pool near the break. The quantity of leak that forms into a pool at the surface is estimated to be 10% of the total quantity that is postulated to leak during the stated release time. This is based on a buried line break incidence at 242-S evaporator and its accident evaluation as noted in Section 9.1 of WHC-SD-WM-SAR-023 (Lavender 1993). A similar approach will be used for estimating leak for C-106 waste retrieval project, i.e., 10 percent will leak to form a pool at the surface from a total quantity of the liquid that is postulated to leak into soil during the entire release time period. Further, the buried piping configuration and the process conditions for the W320 project is similar in configuration and/or less severe in process conditions as compared to the 242-S buried lines.

The amount of particles from a radioactive leak that would be available to the offsite individual is low, e.g., 10% of C-106, 90% of AY-102; 30% of C-

106, 70% of AY-102. If 10% of the amount in the pipe breaks makes it to the surface, only a small amount of this is available to the offsite individual; the major part of the substance that does make it to the surface for the pool is the salts formed when C-106 and AY-102 combined. Even if some material pools there will be a tendency for the liquid to percolate down along with all the other that is going down.

To estimate the amount of waste material released, a release fraction equal to $1.0\text{E-}5$ (Mishima 1993) is used. It is assumed that the liquid/particle agglomeration will not be carried to the offsite individual, i.e., the drying will force the formation of these agglomerations which will not be so readily available to the offsite individual.

Environmental impact is significantly reduced by use of a seismic switch to shutdown the WRSS, as noted in Appendix A, Scenarios 1c and 1d. The safety class transfer lines and seismic switch have been specified to protect the environment.

The EDE values with seismic switch shutoff and ten second release time, as calculated in Appendix A, and multiplied by the scaling factor of 2.4 are as follows:

EDE dose consequence for offsite individual for both solids and liquids,
 $4.2\text{E-}5 \text{ rem} \times 2.4 = 1.0\text{E-}4 \text{ rem}$;

EDE dose consequence for onsite individual for both solids and liquids,
 $8.0\text{E-}2 \text{ rem} \times 2.4 = 1.9\text{E-}1 \text{ rem}$.

Therefore, the offsite and onsite radiological risks are within the acceptance guidelines of Table 4-3 with the mitigative features previously described, and provide assurance that the low probability of a seismic event at the Hanford Site is an acceptable risk.

Table 4-3. Radiological Risk Acceptance Guidelines. (a)

Frequency category	Frequency range	Effective dose equivalent (rem)	Organ dose equivalent for lens of eye (rem)	Organ dose equivalent for all other organs (rem)
Offsite guidelines				
Anticipated	1 to $1\text{E-}2$	0.01 to 0.5	0.03 to 1.5	0.1 - 5
Unlikely	$1\text{E-}2$ to $1\text{E-}4$	0.5 - 4	1.5 - 12	5 - 40
Extremely unlikely	$1\text{E-}4$ to $1\text{E-}6$	4 - 25	12 - 75	40 - 250
Onsite Guidelines				
Anticipated	$1\text{E-}1$ to $1\text{E-}2$	1 to 5	3 - 15	10 - 50
Unlikely	$1\text{E-}2$ to $1\text{E-}4$	5 - 25	15 - 75	50 - 250
Extremely unlikely	$1\text{E-}4$ to $1\text{E-}6$	25 - 100	75 - 300	250 - 1,000

(a) Section 7.0, WHC-CM-4-46

4.2.1.2 HVAC Recirculation Duct Break. It is postulated that a recirculation duct for HVAC may fail during WRSS operation because of a vehicular accident. HVAC is not required for safe shutdown of WRSS, i.e., if HVAC is shutdown or otherwise, tank will be on passive mode with no forced cooling. The recirculation ventilation flow is 860 scfm (see Figure 1-4), the vapor space capacity is 100 mg/m³ (Bailey 1994) and the vapor SpG has been taken as equivalent to 1.0 for calculation purposes since vapor space is 100% humid and 90% supernatant is in the vapor space according to Bailey 1994. Because of the ALARA concerns noted in Appendix A, Scenario 2, a safety class system is required to protect the environment.

Based upon a one hour release, that is equivalent to a volume release of 0.15 L, the offsite and onsite EDE dose consequences, as calculated in Appendix A and multiplied by the scaling factor of 2.4, with plume meander (PM) are as follows:

EDE dose consequence for offsite individual is $2.2\text{E-}4 \text{ rem} \times 2.4 = 5.3\text{E-}4 \text{ rem}$;

EDE dose consequence for onsite individual (without PM effect) is $2.1\text{E-}1 \text{ rem} \times 2.4 = 5.0\text{E-}1 \text{ rem}$.

The HVAC system with mitigative features such as fence barricades around C-106 facility along with administrative controls will potentially eliminate the possibility of a vehicular accident and duct break occurrence. In addition, the seismic switch will shutdown the HVAC system to mitigate the consequences due to a duct break during a seismic event. Therefore, the radiological risks are within the acceptance guidelines of Table 4-3 with the mitigative features described herein.

4.2.1.3 Spray Leak from a Jumper/Connector with Pit Cover Off. A spray leak at a jumper/connector in a pit caused by an improper connection or mechanical failure of a gasket at the connection could cause a significant aerosol release to the atmosphere during a transfer of radioactive liquid waste with the pit cover block not in place. The radiological consequences to the offsite and onsite individual, as noted in Appendix A, Scenario 3a, are considered unacceptable according to the radiological risk acceptance guidelines of WHC-CM-4-46.

4.2.1.4 Spray Leak from a Jumper/Connector with Pit Cover On. It is postulated that the jumper or connector is breached; the pit drain is blocked; the pit is pressurized; the waste liquid is postulated to leak through a 5 cm (2 in.) valve handle opening as an atomizing spray with the pit cover on, and the release is assumed to be eight hours, equivalent to one shift.

The mechanism for dispersing the aerosol from the pit to the environment is through the leak paths around the cover blocks. The volume of waste is calculated on the basis that the pit drains are plugged and the leak detection system has failed to detect leaks accumulated in the pit. The leak is modeled as the discharge from a sharp-edged orifice of size 5 cm (2 in.) diameter opening as an atomizing spray. The discharge rate through the orifice is calculated as follows:

$$Q = 0.61 A \sqrt{2 g h}$$

Where

Q = Discharge rate, m³/sec
 0.61 = Discharge coefficient for a sharp edged orifice
 A = Area of opening, m²
 g = Gravitational acceleration, 9.8 m/sec²
 h = system pressure (pumping pressure) expressed as water head, m
 1 psig = 27.7 inches of water.

Therefore, $h = 180 \text{ psig} \times 27.7 \text{ inches/psig} \times 2.54 \text{ cm/inch} \times 1 \text{ m/100 cm}$
 $= 1.27\text{E}+2 \text{ m}$

and 2 inch square edged
 opening area, $A = \pi \times \{(2.0 \text{ inch} \times 2.54 \text{ cm/inch})^2 / 4\} \times (1 \text{ m/100 cm})^2$
 $= 2.03\text{E}-3 \text{ m}^2$

Substituting in the above equation,

$$Q = 0.062 \text{ m}^3/\text{sec}$$

The displacement rate of moist air from the pit is taken to be equal to the discharge rate from the leak. With the release time of 8 hours, which is equivalent to the length of one shift,

The amount of waste volume released
 to the environment is $= 0.062 \text{ m}^3/\text{sec} \times 8 \text{ hrs} \times 3600 \text{ sec/hr}$
 $= 1.77\text{E}+3 \text{ m}^3$

The EDE dose consequences, estimated in Appendix A and multiplied by the scaling factor of 2.4, are as follows:

Offsite individual with PM for both solids and liquid is
 $6.5\text{E}-5 \text{ rem} \times 2.4 = 1.6\text{E}-4 \text{ rem};$

Onsite individual with PM for both solids and liquid is
 $6.2\text{E}-2 \text{ rem} \times 2.4 = 1.5\text{E}-1 \text{ rem}.$

The mitigating features of leak detection and monitoring, pit drains, covered negative pressurized pits, and administrative controls will reduce the probability of a spray leak to the atmosphere. A sequence of activities that are required as a minimum to ensure the cover block is in place before starting the transfer operation is provided in Figure 4-2.

Figure 4-2. Sequence of Activities for Jumper Installation or Other Modifications Within a Pit.

The probability of failure of a gasket at a jumper connection is estimated to be $3\text{E-}5/\text{hr}$ or $2.6\text{E-}1/\text{yr}$ according to WHC-IP-690 (Garza 1992). Operator (human) errors and/or procedural violations at critical steps have been considered in the preceding analysis. Operator or human error failure is estimated to be of the order of $1\text{E-}2/\text{yr}$ according to Appendix III, Section 6.0, Reactor Safety Study, NUREG-75/014 (NUREG 1975). The probability of failure of sequence of activities due to operator actions is as follows: installation of cover blocks require a minimum of two crane operators (failure probability is $1\text{E-}2 \times 1\text{E-}2 = 1\text{E-}4/\text{yr}$), and verification by the quality control inspector to ensure installation of the cover block (failure probability is $1\text{E-}2/\text{yr}$). The resulting failure probability is equivalent to $1\text{E-}6/\text{yr}$. When this is combined with a jumper connection failure probability of $2.6\text{E-}1/\text{yr}$, it will result in an overall probability of the order of $2.6\text{E-}7/\text{yr}$. This low probability is considered beyond the design basis accident probability of occurrence.

The estimated annual probability with mitigation will result in the "unlikely" category of occurrence per Table 4-2. Offsite and onsite radiological risks are within the acceptance guidelines of Table 4-3 with the mitigative features described previously. The risk acceptance criteria of WHC-CM-4-46, Section 4.0, provide assurance that the annual probability of occurrence is an acceptable risk with mitigative features, and implementation of operational safety requirements for the pit cover to be in place during transfers (see WHC-SD-WM-QSR-005, Section 3.6.1 [Dougherty 1993]). The requirement also will include that the pit covers are not easily moved once they are in place.

4.2.2 Chemical Consequences

The compatibility report (Sederburg 1994) evaluates the chemical compatibility of waste currently stored in C-106, AY-101, and AY-102. Specifically, this report describes the acceptability to transfer these wastes in the following sequence:

1. AY-102 supernatant to any other tank
2. AY-101 supernatant or an alternate fluid as stated in ES-290 (Sederberg) to AY-102
3. Sluice C-106 to AY-102, using the supernatant of AY-102 as the sluicing media.

The following safety criterion from WHC-SD-WM-DQO-001, *Data Quality Objectives for the Waste Compatibility Program* (Carothers 1994), were analyzed in WHC-SD-WM-ES-290 (Sederburg 1994) and are summarized below:

1. Criticality
2. Reactivity
3. Corrosivity
4. Flammable gas accumulation.

These safety criteria are based on empirical study of previous waste transfers within the tank farm system as well as the resulting behavior of the waste tanks after transfer.

4.2.2.1 Criticality. The bases for criticality safety in the waste tanks are provided by WHC-SD-SQA-CSA-20363, *CSER 94-001: Criticality Safety of Single Shell Waste Storage Tanks* (Rogers 1994). The basic principles by which Project W-320 shows criticality safety derive from the double contingency principle using fuel concentration and poison content. This implies that the waste tanks are subcritical if plutonium concentration is limited independently of the poisons in the tanks; and, the tanks are subcritical if the poison-to-plutonium ratios are sufficient, independent of the fuel concentration.

None of the sluicing operations will increase the plutonium concentration past 1.0 g/L. Additionally, the sluicing operation will not change the form of the dominant plutonium species. The highly basic waste tank environment designed to limit corrosion of tank structural materials yields a Ph well in excess of 8.0, the limit above which insoluble plutonium predominates. The waste tank chemistry specifications ensure that the dominant plutonium species in the tanks will be an insoluble hydroxide.

Waste tank criticality does not depend on geometry, mass, or plutonium distribution. The tanks are much larger than the geometry required for criticality. The tank contents are known to be heterogeneous and proving that the plutonium is nearly uniformly distributed in the tanks is difficult.

The tanks are safe by the plutonium concentration as the tanks currently exist. The maximum plutonium concentration in C-106, AY-101, and AY-102 is less than 0.2 g/L. The minimum critical concentration for plutonium in waste is 2.6 g/L (Rogers 1994).

The tanks are safe by poisons with the plutonium. If the necessary absorber-to-plutonium ratios can be demonstrated, subcriticality can be shown based solely upon that fact (Rogers 1994). The safe poison mass ratios for selected materials are 770 for natural or depleted uranium, 160 for iron, 32 for manganese, 135 for chromium, and 105 for nickel. These materials were formed as insoluble hydroxide complexes with the plutonium during waste processing. The safe poison mass ratios for selected soluble materials are 910 for aluminum, 360 for sodium, and 270 for nitrates. The safe poison mass ratio is defined such that k-effective will be less than 0.95 for any combination of water and plutonium (Rogers 1994). The solid and liquid portions of C-106, AY-101, and AY-102 meet at least one of these poison ratios. Appendix C of WHC-SD-WM-ES-290 (Sederburg 1994) lists the mass ratios normalized to plutonium for the wastes in C-106, AY-101, and AY-102.

4.2.2.2 Reactivity. The bases for chemical reactivity safety are provided by actual measure of the energetic potential of the wastes. Energetics safety is demonstrated by the absence of net exotherms in the waste conducted by laboratory thermal analyses up to 500 C° (932 °F). However, historical characterization data that support these analyses for Project W-320 do not exist. Therefore, an engineering analysis is made based on additional criteria which are judged to evaluate the same potential for adverse waste behavior.

These additional criteria have been developed by WHC-EP-0681, *Interim Criteria for Organic Watch List Tanks at the Hanford Site* (Babad and Turner 1994). The criteria are based on conservative bounding assumptions about the nature of the energetic chemical reactants present in the waste as well as additional factors which could affect an energy release. These factors are as follows:

- The waste organic concentration
- The waste moisture content
- The waste temperature.

The Waste Tank Safety program justifies that sodium acetate is energetically representative of the water-soluble organic complexants used in various Hanford Site chemical processing operations and likely to bound the organic contents of the waste tanks to be used by Project W-320. This justification is qualified in several ways. First, it does not account for the presence of process solvents (e.g., NPH and TBP) that were inadvertently sent to SSTs is unknown, but possibly significant, quantities. Second, it is believed that the organic complexants have undergone significant degradation by oxidation over the years, lowering the potential energy content of the organics. Finally, any carboxylic acids generated from contact of NPH/TBP mixtures with nitric acid, which exhibit significant reactivity with nitrate-nitrite oxidizer systems, are assumed not to be present in the waste tanks in sufficient concentrations to enhance the fuel value of the waste tank organics.

With these qualifications, testing of surrogate sodium acetate waste mixtures led to the development of criteria that specify the degree to which the energetic content of the waste poses a safety problem. Because Hanford Site waste tank organic analyses have almost exclusively been historically reported in terms of total organic carbon (TOC), TOC is equated to an energetically equivalent mass of sodium acetate ($\text{NaC}_2\text{H}_3\text{O}_2$) by the mass ratio of the sodium acetate molecule to the mass of carbon contained within it (mass of $\text{NaC}_2\text{H}_3\text{O}_2$ equals $3.4 \times$ mass of TOC). The degree of hazard caused by chemical energetics falls into three categories: safe, conditionally safe, and unsafe. These categories, specified with the organic concentration in terms of TOC mass in dry waste, are as follows:

Safe: ≤ 5 wt% TOC, and waste temperature < 149 °C (< 300 °F)

Conditionally safe: > 5 wt% TOC, and water content ≥ 17 wt%, and waste temperature ≤ 90 °C (≤ 194 °F)

Unsafe: failure to meet either of the above criterion.

The controlling document (Babad and Turner 1994) develops the analytical model used in evaluating the above energetic safety conditions. This compatibility study uses the model to evaluate the chemical energetic conditions that are posed by the sequence of Project W-320 waste processing operations. Any condition evaluated as safe or conditionally safe is equated to satisfying the energetics requirements of the waste compatibility program (Carothers 1994). An analysis of the radioactive decay heat loading in C-106

wastes and potential relationships with chemical reactivity are presented in WHC-SD-WM-ES-290, Appendix E (Sederburg 1994).

4.2.2.3 Corrosivity. The waste compatibility requirements for corrosion are provided by OSD-T-151-00007, *Operating Specifications for the 241-AN, AP, AW, AY, AZ, and SY Tank Farms* (Harris 1992). The intent is to limit the corrosion rate of tank structural materials to minimize structural damage and possible leak formation. The waste chemistry is controlled with the intent of restricting general corrosion rates to less than 1 mil/yr and to inhibit stress corrosion cracking. These criteria are based on waste temperature and the concentrations of hydroxide, nitrate, and nitrite.

4.2.2.4 Flammable Gas Accumulation. Referring to the prevention of waste formation that has the potential to trap and contain flammable gases, WHC-SD-WM-DQO-001 (Carothers 1994) states:

"The premise of the current approach is that we can use SpG of the source and receiving waste to identify transfers that may lead to flammable gas accumulation. A key consideration in future revisions to this Waste Compatibility DQO is the validation of this approach through evaluation of historical data and development of other potential indicators of flammable gas accumulation."

All tanks at the Hanford Site generate hydrogen. This hydrogen could come from corrosion, radiolysis of water, or an organic chemical reaction. The rate that hydrogen is generated in any tank is low. If the hydrogen is vented as it is made, there is no problem with hydrogen flammability. The main issue for gas flammability is the trapping of potentially flammable gases in the wastes of a tank.

The question is what parameters in tanks can be used to prevent the waste from trapping gas. It is believed that solids must be present in the tanks to trap gas. Those tanks which contain clear liquid do not present any problem with gas retention. The parameter used to identify which tanks may retain gas was chosen to be SpG. Specific gravity is a measure of waste concentration. Below about 1.35 SpG, the waste is liquid with only small amounts of solids. Once the wastes are concentrated above 1.35 SpG, chemicals start to precipitate from the waste. It was noted that all the double-shell tanks that are on the flammable gas watch list had an average SpG of greater than 1.4. None of the double-shell tanks not on the flammable gas watch list had an average SpG greater than 1.4. This seemed to be a prudent limit to set until further study could set better limits. Those studies are ongoing but not complete at this time. As of this writing, a SpG of >1.4 has been the only waste property identified that is exclusive to tanks on the flammable gas watch list, and the 1.4 SpG limit remains the most defensible discriminator yet identified.

It should be pointed out that the average SpG of the tanks to be used in the retrieval operations are all below 1.4. Mixing them will not raise the SpG beyond the 1.4 limit.

The implemented method of mitigating tank SY-101 is by mixing the waste contents. The mixing disturbs the solids and allows any trapped gas to escape. The tanks used in Project W-320 are going to be mixed during sluicing operations. The sluicing activity will consequently perform a function the same way for the wastes in C-106 and AY-102.

The ventilation system will be operating in both tanks and will sweep away any hydrogen that is released. The safety documentation for this project will look specifically at hydrogen and will address hydrogen burns and applicable administrative controls on ignition sources.

Ammonia. A definitive concentration in slurry gas has not been established, although as much as 1.3% ammonia has been reported at the tank SY-101 exhaust stack (implying a higher concentration in undiluted slurry gas). Ammonia has a flammability range in air of 16 to 25 %. The heat of reaction of ammonia with oxygen is 75.8 kcal/g-mol NH_3 (25 °C). Before the start of sluicing operations, the supernatant in AY-102 will be transferred to another tank and 2,044 Kl (588,000 gal) supernatant from AY-101 will be transferred into AY-102 for use as the sluicing medium. The ammonia concentration of this fluid, as shown in Table 3-9, is 2.79 g/L. This concentration is based on a sample taken in August 1988. The concentration of 2.79 g/L should be significantly lower in AY-101 as AWF ventilation has been in operation since that time, and it is assumed that the ventilation has removed sufficient ammonia from AY-101. Characterization of AY-101 is scheduled to be completed in the near future. Until such time as the new characterization data for AY-101 is available, it is conservative to assume and use the value of 2.79 g/L concentration for ammonia gas flammability evaluation.

As noted in Section 2.3.1, AY-102 will contain AY-101 supernatant before WRSS is started. Since AY-101 supernatant data shows a liquid phase ammonia concentration, Henry's law for ideal fluids is used for determining the vapor phase concentration of ammonia.

Henry's law is defined as $y = xH/P$,

where y = vapor phase mole fraction
 x = liquid phase molality
 P = system pressure, atm
 H = Constant (at system temperature)
 $= 2.6\text{E-}2$ atm kg/mol for NH_3 in H_2O at 95°F

Following data from Table 3-9, and performance data from Appendix D of Bailey 1994 have been used:

Supernatant temperature: 95 °F
 Specific gravity of supernatant: 1.09 g/Ml
 Tank C-106 vapor space pressure: 1 atm

Ammonia concentration in supernatant = 2.79 g/L
 = 2,560 mg/kg

$$\begin{aligned} x &= 2,560 \text{ mg/kg} \times 1\text{E-}3 \text{ g/mg} \times \text{mol NH}_3/17\text{g} \\ &= 1.51\text{E-}1 \text{ mol/kg} \end{aligned}$$

$$\begin{aligned} \text{Therefore, } y &= 1.51\text{E-}1 \text{ mol/kg} \times 2.6\text{E-}2 \text{ atm kg/mol} / 1 \text{ atm} \\ &= 3.93\text{E-}3 \text{ mole fraction} \end{aligned}$$

Based on the mole fraction of 3.93E-3 of ammonia, the ammonia concentration by volume is equivalent to 0.393 % in any specific volume such as 1 m³ of C-106 vapor space. An increase of temperature will increase the ammonia concentration in air (i.e., 28 °C or 82.4 °F increase will equal a three-fold increase in ammonia concentration). Ammonia is explosive at about 16 to 25% by volume in air; increasing the temperature and pressure of the ammonia broadens the flammability range. The vapor space of C-106 or AY-102 is estimated to contain a 0.39% ammonia concentration at 35 °C (95 °F) during sluicing operation. Assuming a three-fold increase, the concentration would be about 1% at 80.5 °C (177 °F) that would discount any explosive potential from ammonia.

An article in the Canadian Journal of Chemical Engineering dated 1962, "Effects of Waste Vapor Content on the Inflammable Limits of Ammonia - Oxygen - Nitrogen Mixtures" authored by W.J. De Coursy, N. Zubrychyp, and Yoshida evaluated ammonia explosion in dry air. This evaluation indicated that at atmospheric pressure, the inflammable limits of ammonia in dry air were found to be 17.2 and 30.2 % ammonia, but these limits were found to narrow as the water vapor is increased, until at 10.8 % and 20.8 % NH₃, no flame could survive with a dry - oxygen - ammonia mixture, the lower limit was found to be 15.2 % ammonia. At ten atmospheres, the explosive limits of ammonia in dry air were found to be 16.9 % and 28 %, and these limits were narrowed with addition of waste vapor until above 9 % water, no explosion occurred.

4.2.2.5 Summary of Chemical Consequences

Appendix F of WHC-SD-WM-ES-290 (Sederburg 1994) documents the outcome of previous waste compatibility work performed in support of Project W-320. These studies yielded the basis for selection of the receiver tank and sluicing material.

The proposed sluicing of C-106 to AY-102 is acceptable from an engineering process control perspective. Large margins of safety have again been identified in the analysis eliminating the need for further sampling of these tanks.

The analyses conducted demonstrate that the transfer of waste from C-106 to AY-102 is allowable. The only provision is that the corrosion specifications be met by chemical adjustment either before or during the transfer.

4.2.3 Toxicological Consequences

The WHC-SD-WM-ISB-001 document (Leach and Stahl 1993) has detailed discussions on the onsite/offsite toxicological exposure guidelines and methodology in Section 2.4 as well as in topical report on toxicological consequences. Past safety analyses focused mainly on radiological concerns, and tended to only briefly address chemical release hazards. Regulations have been developed to address the issue qualitatively for potential harmful chemical releases. Quantitative guidelines also have been developed for accidents involving chemicals. The safety analysis issue of tank toxic gas emissions is not fully characterized at this time. Significant concern over the release of toxic gas from any tank other than C-103 and SY-101 has been shown only recently. In addition, operations will be conducted to ensure a safe working environment in compliance with WHC-CM-4-3, *Industrial Safety Manual*.

Three toxic release analyses were performed for AQ Tank Farm and the outcome was well below the acceptable limits. Further, few safety analyses have attempted to characterize the nature and extent of the toxic gas release hazard. As recommended in WHC-SD-WM-ISB-001, guidelines and methodology of that document will be followed to determine whether there are any toxic gases that are available for release before WRSS hot operation. If so, the release rate and concentration of the toxic chemicals will be determined and compared to the appropriate limits as discussed in Section 2.4 of WHC-SD-WM-ISB-001. The consequences of toxic chemicals, if any, will be incorporated into the SA before WRSS hot operation.

4.2.4 C-106 Temperature Anomalies and Steam Release Analysis

As noted in Section 1.1, 23,000 L/month (6,000 gal/month) water are added to C-106 to prevent the sludge from drying out and C-106 from overheating. During a process test that started in March 1994, the liquid level in C-106 was lowered by evaporation to below the sludge level. At the end of the test in June 1994, when water was added, the temperature readings on the thermocouples (TCs) in riser 14 began to rise, averaging about 2.5 °C/day (4.5 °F/day) at TC-1 (16 in. above the bottom of the tank) between June 16 and June 26, 1994. Beginning on June 26, the TCs in the sludge began to manifest a thermal cycle. Fluctuations at TC-2 showed the maximum amplitude during the cycling, with a peak-to-peak variation of about 11 °C (20 °F). The cycling stopped at its high point after five cycles, with an average period of about 29 hours. Temperatures continued to rise at a decreasing rate, and reached 104 °C (219 °F) by August 2. Thereafter they began to slowly decrease.

The authorization basis, Limiting Condition for Operation (LCO) 3.2.1 of Dougherty 1993, states that any significant increase in the waste temperature is an "undesired trend and should be investigated to determine the cause and impact of the increase". The temperature rise in the TCs in riser 14 following the process test, particularly the greater than 20 °F increase during the thermal cycling of TCs in riser 14 between June 26 and July 3, caused an occurrence report to be written. An unreviewed safety question (USQ) is being evaluated in TF-94-0271 (Kummerer 1994) with regard to the

C-106 temperature anomalies and steam release issue. A technical assessment of the potential for a steam release in C-106 has been documented in WHC-SD-WM-ER-379 (Bander 1994) in support of the USQ evaluation.

A steam release event may happen if the temperature of a layer of solids is significantly greater than the atmospheric boiling point of an overlying liquid layer, and the liquid is near its boiling temperature. If a disturbance permits the solids to come into sudden contact with the cooler liquid, some of the liquid will flash to steam. The steam will expand and rise through the overlying layers of waste and may erupt through the surface. The energy of this eruption could carry aerosols and droplets of waste with it into the dome space, potentially pressurize the dome space and may allow a release from the tank.

Thermocouples in riser 8, located closer to the tank wall, did not show unusual temperature trends during this time. The tank vapor space temperature measured at both locations, did not change either. It is suggested that the behavior observed at riser 14 is due to a local phenomenon, and that the bulk of the waste has not seen a significant rate of temperature increase.

Investigation of these phenomena have lead to the conclusion that an opening existed around the TC tree in riser 14, caused by the water lance operation used to facilitate installation of this TC tree. This opening did not fill with solid waste, and therefore provided a pocket of convective waste around the thermocouple location. Because the convective waste cooled more readily, its temperatures were lower than those in the solid waste that had to reject heat by a slower conduction process. Lowering of the liquid level in the tank to below the solids level allowed the waste from above to slump into the cavity. The TCs in riser 14 appear to continue to oscillate and are returning to their pre-process test levels.

The apparent redistribution of waste around riser 14, accompanied by the rise in thermocouple readings, have led to concerns that conditions for steam formation and release could exist. The analysis focused on conditions that existed in Hanford tanks that experienced steam releases in the past, compared to C-106 conditions both under normal cooling and under a four month loss of active cooling in 1992.

The analysis concluded that, since no evidence of a release from the tank occurred under the conditions that existed when active cooling was lost for four months in 1992, the minimum requirements for a steam release were not met. The conclusion drawn was that either there was sufficient heat transport within the sludge to cool and condense any steam bubbles that may have formed, or that the sludge shear strength was insufficient to retain vapor in the saturated zone.

The energy available for a steam release, when active ventilation cooling was lost, was estimated to be about 40 times greater than that present under normal cooling conditions. Changes that may have occurred in the waste since the 1992 event were also examined for their effect on the potential for a steam release. It was concluded that conditions in C-106 are significantly below those necessary to initiate a steam release event. It can be concluded that a significant steam release during normal tank operations is extremely

unlikely. Arguments were provided as part of the analysis that the steam release event could be further mitigated by additional cooling of the waste.

Concerns about C-106 temperature anomalies and potential steam releases are continuously being investigated. Further details and results of this investigation will be provided in SA. Tank temperatures and waste levels are being closely monitored at the present time.

4.2.5 Other Consequences

Other accidents that were not analyzed such as tank bumps for AY-102, ferrocyanide, organics, etc., have been analyzed by other tank farm facilities. These accidents were found to be of less consequence than the accidents previously discussed. As discussed in Chapter 8.0, all hazards will be evaluated during the preparation of the SA.

4.3 CONSTRUCTION HAZARDS

The construction hazards are short-term risks during construction of the WRSS systems and equipment installation/removal operations. These hazards are discussed in Chapter 5.0 and summarized in Table 5-1.

5.0 CONSTRUCTION RISKS

The WRSS activities will be conducted inside the Hanford Site boundary and within the 200 East Area. Normal construction risks will be encountered during the modification of the facility. Construction activities are not expected to cause any risks to the existing operating facilities located near the WRSS site in the 200 East Area.

The safety assessment specified in Chapter 6.0 will be completed and approved before the start of WRSS hot operation.

5.1 EQUIPMENT REMOVAL

This section identifies those activities required to prepare the C-106 and AY-102 pits for cleaning, removal of equipment and excavation for transfer lines, grading, utilities, access routes, etc., as required for installation of WRSS equipment.

5.1.1 Pit Cleaning

The following pits will be cleaned and decontaminated by using water washing or CO₂ decontamination equipment in accordance with WHC procedures. Cleaning will occur before entry into the following pits to remove existing equipment and to install new equipment:

- 241-C-06A, 241-C-06B, and 241-C-06C
- 241-AY-02A, 241-AY-02B, and 241-AY-02E.

Pits 241-C-06A, 241-C-06B, and 241-AY-02E will be cleaned as part of the equipment removal system (ERS). The remaining pits will be cleaned before construction activities are started within the pits.

5.1.2 ERS

The ERS will be used to remove the existing installed equipment, such as pumps, within pits 241-C-06A, 241-C-06B, and 241-AY-02E.

The ERS will consist of five major systems: water washing or CO₂ decontamination system; the flexible receiver system; the strongback system; the equipment storage containers; and the container transport trailer. Other components, such as equipment storage and disposal containers, shield plugs, and lifting fixtures, also will be designed and fabricated as necessary to support the equipment removal effort.

5.1.2.1 ERS Assembly. Before operation of the ERS, all components will be staged at the tank farm. An empty container will be loaded onto the strongback and bolted into place.

The flexible receiver system components consist of a decontamination spray ring, a transition spool piece, an equipment lifting fixture, and a flexible receiver support structure. The flexible receiver system will be positioned and assembled within, or attached directly to, the tank pit, as required. In addition, the flexible receiver will be connected to the control trailer and high-pressure pump system.

The strongback system consists of a crane, a jack frame, and a control trailer. Following assembly of the flexible receiver, a jack frame will be positioned over, and connected, to the lifting eye on top of the flexible receiver.

5.1.2.2 ERS Operation. Before the start of pump removal activities, the jack frame will be used to overcome the initial breakout force resulting from the solidified material anticipated to be surrounding the bottom of the pump. Once the pump is broken free, the jack frame will be removed and decontaminated, if required. The crane will be positioned and connected to the flexible receiver lifting eye. The container will be raised to the vertical position by the strongback.

The pump will be extracted from the tank by the crane. As the pump is extracted, it will pass through the decontamination spray ring, which will rinse the pump with water at a pressure of about 2×10^7 Pa (3,000 psi). Gamma monitoring also will be performed and recorded continuously during the extraction process. As the pump passes through the flexible receiver, a bag will be applied which will provide containment of the pump before placement within the container. Once the pump is fully extracted from the tank, the bag will be remotely cinched, sheared, and tied. The bagged piece of equipment then will be positioned in the secondary bagging apparatus where a secondary bag (boot) will be remotely placed and secured.

The bagged pump then will be raised by the crane and positioned above the vertical container. A remote camera located on the boom will assist the crane operator in placement of the bagged pump into the container. Once the pump is centered over the top of the container it is slowly lowered into position within the container. The container then will be lowered from vertical to horizontal position by the strongback hydraulic positioning system. Once the container is in the horizontal position, the crane will be disconnected and inner and outer flanges will be placed on the container.

A radiation survey will be performed following closure of the container. Shielding (lead or steel shot) will be added to the annulus of the container to meet transportation criteria. The container then will be secured to the control trailer at the attachment points. The strongback will be disconnected from the power supply and prepared for transportation.

The strongback has been designed, and is anticipated to be used, as the primary means of transportation of the loaded container from the tank farm to the onsite storage facility. However, in the unlikely event the strongback is contaminated during placement of the removed equipment into the container, a container transport trailer with engineered attachment points and supports will be available. Should it be necessary to use the container transport trailer, the container will be decontaminated, unbolted from the strongback,

and transferred to the container transport trailer. The container then will be transported to the onsite storage location.

Equipment removal activities will be concluded by disconnecting, decontaminating (as required), and removing all components of the ERS from the tank farm. The equipment then will be stored or staged for further use.

5.1.3 Other Activities

Other site preparation activities include, but are not limited to, the following: improvements to land such as grading; access route modifications including addition of new routes; providing utilities such as electrical power, water supply, fire protection, sanitary facilities, etc., needed for construction; installation of construction trailer; rerouting the fence line for access control as well as to accommodate necessary construction activities for WRSS; and preparing a construction staging area. These activities will be performed in accordance with the WHC procedures and the detail layout drawings.

5.1.4 Risks and Hazards

Analyses for potential risks, hazards, and controls for the various site preparation activities will be documented in the Tank Waste Remediation System Safety Analysis Reports Reference (TWRS SARR). The results of these analyses will be referenced and/or used in the WHC work procedures and job control system work packages as part of the above site preparation activities. These will be reviewed under ALARA program.

5.2 CONSTRUCTION

Routine construction hazards will exist while the WRSS is under construction and during equipment installation at the tank farm. Following equipment removal, the pits will be cleaned again before construction work is conducted in the pits. Field operations and construction operations will be conducted to ensure a safe working environment in compliance with WHC-CM-4-3, *Industrial Safety Manual*, and applicable Occupational Safety and Health Administration and Washington Industrial Safety and Health Act standards.

The Project W-320 design precludes as far as possible the need for confined space entry during construction and operations. Project W-320 systems, equipment, and modifications will be designed to facilitate compliance with lock and tag procedures.

5.3 WASTE MINIMIZATION AND HAZARDOUS MATERIAL USAGE

The project design minimizes hazardous and nonhazardous waste generation and the use of hazardous materials during construction, maintenance, operation, and decommissioning/closure. Some materials used for construction and operation are not considered hazardous until they are designated as waste.

Where materials are used that are hazardous or will become hazardous upon designation as a waste, a note or similar identification will be added to the definitive design media. Where lead or a similar hazardous material must be used for shielding or other purposes, the item will be encapsulated to prevent contamination and to allow future retrieval in an uncontaminated condition. The item will be permanently marked to identify the contents. Material safety data sheets will be supplied for all materials (i.e., paints, sealants, oils, lubricants, chemicals, fill fluids) used in the facility and for construction. No polychlorobiphenyls (PCB) or asbestos are allowed. Transformers, lighting ballasts, etc., which use an insulating oil will be certified PCB-free.

5.4 RISKS RELATED TO NUCLEAR SAFETY

The risks related to either toxicological or nuclear safety are related to breaches in the existing primary or secondary confinement barriers by accident. Table 5-1 provides a matrix summary of construction sequences, potential accident, and target/potential consequences.

Table 5-1. Matrix Summary for Accident Scenarios.

Construction sequence	Potential accident and sequence	Target/potential consequences	Remarks
Overhead lifts	Load drop	Impacts and breaches primary/secondary confinements Personal injuries due to missile, etc.	Restrict lift areas by work procedures
Construction equipment	Exceed tank dome dead weight loads Vehicle impacts HVAC equipment	Impacts and breaches primary/secondary confinements Unfiltered release from tanks	Restrict access by work procedures and/or limit per design (a)
Modification of electrical service	Loss of facility power results in shutdown of tank ventilation and monitoring systems/nearby tank release occurs	Loss of tank ventilation, loss of process monitors	Procedural Lock outs Temporary back-up power (b)
Excavation	Tank riser damaged Tank dome damaged Process pipes Electrical cable runs	Unfiltered releases, personnel exposure/contamination Fire hazard and/or electrocution	Hand excavate
Entry to pits	Surface contamination on pit walls, waste trapped in jumpers exposed to environment	Personnel exposure/contamination	HPT support/personal protective gear (b)
Modification of pit(s) including core drilling of pit walls	Riser damage Tank dome change Wall damage	Unfiltered releases, personnel exposure/contamination Fire hazard and/or electrocution	Work procedures (b)

(a) Survey of affected areas is required to determine the risks/hazards

(b) Will be reviewed in safety assessment. Limited consequence or typical risk.

HPT = Health physics technician

HVAC = Heating, ventilating, and air conditioning

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6.0 SAFETY DOCUMENTATION

Safety assessment documentation will be provided for construction acceptance and startup testing before WRSS operations.

6.1 SAFETY

This section includes criticality safety, safety analysis, fire protection, and natural forces.

6.1.1 Criticality Safety

Criticality requirements have been determined by WHC-SD-SQA-CSA-20363, *CSEER 94-001: Criticality Safety of Single Shell Waste Storage Tanks*, (Rogers 1994). The requirements indicate that the occurrence of a criticality accident in the underground waste storage tanks or transfer lines is incredible because of the low concentration of fissile materials present in the waste.

6.1.2 Safety Analysis

A safety analysis will be conducted in accordance with DOE Orders 5480.23, *Nuclear Safety Analysis Reports* (DOE 1992c) and 6430.1A, Section 0110-5.2 (DOE 1989); and WHC-CM-4-46. A preliminary assignment of safety classes for the various systems in the WRSS is identified in Table 2-1.

An SA will be prepared instead of a preliminary safety analysis report and a final safety analysis report, in accordance with guidance from the U.S. Department of Energy-Headquarters. The SA shall be completed and approved before start of hot operation.

6.1.2.1 DBAs. The C-106 WRSS is designed to withstand the effects of DBAs, as delineated in DOE Order 6430.1A (DOE 1989) with confinement of radioactive and toxic materials to within allowable limits. Simultaneous occurrences of more than one DBA were considered when a joint occurrence, causally related to a common-mode failure, is possible.

6.1.3 Fire Protection

Fire protection sprinklers are located in Building 271-CR which houses the motor control center and control equipment for the C Tank Farm. The fire protection system performs no unique safety functions, and is classified as Safety Class 3. Fire protection systems in temporary facilities provided by this project, e.g., the sluicing control room, are designed to meet the following applicable requirements:

- DOE Order 6430.1A, all sections (DOE 1989)
- DOE Order 5480.4, Attachment 2, "Mandatory ES&H Standards," Section 2C, "Fire Protection" (DOE 1984a)
- DOE RLID Order 5480.7, *Fire Protection* (DOE-RL 1990c); DOE Order 5480.7A, *Fire Protection* (DOE 1993)
- DOE/EV-0043, *Standards on Fire Protection for Portable Structures* (DOE 1979)
- Applicable codes and standards listed in Section 2.0.

The fire alarm system will be compatible with, and tie in to, the Hanford Site 200 Area radio fire alarm system.

6.1.4 Natural Forces

The following naturally occurring DBA scenarios will be considered during preparation of the safety analysis (Section 6.1.2) and project design phase.

6.1.4.1 DBE. All systems, components, and structures will be designed for the DBE as specified in HPS-SDC-4.1, "Standard Architectural Civil-Design Criteria: Design Loads for Facilities" (DOE-RL 1989b) and DOE Order 6430.1A for Safety Class 1, 2, and 3, and Nonsafety Class 4.

6.1.4.2 Design Basis Wind. The WRSS, and features necessary to protect onsite personnel and the public, will be designed to withstand the 100-year recurrence interval windloading according to HPS-SDC-4.1.

6.1.4.3 Design Basis Flood. The elevation of the 200 Area site has been judged to be outside the maximum extent of the Hanford Site design basis flood.

6.1.4.4 Volcanic Eruptions. The WRSS location suggests that volcanic action could provide ashfall that may affect operation. The WRSS safety class criteria design will include protection from ash resulting from volcanic eruption, as required by HPS-SDC-4.1.

6.2 QUALITY ASSURANCE

Quality assurance/control activities for all contractors involved in design, construction, and acceptance testing will be executed in accordance with the project-specific quality assurance program plan (QAPP). The QAPP will be developed during conceptual design and approved/released before definitive design. This QAPP will be used by the design contractor to develop verification criteria in design documents (i.e. drawings, specifications, test procedures) and by all contractors to define quality assurance interfaces and specific quality requirements/responsibilities on the projects.

The QAPP will comply with the quality criteria of DOE Order 5700.6C, *Quality Assurance* (DOE 1991a).

The basis for establishing Quality Assurance Program requirements is safety classification as defined in WHC-CM-1-3, *Management Requirements and Procedures*, MRP 5.46, "Safety Classifications of Systems, Components and Structures." The safety classifications of items provide a graded approach to application of quality requirements. This graded approach assigns requirements to items commensurate with the function of each system, component, and structure in preventing or mitigating the consequences of hazards and postulated DBAs. The overall safety classifications for this project will be defined in the preliminary safety document prepared for this project.

A quality assurance implementing matrix will be prepared by each project participant and will identify the documents that implement the DOE Order 5700.6C requirements applicable to the project.

6.3 SAFEGUARDS AND SECURITY

The Safeguards and Security Department maintains an around-the-clock surveillance system of the Hanford Site to protect government property, classified information, and special nuclear material, and to isolate potential hazards. The Hanford Patrol maintains a continuous presence of armed guards to provide security for the site.

Manned barricades are maintained at checkpoints on the vehicle access roads from Richland (the Wye Barricade) and Yakima (the Yakima Barricade), Washington. Personnel must display a badge indicating access authorization to enter the Hanford Site.

No special security and safeguard requirements are needed beyond those currently in force. Control of personnel at the Hanford Site boundary is deemed adequate. The facility conforms with the security and safeguards requirements defined in DOE Orders and WHC policies and procedures. Fencing, access locks, and any other security/safeguard criteria necessary will be provided to meet DOE requirements, environmental regulations, and WHC policies and procedures for installation and operation of the WRSS.

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7.0 PROJECT INTERFACES

This WRSS will not provide any new risks to the occupants of the existing tank farm facilities. The risks to the occupants of the WRSS are risks that comply with the Waste Tanks Administration safety program and are similar to past operations of tank farm facilities and ALARA practices.

The primary tank ventilation, backup ventilation systems, emergency cooling water system, and steam supply system for AWF boundaries may be replaced by Project W-030.

7.1 DESIGN FORMAT

Two-way traceability will be provided between project drawings and the reference drawings from which they were developed. Project drawings will identify existing drawings that will be affected by the project. Project W-320 will update existing drawings that are affected by project design. AutoCAD[®] Version 11 or 12 will be used in Project W-320 design activities. Drafting will be in accordance with applicable sections of WHC-CM-6-3, *Drafting Standards Manual*. Equipment will be identified in accordance with the Data Value Standards; "Tank Farm Equipment Identification Number."

7.2 COMMUNICATIONS AND TELECOMMUNICATIONS SYSTEMS

The WRSS control room is provided with phone, private automatic exchange, and Hanford Local Area Network communication systems. In addition, emergency notification and communication systems are provided and include an evacuation siren.

A generic workstation (computer system and printer including necessary hardware and software) is provided in the control room and tied into the Westinghouse Radiation Access Management system.

7.3 AUTOMATIC DATA PROCESSING

The WRSS will be provided with a centralized data acquisition and control system. This system will be used for data acquisition, data analysis, and inventory control. The system allows standalone monitoring and control. Recording equipment is provided to make a permanent record of alarms and process conditions.

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8.0 ITEMS REQUIRING FURTHER RESOLUTION

Several criteria will require further resolution either at the detail design stage, before construction, or before operation. The following paragraphs discuss each criterion.

8.1 DESIGN ISSUES

Before hot operations, the design issues of the Safety Related Design Criteria; General Design Criteria (Section 2.4.1), Radiological Design (Section 2.4.2), and mechanical process fluid properties (Sections 2.3.4, Capacity; 2.3.5, Tank Environmental; and 2.3.6; Operating Temperature) will be resolved and incorporated into the SA.

8.2 SA ITEMS

During the definitive design, the applicable sections of all WHC manuals will be reviewed and incorporated into the SA (Section 1.0). Also, potential hazards will be evaluated and analyzed with respect to consequence and probability of occurrence and incorporated into the SA (Section 3.0).

8.3 SOURCE TERMS AND ELEMENTAL INVENTORY IN TANKS

Before waste transfer, the waste inventory will be evaluated for compliance within the bounds of the SA. The evaluation will address inventory changes in C-106 and AY-102 during Project W-320 design, construction, and startup testing.

8.4 OPERATIONAL SAFETY ISSUES

Operational safety issues for safe operation will be resolved through administrative controls before WRSS operation. Administrative controls are specific procedural criteria for operation of the facility which define the technical safety requirements (TSR) for safe WRSS operation. The administrative control TSRs (WHC-CM-4-46) will provide the facility organization and management, procedure systems, document control systems, configuration management systems, staffing requirements, training, and other controls deemed necessary for safe operation of the facility. The TSRs will cover all significant hazards and risks identified in Chapter 4.0 of this PSE, and other risks associated with normal operation of the tank farm facility.

The hazards and assumed risks for the C-106 WRSS have been evaluated, and operational safety requirements will be imposed to limit risks to the operation of the facility. Safety limits and limiting control settings will be developed for operation of the tank farm equipment, and safety parameters will be established. Conservative limits will be selected such that automatic or manual protective action will correct the abnormal situation before a

safety limit is exceeded. Hazards and assumed risks for the facility are as follows.

- Corrosion control applies to the stainless steel and carbon steel materials for the tanks and transfer piping during all modes of operation. To prevent damage to the waste tanks and transfer piping, limits on chemistry of the process waste will be established. This limits detrimental chemical concentrations to safe levels to prevent excessive corrosion and extend the life of the facility.
- Double confinement applies to the waste transfer lines and DST waste storage tank. To prevent loss of function of the major barriers that prevent releases of hazardous materials to the environment, limits for transferring and storing of waste will be established by limiting transfer or storage of waste in leaking primary and secondary lines or tanks. Surveillance of leaks will be provided by leak detection devices in the encasement piping of transfer lines, dry well, and annulus for DSTs. Recovery plans will be established for loss of containment during transferring and storage.
- Gaseous effluent applies to the ventilation systems, and the associated instrumentation during operation. To prevent significant hazardous material releases into the atmosphere, WHC-CM-7-5, *Environmental Compliance Manual*, has been established to mitigate, confine, or stabilize releases below the radiological guidelines. Release limits will be established, and surveillance of exhaust stacks emissions will be provided to establish trends in the annual release limits. Recovery plans will be established for exceeding the annual average concentration limits.
- Heat generation calculations will routinely be performed for each tank to ensure that the heat generation limits are not exceeded. The results of these calculations will be evaluated in accordance with the administrative controls. Recovery plans will be established for necessary corrective action.
- Concerns about C-106 temperature anomalies and potential steam releases are continuously being investigated. Further details and results of this investigation will be provided in SA. Tank temperatures and waste levels are being closely monitored at the present time.
- Primary tank pressure applies to the waste storage tanks, the ventilation system, and instrumentation during operation. To prevent damage to the waste tanks, a limit for tank vapor space pressure and vacuum will be established. Surveillance methods and alarms will monitor the tank pressure during operation to control over pressurization of the system. Recovery plans will be established for safety limit violations to minimize impacts to the operation of the facility, and provide a margin of safety for return to safe operation of the systems.

- Minimize/prevent the potential release of contamination from the pits at cover block removal. One proposed solution is the use of sampling and washdown capabilities. This and other solutions will be evaluated during detailed design to resolve the issue. Verification is needed in the final detailed design and in the SA.

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9.0 REFERENCES

9.1 IN-TEXT REFERENCES

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GLOSSARY

ABBREVIATIONS AND ACRONYMS

AC	alternating current
ACI	American Concrete Institute
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
AWF	Aging Waste Facility
AWWA	American Water Works Association
AY-101	tank 241-AY-101
AY-102	tank 241-AY-102
C-106	tank 241-C-106
CF	conservative multiplication factor
DBA	design basis accident
DBE	design basis earthquake
DOE	U.S. Department of Energy
DOE-RL	U.S. Department of Energy, Richland Operations Office
DST	double-shell tank
EDE	effective dose equivalent
EHSC	environmental hazard safety classification
EP	Engineering Procedure
ERS	equipment removal system
FDC	functional design criteria
GENII	a radiation dosimetry computer code
HEPA	high-efficiency particulate air
HPS	Hanford Plant Standards
HVAC	heating, ventilating and air conditioning
ICBO	International Conference of Building Officials
IEEE	Institute of Electrical and Electronics Engineers
ISA	Instrument Society of America
ISEL	interim safety equipment list
L	liter
MJ	million joules
MRP	Management Requirements and Procedures
NFPA	National Fire Protection Association
OSHA	Occupational Safety and Health Administration
OSR	operational safety requirement
P&ID	process and instrumentation diagram
PCB	polychlorobiphenyl
PM	plume meander
PSE	preliminary safety evaluation
PSEL	preliminary safety equipment list
QAPP	quality assurance program plan
SA	safety assessment
SDC	standard design criteria
SpG	specific gravity
SST	single-shell tank
TMACS	tank monitor and control system
TOC	total organic carbon

TSR	technical safety requirement
UPS	uninterruptible power supply
WAC	Washington Administrative Code
WHC	Westinghouse Hanford Company
WRSS	waste retrieval sluicing system

APPENDIX A

DOSE CONSEQUENCES CALCULATIONS

NOTE

THE ANALYSIS IN THIS APPENDIX A IS BASED ON THE RADIOACTIVE INVENTORY DATA PRESENTED IN WHC-SD-WM-TI-543, *RADIONUCLIDE AND CHEMICAL INVENTORIES FOR THE DOUBLE SHELL TANKS* (VAN VLEET 1993a), AND WHC-SD-WM-TI-565, *RADIONUCLIDE AND CHEMICAL INVENTORIES FOR SINGLE SHELL TANKS*, (VAN VLEET 1993b). AS NOTED IN BANDER 1993a AND 1993b, THE RADIOGENIC HEAT LOAD IS 32.2 KW (110,000 BTU/H). THE CALCULATED HEAT LOAD DUE TO RADIOACTIVE ELEMENTS IN C-106 BASED ON DATA FROM VAN VLEET 1993b IS DOMINATED BY DECAY OF ^{90}SR AND ^{137}CS WITH NEGLIGIBLE CONTRIBUTION FROM OTHER RADIONUCLIDES, AND IS EQUIVALENT TO 13.4 KW; 12 KW FROM ^{90}SR AND 1.4 KW FROM ^{137}CS . THE RADIOACTIVE INVENTORY DATA IN VAN VLEET 1993b ARE SUSPECT VALUES ACCORDING TO BANDER 1993a AND 1993b. THIS INTRODUCES A SCALING FACTOR OF 2.4 DUE TO THE DISCREPANCY BETWEEN RADIOGENIC HEAT LOAD IN THE BANDER REPORTS AND THE ESTIMATED HEAT LOAD FROM THE CURRENT RADIOACTIVE INVENTORY IN C-106. CHARACTERIZATION OF C-106 IS SCHEDULED TO BE COMPLETED IN THE NEAR FUTURE. UNTIL SUCH TIME AS THE NEW CHARACTERIZATION DATA FOR C-106 IS AVAILABLE, THE 2.4 SCALING FACTOR MUST BE APPLIED TO THE ESTIMATED EDE VALUES IN APPENDIX A. THE SCALING FACTOR OF 2.4 HAS BEEN APPLIED TO THE EDE VALUES ONLY IN SECTIONS 1.4 AND 4.2.1 OF THIS DOCUMENT, AND THE RESULTS ARE INCLUDED THEREIN.

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					PRELIM.	FINAL	VOID	REVISION				
						X		0				
					DISCIPLINE Mech-Nuclear							
PROJECT TITLE <u>241-C-106 Waste Retrieval</u>						J.O. 7362.007						
STRUCTURE OF SYSTEM <u>N/A</u> DESIGN CLASSIFICATION <u>Varies</u>												
SUBJECT <u>EDE Dose Consequences and EHSC Calculations</u>												
COMPLETED BY <u>Nagaraj S Vembar</u> <i>N. Vembar</i> DATE <u>04/22/94</u>												
CHECKED BY <u>Donnie Harrison (Jason Associates)</u> <i>Donnie Harrison</i> DATE <u>4/22/94</u>												
APPROVED BY <u>A. Edmondson</u> <i>A. Edmondson</i> DATE <u>4/26/94</u>												
REASON FOR REVISION:					TOTAL NUMBER OF SHEETS IN THIS ISSUE <u>48</u>							
					Not applicable				SHEETS REVISED, ADDED OR DELETED OR DELETED			
									Not applicable			
PROBLEM STATEMENT: To provide conservative estimate of the dose consequences and offsite ground contamination from airborne releases for various accident scenarios for the WHC Project W320, 241-C-106 Waste Retrieval, and Environmental Hazard Safety Classification calculations per EP-1.4. The results will also be used to determine the safety classification of systems that are associated with C-106 Waste Retrieval operation.												
SUMMARY/CONCLUSIONS: See Page <u>44</u> of General Computation Sheets for Summary and Conclusions												

FORM 5610 REV. 4/81

CALCULATION SUMMARY & CONTROL SHEET Page 2 of 2	CALCULATION SET NO. 7362-H-006			
PROJECT TITLE <u>241-C-106 Waste Tank Retrieval System</u>	PRELIM.	FINAL	VOID	REVISION
		X		0
	DISCIPLINE Mech-Nuclear			
J.O. 7362.007				
DESIGN BASIS See Page <u> 1 </u> of General Computation Sheets				
UNVERIFIED ASSUMPTIONS/OPEN ITEMS: <div style="text-align: center; font-size: 1.2em;">NONE</div>				
REFERENCES: (SPECIFICATIONS, DRAWINGS, CODES, CALCULATIONS, TEXTS, REPORTS, COMPUTER DATA, FSAR, ETC.) See Page <u> 6 </u> of General Computation Sheets				

Form 5007 (Rev. 4/93)

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				7362-H-006				
	PRELIM.	FINAL	VOID	0	NSY DATE 4-22-94	DATE 4-22-94		
	PROJECT 241-C-106 Waste Retrieval			SHEET 1 OF 48		DATE	DATE	
SUBJECT Dose Consequences and EHSC Calculations			J.O. 7362.007					

DESIGN BASIS

1. A conservative estimate of the radionuclide inventories for the C-106 and AY-102 tanks used in the C-106 waste retrieval is taken from Appendix B, Hanford Site Tank Farm Facilities Interim Basis, WHC-SD-WM-ISB-001 (Ref. 13). Analysis in Appendix C (Ref. 13) was performed for unit liter respirable releases of selected Hanford high level waste tanks which included 241-C-106, 241-AY-101 and 241-AY-102 tanks. The unit liter release calculations were performed using the GENII dosimetry code (Ref. 2) for both laboratory sample data and Track Radioactive Components (TRAC) data. Per FDC (Ref. 12), the supernatant in AY-101 will be transferred to AY-102 prior to WRSS operation. The calculations will be based on unit liter release source term values for laboratory data (conservative values) of C-106 solids and supernatant from AY-101 in AY-102. The source term values have been taken from Table C-7 for AY-101 and Table C-9 for C-106, Appendix C of Ref. 13 and are listed in Table A-1. The individual is assumed to be located at the Hanford site boundary in the direction which provides the highest 99.5% (sector normalized) atmospheric dispersion coefficient. The individual is also assumed to remain on the centerline of the plume during its entire passage. The exposure pathways for the offsite individual include inhalation, submersion, and ground shine. Internal dose from inhalation is calculated as a 50 year committed effective dose equivalent (EDE).

The unit liter inhalation EDE to the bounding offsite individual for solids in C-106 (per Table C-9, Appendix C, Ref. 13) is: 2.30E-2 rem (per liter)

The unit liter inhalation EDE to the bounding offsite individual for supernatant in C-106 (transferred from AY-102 with AY-101 supernatant (see Criterion 14 on Sheet 5 and per Table C-7, Appendix C, Ref. 13) is: 1.80E-5 rem (per liter)

Similarly, the unit liter submersion EDE to the bounding offsite individual for C-106 solids (per Table C-9, Appendix C, Ref. 13) is: 9.50E-7 rem (per liter)

The unit liter submersion EDE to the bounding offsite individual for C-106 supernatant (transferred from AY-102 with AY-101 supernatant, see Criterion 14 on Sheet 5 and per Appendix C, Rev. 14) is: 2.10E-7 rem (per liter)

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	PRELIM.	FINAL	VOID	0	NJV DATE 4-22-94	DATE 4-22-94
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It is concluded from the above that the submersion doses are negligible compared to the inhalation doses, and therefore, are not included in the calculations. In addition, per Criterion 15 below, ground shine doses have not been included in the calculations.

Since all of the radionuclides of concern have long half-lives, doses to the receptors at other locations are proportional to the maximum doses calculated for the offsite individual. The atmospheric dispersion coefficients used to derive these conversions are provided in Table C-5 of Appendix C of Ref. 13.

To convert from the offsite individual maximum dose to doses for other receptors, the plume meander (PM) conversion factors given in Table C-10 of Appendix C of Ref. 13 are used. For the purpose of this evaluation and as recommended in Appendix C of Ref. 13, the most conservative multiplication factors (CF) with or without plume meander (PM) have been used as follows: CF = 1 for maximum doses to offsite individual without plume meander (PM); CF = 0.64 for maximum doses to offsite individual with PM; CF = 610 for maximum doses to onsite individual with PM; and CF = 1900 for maximum doses to onsite individual without PM.

The unit dose calculations for C-106 waste retrieval are listed in Table A-1 for offsite individual without/with PM, Table A-2 for onsite individual without PM, and Table A-3 for onsite individual with PM.

Table A-1. Unit Liter Release Dose Data for Offsite Individual without/with PM

Offsite	EDE, rem/liter (solids)	EDE, rem/liter (supernatant)
Inhalation without PM (CF = 1)	2.30E-2	1.80E-5
Inhalation with PM (CF = 0.64)	1.47E-2	1.15E-5

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Table A-2. Unit Liter Release Dose Calculation for Onsite Individual Without Plume Meander

Onsite	EDE, rem/liter (solids)	EDE, rem/liter (supernatant)
Inhalation	4.37E + 1	3.42E-2

Table A-3. Unit Liter Release Dose Calculation for Onsite Individual With Plume Meander

Onsite	EDE, rem/liter (solids)	EDE, rem/liter (supernatant)
Inhalation	1.40E + 1	1.10E-2

2. Per Section 4.2 of Onsite/Offsite Worker Definition (Ref. 4), a two hour evaluation period is recommended for all offsite calculations to the general public. This is also in conformance with the guidelines of Regulatory Guide 1.145 (Ref.5).
3. Per Items 9 and 10 of Section 4.2.1, Chapter 4.0 of *Nonreactor Facility Safety Analysis Manual*, WHC-CM-4-46 (Ref. 7), only inhalation and submersion pathways are considered for determining the offsite individual EDE values, as well as for onsite individual EDE values. Per *Safety Classification of Systems, Components, and Structures*, MRP 5.46 (Ref. 8), only EDE values are needed to determine the safety classification of systems, equipment and components.
4. Design data for process systems that are considered for the accident analyses are taken from WHC-SD-W320-FDC-001, Rev. 2, Ref. 12.
5. The solids/sludge within the Tank AY-102 are presumed not to be disturbed during sluicing from C-106 to AY-102.

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6. A ground level point source release is used for all the accident scenarios.
7. The radioactive inventories included in Appendix B of Ref. 13 for C-106 and AY-102 (with AY-101 supernatant, see Criteria 1 and 14) are best available representative laboratory samples as stated in Appendix C of Ref. 13.
8. Transfer lines are buried 1 ft. below grade with a berm. A minimum of 3 ft cover over the pipe will be provided for freeze protection and shielding. Transfer lines are pipe in a pipe design, i. e., inner pipe with an outer pipe encasement. From Ref. 17, the failure probability of a pipe is $1E-9$ hr/ft. Based on 2000 ft. pipe (Ref. 11), an operating period of 8 hours per day for one year, the probability is estimated to be $5.8E-3$ /yr. Therefore, failure probability of both pipes is estimated to be $3.4E-5$. From Ref. 16, the annual probability (frequency) of a design basis earthquake at Hanford is $7E-4$ /yr. For the purposes of the analysis in this calculation, seismic event is considered for the worst case accident scenario.
9. HVAC design data is as shown on drawing H-2-818470 (see Ref. 11). HVAC is not required for safe shutdown of WRSS, i.e., if HVAC is shutdown or otherwise, tank will be on passive mode with no forced cooling.
10. Per FDC Rev. 2 (Ref. 12), air characterization for vapor space within C-106 is based on 100 mg/cu.m and vapor specific gravity of 1.0 (assumed for calculations purposes since the tank vapor space will be 100% humid and will consist of 90 % liquid per Ref. 12). Per Appendix D of Ref. 12, the vapor space will consist of 10% C-106 solids and 90% AY-101 supernatant in AY-102.
11. Radioactive leak from a transfer line break with 3 ft. cover (see Criterion 1 above), will reach the above grade level surface and will form a pool near the break. The quantity of leak that forms in to a pool at the surface is estimated to be 10 percent of the total quantity that is postulated to leak during the stated release time. This is based on a buried line break incidence at 242-S evaporator and its accident evaluation as noted in Section 9.1 of WHC-SD-WM-SAR-023 Rev. 1 (Ref. 14). A similar approach will be utilized for estimating leak for C-106 waste retrieval project, i.e., 10 percent will leak to form a pool at the surface from a total quantity of the liquid that is postulated to leak into soil during the entire release time period. Further, the buried piping configuration and the process conditions for the

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W320 project is similar in configuration and/or less severe in process conditions as compared to the 242-S buried lines.

12. Release fraction is equal to 1.0E-5 (0.00001) per DOE-HDBK-0013-93, Mishima, Ref 15.
13. Based on a preliminary evaluation of the system design and EDE dose consequences due to transfer line break, Project W320 has decided to install a seismic switch to mitigate the impact to the personnel safety and health due to dose consequences, and to the environment due to soil contamination. A 10 seconds time is used for calculations purposes as the release time period for pump coast down before the pump completely shuts down. This time period will be used for calculating the dose consequences from the waste released to the environment during the line break scenario.
14. Per Appendix D of FDC, Ref. 12, the supernatant from AY-101 will be transferred to AY-102. This supernatant will be transferred to C-106 for sluicing out the solid contents in C-106 to form slurry. This slurry will be transferred to AY-102.
15. For supply transfer line break calculations, 30 percent solids (Sp. Gr. of 1.2) from C-106 and 70 percent supernatant (Sp. Gr. of 1.0) from AY-102 (AY-101 supernatant) source term values have been used per FDC Rev. 2 (Ref. 12). For return transfer line break calculations, 10 percent solids (Sp. Gr. of 1.0) from C-106 and 90 percent supernatant (Sp. Gr. of 1.0) have been used per FDC Rev. 2 (Ref. 12).
16. Both transfer lines will break during a seismic event since they are buried in the same trench and bermed as noted in Criterion No. 8 above.
17. Syphon breakers are provided to prevent syphoning of AY-102 tank contents into C-106 since AY-102 tank is located at higher elevation than C-106.

UNVERIFIED ASSUMPTIONS/OPEN ITEMS

None

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CALCULATIONS

A. ONSITE AND OFFSITE DOSE CONSEQUENCE CALCULATIONS

1. SCENARIO NO. 1: WASTE TRANSFER LINES BREAK

Waste transfer line break scenarios are shown schematically in Figures 1a and 1b.

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1.a SCENARIO NO. 1a: WASTE TRANSFER SUPPLY LINE BREAK (2 Hours Release Time)

At a pump flow rate of 350 gpm (Para 2.2.2 of Ref 12) at the time of the accident and 2 hours release time per Criterion 2 on Sheet 3,

$$\begin{aligned}\text{Volume release is} &= 2 \text{ hrs} \times 60 \text{ min/hr} \times 350 \text{ gpm} \\ &= \underline{4.20\text{E}+4 \text{ gallons}}\end{aligned}$$

It is assumed that the accident occurs at C-106 tank farm. Then the volume (due to gravity) for the portion of approximately 2000 ft. 4 inch Sch. 40 piping (Ref. 11) is calculated as follows:

$$\begin{aligned}\text{Volume of 2000 ft of piping} &= \pi \times (4.026 \text{ in})^2 / 4 \times 2000 \text{ ft} \times 12 \text{ in/ft} \\ &= \underline{3.06\text{E}+5 \text{ in}^3}\end{aligned}$$

$$\text{Per Appendix B-11 of Ref. 1, } 231 \text{ in}^3 \text{ volume} = 1 \text{ gallon}$$

$$\begin{aligned}\text{Therefore, volume of liquid} &= 3.06\text{E}+5 \text{ in}^3 / 231 \text{ in}^3 / \text{gal} \\ \text{in the pipe} &= \underline{1.32\text{E}+3 \text{ gallons}}\end{aligned}$$

$$\begin{aligned}\text{Total volume release} &= 4.20\text{E}+4 + 1.32\text{E}+3 \\ &= \underline{4.33\text{E}+4 \text{ gallons}}\end{aligned}$$

$$\begin{aligned}\text{At 3.785 liters per gallon, per Ref. 1,} & \\ \text{Effective radioactive volume} &= 4.33\text{E}+4 \text{ gal} \times 3.785 \text{ liters/gal} \\ \text{release} &= \underline{1.64\text{E}+5 \text{ liters}}\end{aligned}$$

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Per Criteria No. 11 on Sheet 4, the release volume to the surface (forming a pool) is equivalent to 10 per cent of the total release, which is equal to

$$= 1.64E+5 \times 0.1$$

$$= \underline{1.64E+4 \text{ liter}}$$

Per Criterion 12 on Sheet 5, the release fraction for liquid release aerosol (from the pool surface) is $1.0E-5$.

Therefore, the amount of hazardous material suspended is

$$= 1.0E-5 \times 1.64E+4$$

$$= \underline{1.64E-1 \text{ liters}}$$

Solids Radioactive Release calculation

Based on 30 percent solids (per Criterion 15 on Sheet 5)

Effective radioactive volume release = $1.64E-1 \times 0.30 / 1.2$

$$= \underline{4.10E-2 \text{ liters}}$$

From Table A-1, offsite unit liter release EDE inhalation dose for C-106 solids is equal to $1.47E-2$ rem/liter with PM.

Therefore, for a total volume release of $4.10E-2$ liters,

the offsite EDE = $1.47E-2$ rem/liter $\times 4.10E-2$ liter

$$= \underline{6.03E-4 \text{ rem}}$$

Similarly from Table A-3, onsite unit liter release EDE inhalation dose for C-106 solids is equal to $1.40E+1$ rem/liter.

Therefore, for a total volume release of $4.10E-2$ liters,

the onsite EDE = $1.40E+1$ rem/liter $\times 4.10E-2$ liter

$$= \underline{5.75E-1 \text{ rem}}$$

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Liquid Radioactive Release calculation

Based on 70 percent supernatant liquid
 (per Criterion 15, Sh. 5),

Effective radioactive volume release = $1.64\text{E-}1 \times 0.70$

= $1.15\text{E-}1$ liters

From Table A-1, offsite unit liter release EDE inhalation dose for AY-102 (with AY-101 supernatant) is equal to $1.15\text{E-}5$ rem/liter with PM.

Therefore, for a total volume release of $1.15\text{E-}1$ liters,

the offsite EDE = $1.15\text{E-}5$ rem/liter $\times$ $1.15\text{E-}1$ liter

= $1.32\text{E-}6$ rem

Similarly from Table A-3, onsite unit liter release EDE inhalation dose for AY-102 with AY-101 supernatant is equal to $1.10\text{E-}2$ rem/liter.

Therefore, for a total volume release of $1.15\text{E-}1$ liters,

the onsite EDE = $1.10\text{E-}2$ rem/liter $\times$ $1.15\text{E-}1$ liter

= $1.26\text{E-}3$ rem

EDE dose consequence for Offsite Individual for both solids and liquid,

Scenario No. 1a = $6.0\text{E-}4$ rem

EDE dose consequence for Onsite Individual for both solids and liquid,

Scenario No. 1a = $5.8\text{E-}1$ rem

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1.b SCENARIO NO. 1b: WASTE TRANSFER RETURN LINE BREAK (2 Hours Release Time)

Based on Scenario 1a calculations, Sheet 9, the amount of hazardous material suspended is = 1.64E-1 liters

Solids Radioactive Release Calculation

The return line has 10 percent C-106 solids (per Criterion 15, Sh. 5)

Effective radioactive volume release = $1.64E-1 \times 0.10$
 = 1.64E-2 liters

From Table A-1, offsite unit liter release EDE inhalation dose for C-106 solids is equal to 1.47E-2 rem/liter with PM.

Therefore, for a total volume release of 1.64E-2 liters, the offsite EDE = $1.47E-2 \text{ rem/liter} \times 1.64E-2 \text{ liter}$
 = 2.41E-4 rem

Similarly from Table A-3, onsite unit liter release EDE inhalation dose for C-106 solids is equal to 1.40E+1 rem/liter.

Therefore, for a total volume release of 1.64E-2 liters, the onsite EDE = $1.40E+1 \text{ rem/liter} \times 1.64E-2 \text{ liter}$
 = 2.30E-1 rem

Liquid Radioactive Release Calculation

The return line has 90 percent supernatant liquid (per Criterion 15, Sh. 5)

Effective radioactive volume release = $1.64E-1 \times 0.90$
 = 1.48E-1 liters

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From Table A-1, offsite unit liter release EDE inhalation dose for AY-102 with AY-101 supernatant is equal to $1.15\text{E-}5$ rem/liter with PM.

Therefore, for a total volume release of $1.48\text{E-}1$ liters,
 the offsite EDE $= 1.15\text{E-}5$ rem/liter $\times 1.48\text{E-}1$ liter
 $= \underline{1.70\text{E-}6 \text{ rem}}$

Similarly from Table A-3, onsite unit liter release EDE inhalation dose for AY-102 with AY-101 supernatant is equal to $1.10\text{E-}2$ rem/liter.

Therefore, for a total volume release of $1.48\text{E-}1$ liters,
 the onsite EDE $= 1.10\text{E-}2$ rem/liter $\times 1.48\text{E-}1$ liter
 $= \underline{1.62\text{E-}3 \text{ rem}}$

EDE dose consequence for Offsite Individual for both solids and liquid,
 Scenario No. 1b $= \underline{2.4\text{E-}4 \text{ rem}}$

EDE dose consequence for Onsite Individual for both solids and liquid,
 Scenario No. 1b $= \underline{2.3\text{E-}1 \text{ rem}}$

EVALUATION OF BOTH SUPPLY AND RETURN LINE BREAK

EDE dose consequence for Offsite Individual for both solids and liquid,
 Scenarios 1a and 1b $= (6.0\text{E-}4) + (2.4\text{E-}4) \text{ rem}$
 $= \underline{8.4\text{E-}4 \text{ rem}}$

EDE dose consequence for Onsite Individual for both solids and liquid,
 Scenarios 1a and 1b $= (5.8\text{E-}1) + (2.3\text{E-}1) \text{ rem}$
 $= \underline{8.1\text{E-}1 \text{ rem}}$

From above, the EDE dose consequence values are < 500 mrem for offsite individual and < 5 rem for onsite individual doses. Therefore, the waste transfer return line should be classified as SC-3, per Ref. 9. Further confirmation is needed via Environmental Hazard Safety Classification (EHSC) per EP 1.4 (Ref. 9) to determine the safety classification. See Step B.1 for EHSC calculation for Scenarios 1a and 1b.

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1.c SCENARIO NO. 1c: WASTE TRANSFER SUPPLY LINE BREAK (Seismic Switch Shutoff and 10 Seconds Release Time)

Per Criterion No. 13 on Sheet 4, the seismic switch will terminate operation of the pumps, and the liquid release to the environment will stop within 10 seconds allowing for pump coast down, etc. In addition, release of the waste contents within the pipeline to the environment due to gravity flow has been included in the calculations.

At a pump flow rate of 350 gpm (Para 2.2.2 of Ref. 12) at the time of the accident and 10 seconds release time,

$$\begin{aligned}
 \text{Volume release} &= 10 \text{ sec} \times 350 \text{ gpm} / 60 \text{ sec/min} \\
 &= \underline{5.83\text{E}+1 \text{ gallons}}
 \end{aligned}$$

It is assumed that the accident occurs at C-106 tank farm. Then the volume (due to gravity) for the portion of approximately 2000 ft. 4 inch Sch. 40 piping (Ref. 11) is calculated as follows:

$$\begin{aligned}
 \text{Volume of 2000 ft of pipe} &= \pi \times (4.026 \text{ in})^2 / 4 \times 2000 \text{ ft} \times 12 \text{ in/ft} \\
 &= \underline{3.06\text{E}+5 \text{ in}^3}
 \end{aligned}$$

$$\text{Per Appendix B-11 of Ref. 1, } 231 \text{ in}^3 \text{ volume} = 1 \text{ gallon}$$

$$\begin{aligned}
 \text{Therefore, volume of liquid in the pipe} &= 3.06\text{E}+5 \text{ in}^3 / 231 \text{ in}^3 / \text{gal} \\
 &= \underline{1.32\text{E}+3 \text{ gallons}}
 \end{aligned}$$

$$\begin{aligned}
 \text{Total volume release} &= (5.83\text{E}+1) + (1.32\text{E}+3) \\
 &= \underline{1.38\text{E}+3 \text{ gallons}}
 \end{aligned}$$

At 3.785 liters per gallon, per Ref. 1,

$$\begin{aligned}
 \text{Effective radioactive volume release} &= 1.38\text{E}+3 \text{ gal} \times 3.785 \text{ liters/gal} \\
 &= \underline{5.22\text{E}+3 \text{ liters}}
 \end{aligned}$$

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Per Criterion No. 11 on Sh. 4, the release volume to the surface (forming a pool) is equivalent to 10 percent of the total release, which is equal to

$$= 5.22\text{E}+3 \times 0.1$$

$$= \underline{5.22\text{E}+2 \text{ liter}}$$

Per Criterion 12 on Sh. 5, the release fraction for liquid release aerosol (from the pool surface) is 1.0E-5.

Therefore, the amount of hazardous material suspended is

$$= 1.0\text{E}-5 \times 5.22\text{E}+2$$

$$= \underline{5.22\text{E}-3 \text{ liters}}$$

Solids Radioactive Release Calculation

Based on 30 percent solids (per Criterion 15, Sh. 5)

Effective radioactive volume release

$$= 5.22\text{E}-3 \times 0.30 / 1.2$$

$$= \underline{1.31\text{E}-3 \text{ liters}}$$

From Table A-1, offsite unit liter release EDE inhalation dose for C-106 solids is equal to 2.30E-2 rem/liter without PM.

Therefore, for a total volume release of 1.31E-3 liters, the offsite EDE

$$= 2.30\text{E}-2 \text{ rem/liter} \times 1.31\text{E}-3 \text{ liter}$$

$$= \underline{3.01\text{E}-5 \text{ rem}}$$

Similarly from Table A-2, onsite unit liter release EDE inhalation dose for C-106 solids is equal to 4.37E+1 rem/liter.

Therefore, for a total volume release of 1.31E-3 liters, the onsite EDE

$$= 4.37\text{E}+1 \text{ rem/liter} \times 1.31\text{E}-3 \text{ liter}$$

$$= \underline{5.72\text{E}-2 \text{ rem}}$$

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Liquid Radioactive Release Calculation

Based on 70 percent supernatant liquid
 (per Criterion 15, Sh. 5),

Effective radioactive volume release $= 5.22\text{E-}3 \times 0.70$
 $= \underline{3.65\text{E-}3 \text{ liters}}$

From Table A-1, offsite unit liter release EDE inhalation dose for AY-102 (with AY-101 supernatant) is equal to $1.80\text{E-}5$ rem/liter without PM.

Therefore, for a total volume release of $3.65\text{E-}3$ liters,
 the offsite EDE $= 1.80\text{E-}5 \text{ rem/liter} \times 3.65\text{E-}3 \text{ liter}$
 $= \underline{6.57\text{E-}8 \text{ rem}}$

Similarly from Table A-2, onsite unit liter release EDE inhalation dose for AY-102 (with AY-101 supernatant) is equal to $3.42\text{E-}2$ rem/liter.

Therefore, for a total volume release of $3.65\text{E-}3$ liters,
 the onsite EDE $= 3.42\text{E-}2 \text{ rem/liter} \times 3.65\text{E-}3 \text{ liter}$
 $= \underline{1.25\text{E-}4 \text{ rem}}$

EDE dose consequence for Offsite Individual for both solids and liquid,
 Scenario No. 1c $= \underline{3.0\text{E-}5 \text{ rem}}$

EDE dose consequence for Onsite Individual for both solids and liquid,
 Scenario No. 1c $= \underline{5.7\text{E-}2 \text{ rem}}$

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1.d SCENARIO NO. 1d: WASTE TRANSFER RETURN LINE BREAK (Seismic Switch Shutoff and 10 Seconds Release Time)

From Scenario 1c calculations, Sheet 14, the amount of hazardous material suspended is = 5.22E-3 liters

Solids Radioactive Release Calculation

The return line has 10 percent solids (per Criterion 15, Sh. 5)

Effective radioactive volume release = $5.22\text{E-}3 \times 0.10$
 = 5.22E-4 liters

From Table A-1, offsite unit liter release EDE inhalation dose for C-106 solids is equal to 2.30E-2 rem/liter without PM.

Therefore, for a total volume release of 5.22E-4 liter, the offsite EDE = $2.30\text{E-}2 \text{ rem/liter} \times 5.22\text{E-}4 \text{ liter}$
 = 1.20E-5 rem

Similarly from Table A-2, onsite unit liter release EDE inhalation dose for C-106 solids is equal to 4.37E+1 rem/liter.

Therefore, for a total volume release of 5.22E-4 liter, the onsite EDE = $4.37\text{E+}1 \text{ rem/liter} \times 5.22\text{E-}4 \text{ liter}$
 = 2.28E-2 rem

Liquid Radioactive Release Calculation

Based on 90 percent supernatant liquid (per Criterion 15 on Sheet 5),

Effective radioactive volume release = $5.22\text{E-}3 \times 0.90$
 = 4.70E-3 liters

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From Table A-1, offsite unit liter release EDE inhalation dose for AY-102 with AY-101 supernatant is equal to $1.80\text{E-}5$ rem/liter without PM.

Therefore, for a total volume release of $4.70\text{E-}3$ liters,
 the offsite EDE $= 1.80\text{E-}5$ rem/liter x $4.70\text{E-}3$ liter
 $= \underline{8.46\text{E-}8 \text{ rem}}$

Similarly from Table A-3, onsite unit liter release EDE inhalation dose for AY-102 (with AY-101 supernatant) is equal to $3.42\text{E-}2$ rem/liter.

Therefore, for a total volume release of $4.70\text{E-}3$ liters,
 the onsite EDE $= 3.42\text{E-}2$ rem/liter x $4.70\text{E-}3$ liter
 $= \underline{1.61\text{E-}4 \text{ rem}}$

EDE dose consequence for Offsite Individual for both solids and liquid,
 Scenario No. 1d $= \underline{1.2\text{E-}5 \text{ rem}}$

EDE dose consequence for Onsite Individual for both solids and liquid,
 Scenario No. 1d $= \underline{2.3\text{E-}2 \text{ rem}}$

Form 5007 (Rev. 4/83)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET	CALCULATION SET NO. 7362-H-006			REV.	COMP. BY	CHK'D BY
PROJECT <u>241-C-106 Waste Retrieval</u> SUBJECT <u>Dose Consequences and EHSC Calculations</u>		PRELIM.	FINAL X	VOID	0	NSV DATE 4-22-94	[Signature] DATE 4-22-94
		SHEET 20 OF 48				DATE	DATE
		J.O. 7362.007					

EVALUATION OF BOTH SUPPLY AND RETURN LINE BREAK WITH SEISMIC SWITCH SHUTOFF

EDE dose consequence for Offsite Individual for both solids and liquid,
 Scenarios 1c and 1d

$$= (3.0E-5) + (1.2E-5) \text{ rem}$$

$$= \underline{4.2E-5 \text{ rem}}$$

EDE dose consequence for Onsite Individual for both solids and liquid,
 Scenarios 1c and 1d

$$= (5.7E-2) + (2.3E-2) \text{ rem}$$

$$= \underline{8.0E-2 \text{ rem}}$$

From above, the EDE dose consequence values are $< 500 \text{ mrem}$ for offsite individual and $< 5 \text{ rem}$ for onsite individual doses. Therefore, the waste transfer lines can be classified as SC-3, per Ref. 9. Further confirmation is needed via Environmental Hazard Safety Classification (EHSC) per EP 1.4 (Ref. 9) to determine the safety classification. See Step B.2 for EHSC calculation for Scenarios 1c and 1d.

Based on the onsite EDE values (using release time of 10 seconds), the time to reach the SC-1, Criterion 1 of Ref. 9 (500 mrem for offsite individual), will be equal to 33 hours [$500 \text{ mrem} / (4.2E-5 \text{ rem} \times 1000 \text{ mrem/rem} \times (3600 \text{ sec/hr} / 10 \text{ sec}))$], and the time to reach the SC-2, Criterion 2 of Ref. 9 (5 rem for onsite individual), will be equal to 10.4 minutes [$5 \text{ rem} / (8.02E-2 \text{ rem} \times (60 \text{ sec/min} / 10 \text{ sec}))$]. Since the estimated time (10.4 minutes) to reach SC-2 criterion does not provide sufficient time to implement recovery actions, the waste transfer lines should be classified as SC-2.

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Bedger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO. 7362-H-006		REV.	COMP. BY	CHK'D BY
				PRELIM.	FINAL	VOID	0	NSV DATE 4-22-94
	PROJECT 241-C-106 Waste Retrieval			SHEET 21 OF 48			DATE	DATE
	SUBJECT Dose Consequences and EHSC Calculations			J.O. 7362.007				

2. SCENARIO NO. 2: UNFILTERED RELEASE THROUGH HVAC RECIRCULATION DUCT BREAK

Unfiltered release through the HVAC recirculation duct break scenario is schematically shown in Figure 2 below.

SCHEMATIC DIAGRAM
C-106
VENTILATION SYSTEM
FIGURE 2

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO.	REV.	COMP. BY	CHK'D BY
				7362-H-006			
	PRELIM.	FINAL	VOID		0	NSV DATE 4-22-94	NSV DATE 4-22-94
	PROJECT <u>241-C-106 Waste Retrieval</u>			SHEET 22 OF 48		DATE	DATE
SUBJECT <u>Dose Consequences and EHSC Calculations</u>				J.O. 7362.007			

Per Criterion 9 on Sheet 4, recirculation ventilation flow is 860 scfm. Per Appendix D of FDC (Ref. 12) the vapor space capacity is 100 mg/m³. The vapor specific gravity has been taken as equivalent to 1.0 for calculation purposes since vapor space is 100 percent humid and 90 percent supernatant is in the vapor space per FDC.

Assumed that recirculation duct has breached, and the unfiltered vapor from C-106 exhaust ventilation is flowing through the break into the environment.

30 Minutes Release

Therefore, the volume release will be

$$= 100 \text{ mg/m}^3 \times (1\text{g}/1000\text{mg}) \times 860 \text{ ft}^3/\text{min} \times 30 \text{ min} \times (1\text{m}^3/35.3 \text{ ft}^3) \\ \times 1\text{cc}/1\text{g} \times 1 \text{ L}/1000\text{cc} \times 1/1.0$$

$$= \underline{0.073 \text{ liter}}$$

Using this value in Tables A-1 and A-2, the offsite and onsite EDE dose consequences without PM are calculated as follows:

EDE dose consequence for Offsite Individual for

Tank C-106,

$$\text{Scenario No. 2} = (2.30\text{E-}2) \times 0.10 \times 0.073 + (1.80\text{E-}5) \times 0.90 \times 0.073 \\ = \underline{1.7\text{E-}4 \text{ rem}}$$

EDE dose consequence for Onsite Individual for

Tank C-106 (without plume meander effect),

$$\text{Scenario No. 2} = (4.37\text{E-}1) \times 0.10 \times 0.073 + (3.42\text{E-}2) \times 0.90 \times 0.073 \\ = \underline{3.2\text{E-}1 \text{ rem}}$$

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO.		REV.	COMP. BY	CHK'D BY
				7362-H-006				
	PRELIM.	FINAL	VOID	0	NSV	DATE	DATE	
		X			4-22-94	4-22-94		
PROJECT <u>241-C-106 Waste Retrieval</u>			SHEET 23 OF 46			DATE	DATE	
SUBJECT <u>Dose Consequences and EHSC Calculations</u>			J.O. 7362.007					

Following evaluations are made for 1 and 2 hour release times for determining the impact on safety classification of HVAC:

1 Hour Release

EDE dose consequence for Offsite Individual for
 Tank C-106 with PM,
 Scenario No. 2 $= (1.7E-4) \times (0.64) / 0.5$
 $= \underline{2.2E-4 \text{ rem}}$

EDE dose consequence for Onsite Individual for
 Tank C-106 (Table A-3, with plume meander effect),
 Scenario No. 2 $= (1.40E+1) \times 0.10 \times 0.073 \times 2 + (1.10E-2) \times 0.90 \times 0.073 \times 2$
 $= \underline{2.1E-1 \text{ rem}}$

2 Hour Release, plume meander effect will be considered per Appendix C, Ref. 13

Per Tables A-1 and A-3, offsite and onsite EDE doses are calculated as follows:

EDE dose consequence for Offsite Individual for
 Tank C-106, Scenario No. 2 $= (2.2E-4) \times 2$
 $= \underline{4.4E-4 \text{ rem}}$

EDE dose consequence for Onsite Individual for
 Tank C-106 (with plume meander effect,
 Table A-3 data used),
 Scenario No. 2 $= 2.1E-1 \times 2$
 $= \underline{4.2E-1 \text{ rem}}$

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO.	REV.	COMP. BY	CHK'D BY
PROJECT <u>241-C-106 Waste Retrieval</u>			7362-H-006	0	NSV DATE 4-22-94	[Signature] DATE 4-22-94	
SUBJECT <u>Dose Consequences and EHSC Calculations</u>			SHEET 24 OF 48		DATE	DATE	
			J.O. 7362.007				

Based on the onsite individual EDE value with plume meander effect, the time to reach SC-2, Criterion No. 2 per Ref. 9 (5 rem for onsite individual), will be equal to 24 hours.

From the Scenario 2 calculations, the EDE dose consequence values are < 500 mrem for offsite individual and < 5 rem for onsite individual doses. The active new HVAC system shall be classified as SC-3, per Ref. 8 and Ref. 9 based on dose rates. Further confirmation is needed via Environmental Hazard Safety Classification (EHSC) calculation per EP-1.4 (Ref. 9) to determine the safety classification of the new HVAC. See step B.3 for EHSC calculation for this Scenario No. 2.

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO.		REV.	COMP. BY	CHK'D BY	
				7362-H-006					
				PRELIM.	FINAL	VOID	0	NSV DATE 4-22-94	SJA DATE 4-22-94
				X					
PROJECT <u>241-C-106 Waste Retrieval</u>	SHEET 25 OF 48			DATE	DATE				
	J.O. 7362.007								
	SUBJECT <u>Dose Consequences and EHSC Calculations</u>								

3. SPRAY LEAK FROM A JUMPER OR CONNECTOR INSIDE THE PIT

3a. SCENARIO NO. 3a: SPRAY LEAK FROM A JUMPER OR CONNECTOR INSIDE THE PIT WITH COVER BLOCK OFF

Aerosol leak due to a spray leak from a jumper or connector misalignment inside the pit with pit cover block off is schematically shown in Figure 3a.

SCHEMATIC DIAGRAM
 C-106
 JUMPER LEAK
 FIGURE 3a
 (COVER BLOCK REMOVED)

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO.	REV.	COMP. BY	CHK'D BY			
PROJECT <u>241-C-106 Waste Retrieval</u>				7362-H-006			DATE <u>4-22-94</u>			
SUBJECT <u>Dose Consequences and EHSC Calculations</u>				PRELIM.	FINAL	VOID				
				X		0	NSV DATE <u>4-22-94</u>			
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				J.o. 7362.007			DATE			

Assumed that the jumper or connector is breached, and the waste liquid is postulated to leak as an atomizing spray with the pit cover off and the release time is assumed to be 2 hours per Criterion 2 on Sheet 3.

From Table 3 of Ref. 10, the respirable liquid aerosol generation for droplets of < 50 microns is interpolated as noted below. Viscosity, density, surface tension, and temperature of liquid have already been considered in Ref. 10. The leak is modeled as the discharge from a sharp-edged orifice of size 0.1 mm wide and 5 cm long. The width of the leak is assumed to be 0.1 mm which is the worst case opening for producing droplets. The length is assumed to be 5 cm which is a typical circumference of an opening for a valve stem.

At 150 psi, value is $9.94\text{E-}5 \text{ m}^3/\text{sec}$

At 200 psi, value is $1.26\text{E-}4 \text{ m}^3/\text{sec}$

Therefore, for a 50 psi differential, the interpolated value is $2.66\text{E-}5 \text{ m}^3/\text{sec}$

Therefore, for 180 psi system/pump pressure (without any significant error due to interpolation method noted above),

the aerosol generation rate = $9.94\text{E-}5 + (2.66\text{E-}5 \times 30/50)$

$$= \underline{1.15\text{E-}4 \text{ m}^3/\text{sec}}$$

Therefore, volume of material suspended
in 2 hours

$$= 1.15\text{E-}4 \text{ m}^3/\text{sec} \times 2 \text{ hours} \times 3600 \text{ sec/hr} \times 1000 \text{ L/m}^3$$

$$= \underline{8.31\text{E+}2 \text{ liters}}$$

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO.		REV.	COMP. BY	CHK'D BY
				7362-H-006				
	PRELIM.	FINAL	VOID	0	NTV DATE 4-22-94	DATE 4-22-94		
	PROJECT 241-C-106 Waste Retrieval			SHEET 27 OF 48				
SUBJECT Dose Consequences and EHSC Calculations			J.o. 7362.007					

Solids Radioactive Release Calculations

Based on 30 percent solids criteria per Criterion 15, Sh. 5,

the effective volume of material suspended that will be released to the environment will be

$$= 8.31E+2 \times 0.3 / 1.2$$

$$= \underline{2.08E+2 \text{ liters}}$$

Therefore, substituting this value in Tables A-1 and A-3, the EDE dose consequences are calculated as noted below for the Scenario No. 3a:

EDE dose consequence for Offsite Individual with PM for C-106 solids = 3.1E+0 rem

EDE dose consequence for Onsite Individual for C-106 solids = 2.9E+3 rem

Liquid Radioactive Release Calculations

Based on 70 percent liquid criteria per Criterion 15, Sh. 5,

the effective volume of material suspended that will be released to the environment will be

$$= 8.31E+2 \times 0.7$$

$$= \underline{5.81E+2 \text{ liters}}$$

Therefore, substituting this value in Tables A-1 and A-3, the EDE dose consequences are calculated as noted below for the Scenario No. 3a:

EDE dose consequence for Offsite Individual with PM for liquid = 6.7E-3 rem

EDE dose consequence for Onsite Individual for liquid = 6.4E+0 rem

EDE dose consequence for Offsite Individual for solids and liquid, Scenario No. 3a = 3.1E+0 rem

EDE dose consequence for Onsite Individual for solids and liquid, Scenario No. 3a = 2.9E+3 rem

From the above, the EDE dose consequence values are > 500 mrem for offsite individual and > 5 rem for onsite individual doses. Therefore, the pit covers shall be classified as SC-1, per Ref. 7.

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO. 7362-H-006		REV.	COMP. BY	CHK'D BY
				PRELIM.	FINAL	VOID	0	NSV DATE 6-22-94
	PROJECT <u>241-C-106 Waste Retrieval</u>			SHEET 28 OF 48			DATE	DATE
	SUBJECT <u>Dose Consequences and EHSC Calculations</u>			U.o. 7362.007				

3b. SCENARIO NO. 3b: SPRAY LEAK FROM A JUMPER OR CONNECTOR INSIDE THE PIT WITH COVER BLOCK ON (WITH ALL OPENINGS CLOSED)

Aerosol leak due to a spray leak from a jumper or connector misalignment inside the pit with pit cover block on with all openings closed is schematically shown in Figure 3b.

SCHEMATIC DIAGRAM
C-106
JUMPER LEAK
FIGURE 3b
 (COVER BLOCK INSTALLED)

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO.	REV.	COMP. BY	CHK'D BY
				7362-H-006			
	PRELIM.	FINAL	VOID		0	NDV DATE 4-22-94	DATE 4-22-94
	SHEET 29 OF 48						
PROJECT <u>241-C-106 Waste Retrieval</u>				J.O. 7362.007		DATE	DATE
SUBJECT <u>Dose Consequences and EHSC Calculations</u>							

Assumed that the jumper or connector is breached, and the waste liquid is postulated to leak as an atomizing spray with the pit cover on with all openings closed, and the release time is assumed to be 8 hours, which is equivalent to one shift.

The mechanism for dispersing the aerosol from the pit to the environment is through the leak paths around the cover blocks. The volume of waste is calculated as the basis that the pit drains are plugged and the leak detection system has failed to detect leaks accumulated in the pit for approximately 8 hours.

The leak is modeled as the discharge from a sharp-edged orifice of size 0.1 mm wide and 5 cm long. The width of the leak is assumed to be 0.1 mm which is the worst case opening for producing droplets. The length is assumed to be 5 cm which is typical circumference of an opening for a valve stem. The discharge rate through the orifice is calculated, per Ref. 1, as follows:

$$Q = 0.61 A \sqrt{2 g h}$$

Where

- Q = Discharge rate, m³/sec
- 0.61 = Discharge coefficient for a sharp edged orifice
- A = Area of opening, m²
- g = Gravitational acceleration, 9.8 m/sec²
- h = system pressure (pumping pressure) expressed as water head, m

From Appendix B-11 of Ref. 1, 1 psig = 27.7 inches of water.

$$\begin{aligned} \text{Therefore, } h &= 180 \text{ psig} \times 27.7 \text{ inches/psig} \times 2.54 \text{ cm/inch} \times 1 \text{ m}/100 \text{ cm} \\ &= \underline{1.27\text{E}+2 \text{ m}} \end{aligned}$$

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO.		REV.	COMP. BY	CHK'D BY		
				7362-H-006						
	PROJECT	241-C-106 Waste Retrieval			PRELIM.	FINAL	VOID	0	NSV DATE 4-22-94	DATE 4-22-94
	SUBJECT	Dose Consequences and EHSC Calculations			SHEET 30 OF 48				DATE	DATE
					J.O. 7362.007					

and slit area, $A = (0.1 \text{ mm} \times 1 \text{ m}/1000 \text{ mm}) \times (5 \text{ cm} \times 1 \text{ m}/100 \text{ cm})$

$$= 5.0\text{E-}6 \text{ m}^2$$

Substituting in the above equation,

$$Q = 1.52\text{E-}4 \text{ m}^3/\text{sec}$$

The displacement rate of moist air from the pit is taken to be equal to the discharge rate from the leak. With the release time of 8 hours, which is equivalent to length of one shift,

The amount of waste volume released to the environment is

$$= 1.52\text{E-}4 \text{ m}^3/\text{sec} \times 8 \text{ hrs} \times 3600 \text{ sec/hr}$$

$$= 4.38 \text{ m}^3$$

Solids Radioactive Release Calculation

At 30 percent solids content per Criterion 15 on Sheet 5, the volume released to the environment is

$$= 4.38 \text{ m}^3 \times 0.30 \times 1/1.2$$

$$= 1.1 \text{ m}^3$$

Assuming the aerosol capacity is equivalent to 10 mg/m³ per Ref. 3, the amount of aerosol released to

the environment

$$= 1.1 \text{ m}^3 \times 10 \text{ mg/m}^3 \times 1\text{E-}6 \text{ L/mg}$$

$$= 1.10\text{E-}5 \text{ liters}$$

Therefore, substituting this value in Tables A-1 and A-3, the EDE dose consequences are calculated as noted below for the Scenario No. 3b:

EDE dose consequence for Offsite Individual with PM for C-106 solids = 1.6E-7 rem

EDE dose consequence for Onsite Individual for C-106 solids = 1.5E-4 rem

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO. 7362-H-006		REV.	COMP. BY	CHK'D BY
				PRELIM.	FINAL	VOID	0	NSV DATE 4-22-94
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	SUBJECT Dose Consequences and EHSC Calculations			J.o. 7362.007				

Liquid Radioactive Release Calculation

At 70 percent liquid content per Criteria 15, the volume released to the environment is

$$= 4.38 \text{ m}^3 \times 0.70$$

$$= \underline{3.07 \text{ m}^3}$$

Assuming the aerosol capacity is equivalent to 10 mg/m³ per Ref. 3, the amount of aerosol released to the environment

$$= 3.07 \text{ m}^3 \times 10 \text{ mg/m}^3 \times 1\text{E-}6 \text{ L/mg}$$

$$= \underline{3.07\text{E-}5 \text{ liters}}$$

Therefore, substituting this value in Table A-1 and A-3, the EDE dose consequences are calculated as noted below for the Scenario No. 3b:

EDE dose consequence for Offsite Individual with PM for liquid = 3.5E-10 rem

EDE dose consequence for Onsite Individual for liquid = 3.4E-7 rem

EDE dose consequence for Offsite Individual for solids and liquid, Scenario No. 3b = 1.6E-7 rem

EDE dose consequence for Onsite Individual for solids and liquid, Scenario No. 3b = 1.5E-4 rem

From the above, based on 8 hour release, the dose consequence values are < 500 mrem for offsite individual and < 5 rem for onsite individual doses. Therefore, the pit covers should always be covered prior to any transfer operation via the jumpers/connectors within the pit.

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO. 7362-H-006			REV.	COMP. BY	CHK'D BY		
				PRELIM.	FINAL	VOID	0	NDV DATE 4-22-94	S.H.H. DATE 4-22-94		
				X			SHEET 32 OF 48			DATE	DATE
				J.O. 7362.007							
PROJECT <u>241-C-106 Waste Retrieval</u>											
SUBJECT <u>Dose Consequences and EHSC Calculations</u>											

3c. SCENARIO NO. 3c: SPRAY LEAK FROM A JUMPER OR CONNECTOR INSIDE THE PIT WITH COVER BLOCK ON (WITH 2 INCH VALVE HANDLE HOLE OPEN)

Aerosol leak due to a spray leak from a jumper or connector misalignment inside the pit with pit cover block on with 2 in valve handle hole open is schematically shown in Figure 3c.

SCHEMATIC DIAGRAM
C-106
JUMPER LEAK
FIGURE 3c

(COVER BLOCK INSTALLED AND
 2" INSPECTION PLUG REMOVED)

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO.		REV.	COMP. BY	CHK'D BY
				7362-H-006				
	PRELIM.	FINAL	VOID	0	NSV	DATE	DATE	
		X			4-22-94	4-22-94		
PROJECT <u>241-C-106 Waste Retrieval</u>				SHEET 33 OF 48			DATE	DATE
SUBJECT <u>Dose Consequences and EHSC Calculations</u>				J.O. 7362.007				

Assumed that the jumper or connector is breached, and the waste liquid is postulated to leak as an atomizing spray with the pit cover on with 2 inch valve handle hole open for portable imaging system, and the release time is assumed to be 8 hours which is equivalent to one shift.

The mechanism for dispersing the aerosol from the pit to the environment is through the leak paths around the cover blocks. The volume of waste is calculated on the basis that the pit drains are plugged and the leak detection system has failed to detect leaks accumulated in the pit for approximately 8 hours.

The leak is modeled as the discharge from a sharp-edged orifice of size 0.1 mm wide and 5 cm long. The width of the leak is assumed to be 0.1 mm which is the worst case opening for producing droplets. The length is assumed to be 5 cm which is typical circumference of an opening for a valve stem. The discharge rate through the orifice is calculated, per Ref. 1, as follows:

$$Q = 0.61 A \sqrt{2 g h}$$

Where

- Q = Discharge rate, m³/sec
- 0.61 = Discharge coefficient for a sharp edged orifice
- A = Area of opening, m²
- g = Gravitational acceleration, 9.8 m/sec²
- h = system pressure (pumping pressure) expressed as water head, m

From Appendix B-11 of Ref. 1, 1 psig = 27.7 inches of water.

Therefore, h = 180 psig x 27.7 inches/psig x 2.54 cm/inch x 1 m/100 cm
 = 1.27E+2 m

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET			CALCULATION SET NO.		REV.	COMP. BY	CHK'D BY		
				7362-H-006						
	PROJECT	241-C-106 Waste Retrieval			PRELIM.	FINAL	VOID		NTV	
					X		0	DATE	DATE	
SUBJECT			Dose Consequences and EHSC Calculations			SHEET 34 OF 46				
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and 2 inch square edged

$$\begin{aligned} \text{opening area, } A &= \pi \times \{(2.0 \text{ inch} \times 2.54 \text{ cm/inch})^2 / 4\} \times (1 \text{ m}/100 \text{ cm})^2 \\ &= \underline{2.03\text{E-}3 \text{ m}^2} \end{aligned}$$

Substituting in the above equation,

$$Q = \underline{0.062 \text{ m}^3/\text{sec}}$$

The displacement rate of moist air from the pit is taken to be equal to the discharge rate from the leak. With the release time of 8 hours, which is equivalent to the length of one shift,

The amount of waste volume released

$$\begin{aligned} \text{to the environment is} &= 0.062 \text{ m}^3/\text{sec} \times 8 \text{ hrs} \times 3600 \text{ sec/hr} \\ &= \underline{1.77\text{E}+3 \text{ m}^3} \end{aligned}$$

Solids Radioactive Release Calculation

At 30 percent solids content per Criterion 15 on Sheet 5, the volume released to

$$\begin{aligned} \text{the environment is} &= 1.77\text{E}+3 \text{ m}^3 \times 0.30 \times 1/1.2 \\ &= \underline{4.44\text{E}+2 \text{ m}^3} \end{aligned}$$

Assuming the aerosol capacity is equivalent to 10 mg/m³ per Ref. 3, the amount of aerosol released to

$$\begin{aligned} \text{the environment} &= 4.44\text{E}+2 \text{ m}^3 \times 10 \text{ mg/m}^3 \times 1\text{E-}6 \text{ L/mg} \\ &= \underline{4.44\text{E-}3 \text{ liters}} \end{aligned}$$

Therefore, substituting this value in Table A-1 and A-3, the EDE dose consequences are calculated as noted below for the Scenario No. 3c:

$$\text{EDE dose consequence for Offsite Individual with PM for C-106 solids} = \underline{6.5\text{E-}5 \text{ rem}}$$

$$\text{EDE dose consequence for Onsite Individual for C-106 solids} = \underline{6.2\text{E-}2 \text{ rem}}$$

Form 5007 (Rev. 4/93)

Raytheon Engineers & Constructors A Consolidation of Badger and UE&C	GENERAL COMPUTATION SHEET	CALCULATION SET NO. 7362-H-006			REV. 	COMP. BY 	CHK'D BY
PROJECT <u>241-C-106 Waste Retrieval</u> SUBJECT <u>Dose Consequences and EHSC Calculations</u>		PRELIM. 	FINAL X	VOID 	0	NSV DATE 4-22-94	S.H.H. DATE 4-22-94
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		J.O. 7362.007					

Liquid Radioactive Release Calculation

At 70 percent liquid content per Criterion 15 on Sheet 5,
the volume released to the environment is

$$= 1.77\text{E}+3 \text{ m}^3 \times 0.70$$

$$= \underline{1.24\text{E}+3 \text{ m}^3}$$

Assuming the aerosol capacity is equivalent to 10 mg/m³ per Ref. 3,
the amount of aerosol released to
the environment

$$= 1.24\text{E}+3 \text{ m}^3 \times 10 \text{ mg/m}^3 \times 1\text{E}-6 \text{ L/mg}$$

$$= \underline{1.24\text{E}-2 \text{ liters}}$$

Therefore, substituting this value in Tables A-1 and A-3, the EDE dose consequences are calculated as noted below for the Scenario No. 3c:

EDE dose consequence for Offsite Individual with PM for liquid = 1.4E-7 rem

EDE dose consequence for Onsite Individual for liquid = 1.4E-4 rem

EDE dose consequence for Offsite Individual for solids and liquid,
Scenario No. 3c = 6.5E-5 rem

EDE dose consequence for Onsite Individual for solids and liquid,
Scenario No. 3c = 6.2E-2 rem

From the above, based on a release time of 8 hours, the dose consequence values are < 500 mrem for offsite individual and < 5 rem for onsite individual doses. However, in order to be within the ALARA guidelines of Ref. 6, it is recommended that the valve handle opening shall be opened with necessary precautions and proper procedures, and pits/boxes monitored for leaks.

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				7362-H-006				
	PROJECT <u>241-C-106 Waste Retrieval</u> SUBJECT <u>Dose Consequences and EHSC Calculations</u>	PRELIM.	FINAL	VOID	0	NTV DATE 4-22-94	DAH DATE 4-22-94	
			X					
SHEET 38 OF 48				DATE	DATE			
J.O. 7362.007								

B. EHSC CALCULATIONS

1. EHSC CALCULATIONS FOR SCENARIOS 1a and 1b

Following work sheets from EP-1.4 (Ref. 9) determine the EHSC for both Waste Transfer Supply Line and Return Line Break scenarios with liquid release time of 2 Hours at 350 gpm flow plus line gravity flow.

RADIOACTIVE MATERIAL EHSC CALCULATION																										
COMPONENT/SYSTEM: <u>Supply and Return Transfer Line Break</u>		DATE: <u>04/22/94</u>	Sheet 1 of 2																							
DRAWING: _____		LOCATION: _____																								
1. MATERIAL FORM Is the radioactive material dispersible (i.e., other than a consolidated or stabilized solid)? Dispersible materials include: liquids, sludges, gases, powders, and unconsolidated solids. ☐ No Stop. Associated items are safety class 3, other criteria permitting. ☒ Yes Proceed.																										
2. QUANTITY OF MATERIALS RELEASED The following matrix provides multiplying factors which are a function of the total estimated curie content postulated to be released to the environment from the system(s), component(s), and/or structure(s) of interest. From Tables B-2 and B-3, Group radioisotopes by half-life ($T_{1/2}$) category and show total amount (in Curies (Ci)) for each category for release time of 8 hours: $T_{1/2} < 1$ yr: Amount = <u>0.00</u> Ci $T_{1/2} = 1$ to 100 yrs: Amount = <u>2.48E+5</u> Ci $T_{1/2} > 100$ yrs: Amount = <u>3.71E+2</u> Ci Mark the applicable multiplying factors below.		MF(2.) = 11,000																								
<table border="1"> <thead> <tr> <th rowspan="2">Amount (Ci)</th> <th colspan="3">Half-life</th> </tr> <tr> <th><1 yr.</th> <th>1-100 yrs.</th> <th>>100 yrs.</th> </tr> </thead> <tbody> <tr> <td><1</td> <td>MF=1</td> <td>MF=10</td> <td>MF=100</td> </tr> <tr> <td>1-1000</td> <td>MF=10</td> <td>MF=100</td> <td>MF=1,000</td> </tr> <tr> <td>>1000-10⁶</td> <td>MF=100</td> <td>MF=1,000</td> <td>MF=10,000</td> </tr> <tr> <td>>10⁶</td> <td>MF=1,000</td> <td>MF=10,000</td> <td>MF=100,000</td> </tr> </tbody> </table>				Amount (Ci)	Half-life			<1 yr.	1-100 yrs.	>100 yrs.	<1	MF=1	MF=10	MF=100	1-1000	MF=10	MF=100	MF=1,000	>1000-10 ⁶	MF=100	MF=1,000	MF=10,000	>10 ⁶	MF=1,000	MF=10,000	MF=100,000
Amount (Ci)	Half-life																									
	<1 yr.	1-100 yrs.	>100 yrs.																							
<1	MF=1	MF=10	MF=100																							
1-1000	MF=10	MF=100	MF=1,000																							
>1000-10 ⁶	MF=100	MF=1,000	MF=10,000																							
>10 ⁶	MF=1,000	MF=10,000	MF=100,000																							
MF(2.) = SUM of underlined MFs = 1000+10000 = 11000																										

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		X						
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RADIOACTIVE MATERIAL EHSC CALCULATION	
COMPONENT/SYSTEM: <u>Supply and Return Transfer Lines Break</u> DATE: <u>04/22/94</u> Sheet 2 of 2 DRAWING: _____ LOCATION: _____	
3. TOTAL QUANTITY OF MATERIAL RELEASED ⁽¹⁾ = <u>3.28E+5</u> liter (8.66E+4 gallon)	MF(3.) = 100
≤100 gal MF=1 >1,000-10,000 gal ... MF=10 >100-1,000 gal ... MF=5 ≥10,000 gal MF=100	
4. PROXIMITY TO ENVIRONMENTAL RECEIVERS	
A. DEPTH TO AQUIFER = <u>>150</u> feet	MF(4.A) = 1
>150 feet MF=1 21-75 feet MF=5 76-150 feet MF=2 0-20 feet MF=10	
B. DISTANCE TO SENSITIVE SURFACE WATER ⁽²⁾⁽⁴⁾ = <u>NA</u> feet	MF(4.B) = NA
>2500 feet NA 100-500 feet MF=20 >1000-2500 feet MF=9 <100 feet MF=25 >500-1000 feet MF=16 Direct discharge to surface water ... MF=50	
C. DISTANCE TO OFFSITE BOUNDARY ⁽⁴⁾ = <u>NA</u> feet	MF(4.C) = NA
>2500 feet ... NA 1000-2500 feet ... MF=5 <1000 feet ... MF=10	
5. CALCULATION OF ENVIRONMENTAL IMPACT FOR SAFETY CLASSIFICATION The EHSC is determined as follows where each term is the multiplying factor (MF) from the respective paragraphs above. Terms 4.A, 4.B, and/or 4.C are applied where legitimate pathways to the environment exist. EHSC = (2.)(3.)(4.A)(4.B)(4.C) = (11000)(100)(1)(NA)(NA) = 1,100,000 EHSC = 1,100,000, Safety Class 1	

⁽¹⁾ Total quantity of radioactive material postulated to be released to the environment.

⁽²⁾ 1 gallon = 3.785 liters, 1 gallon = 8.34 lbs (at 4°C, a specific gravity of 1, and atmospheric pressure)

⁽³⁾ Examples of sensitive surface waters are the Columbia River and West Lake.

⁽⁴⁾ Airborne pathways do not apply for this sensitive surface water or offsite boundary criterion (Table 1, Criterion 12, Ref. 9).

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				PRELIM.	FINAL	VOID	0	NSV DATE 4-22-94
	PROJECT <u>241-C-106 Waste Retrieval</u>			SHEET 38 OF 46			DATE	DATE
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2. EHSC CALCULATIONS FOR SCENARIOS 1c and 1d

Following work sheets from EP-1.4 (Ref. 9) determine the EHSC for both Waste Transfer Supply Line and Return Line Break scenarios with liquid release time for 10 seconds after a seismic shutoff plus line gravity flow.

RADIOACTIVE MATERIAL EHSC CALCULATION																							
COMPONENT/SYSTEM: <u>Supply and Return transfer Line Break</u>		DATE: <u>04/22/94</u>	Sheet 1 of 2																				
DRAWING: _____		LOCATION: _____																					
1. MATERIAL FORM Is the radioactive material dispersible (i.e., other than a consolidated or stabilized solid)? Dispersible materials include: Sludges, shudges, gases, powders, and unconsolidated solids. ☐ No Stop. Associated items are safety class 3, other criteria permitting. ☒ Yes Proceed.																							
2. QUANTITY OF MATERIALS RELEASED The following matrix provides multiplying factors which are a function of the total estimated curie content postulated to be released to the environment from the system(s), component(s), and/or structure(s) of interest. From Tables B-2 and B-3, Group radionuclides by half-life ($T_{1/2}$) category and show total amount (in Curies (Ci)) for each category for release time of 8 hours: $T_{1/2} < 1$ yr: Amount = <u>0.00</u> Ci $T_{1/2} = 1$ to 100 yrs: Amount = <u>7.83E+3</u> Ci $T_{1/2} > 100$ yrs: Amount = <u>1.18E+1</u> Ci Mark the applicable multiplying factors below. <table border="1"> <thead> <tr> <th>Amount (Ci)</th> <th><u><1 yr.</u></th> <th><u>1-100 yrs.</u></th> <th><u>>100 yrs.</u></th> </tr> </thead> <tbody> <tr> <td><1</td> <td>MF=1</td> <td>MF=10</td> <td>MF=100</td> </tr> <tr> <td>1-1000</td> <td>MF=10</td> <td>MF=100</td> <td>MF=1,000</td> </tr> <tr> <td>>1000-10⁴</td> <td>MF=100</td> <td>MF=1,000</td> <td>MF=10,000</td> </tr> <tr> <td>>10⁴</td> <td>MF=1,000</td> <td>MF=10,000</td> <td>MF=100,000</td> </tr> </tbody> </table>		Amount (Ci)	<u><1 yr.</u>	<u>1-100 yrs.</u>	<u>>100 yrs.</u>	<1	MF=1	MF=10	MF=100	1-1000	MF=10	MF=100	MF=1,000	>1000-10 ⁴	MF=100	MF=1,000	MF=10,000	>10 ⁴	MF=1,000	MF=10,000	MF=100,000	MF(2.) = 2000 MF(2.) = SUM of underlined MFs = 1000 + 1000 = 2000	
Amount (Ci)	<u><1 yr.</u>	<u>1-100 yrs.</u>	<u>>100 yrs.</u>																				
<1	MF=1	MF=10	MF=100																				
1-1000	MF=10	MF=100	MF=1,000																				
>1000-10 ⁴	MF=100	MF=1,000	MF=10,000																				
>10 ⁴	MF=1,000	MF=10,000	MF=100,000																				

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		X				4-22-94	4-22-94	
PROJECT <u>241-C-106 Waste Retrieval</u>				SHEET 39 OF 46			DATE	DATE
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RADIOACTIVE MATERIAL EHSC CALCULATION	
COMPONENT/SYSTEM: <u>Supply and Return Transfer Line Break</u>	DATE: <u>04/22/94</u>
DRAWING: _____	LOCATION: _____
Sheet 2 of 2	
3. TOTAL QUANTITY OF MATERIAL RELEASED⁽¹⁾ = <u>1.04E+4</u> liter (2.76E+3 gallons)	
≤ 100 gal MF = 1 $> 1,000$ - $10,000$ gal ... MF = 10 > 100 - $1,000$ gal ... MF = 5 $\geq 10,000$ gal MF = 100	MF(3.) = 10
4. PROXIMITY TO ENVIRONMENTAL RECEIVERS	
A. DEPTH TO AQUIFER = <u>> 150</u> feet	
> 150 feet MF = 1 21 - 75 feet MF = 5 76 - 150 feet MF = 2 0 - 20 feet MF = 10	MF(4.A) = 1
B. DISTANCE TO SENSITIVE SURFACE WATER⁽²⁾ = <u>NA</u> feet	
> 2500 feet NA 100 - 500 feet MF = 20 > 1000 - 2500 feet MF = 9 < 100 feet MF = 25 > 500 - 1000 feet MF = 16 Direct discharge to surface water ... MF = 50	MF(4.B) = NA
C. DISTANCE TO OFFSITE BOUNDARY⁽³⁾ = <u>NA</u> feet	
> 2500 feet ... NA 1000 - 2500 feet ... MF = 5 < 1000 feet ... MF = 10	MF(4.C) = NA
5. CALCULATION OF ENVIRONMENTAL IMPACT FOR SAFETY CLASSIFICATION	
The EHSC is determined as follows where each term is the multiplying factor (MF) from the respective paragraphs above. Terms 4.A, 4.B, and/or 4.C are applied where legitimate pathways to the environment exist.	
$EHSC = (2.)(3.)(4.A)(4.B)(4.C) = (2000)(10)(1)(NA)(NA) = 20,000$	
$EHSC = 20,000$, Safety Class 3	

⁽¹⁾ Total quantity of radioactive material postulated to be released to the environment.

⁽²⁾ 1 gallon = 3.785 liters, 1 gallon = 8.34 lbs (at 4°C, a specific gravity of 1, and atmospheric pressure)

⁽³⁾ Examples of sensitive surface waters are the Columbia River and West Lake.

⁽⁴⁾ Airborne pathways do not apply for this sensitive surface water or offsite boundary criterion (Table 1, Criterion 12, Ref. 9).

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		X						
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Table B-1 Radionuclide Source Term (Ref. 13)

ISOTOPE	C-106, SLUDGE (See Notes below)		AY-102, SUPERNATANT(See Notes Below)	
	Bq/L	CURIES/L	Bq/L	CURIES/L
Am-241	5.56E+7	1.50E-3		
C-14	1.23E+4	3.32E-7		
Co-60	4.50E+7	1.22E-3		
Cs-137	1.75E+10	4.73E-1	4.03E+9	1.09E-1
I-129	4.29E+3	1.16E-7		
Pu-239/240	1.42E+8	3.84E-3		
Sr-90	1.05E+11	2.84E+0		
Tc-99	1.16E+7	3.14E-4		
U-238	1.53E+4	4.14E-7		

Notes

- Values are taken from Ref. 13, See Table C-7 for AY-102 (with AY-101 supernatant) and Table C-9 for C-106 solids
- Per Criterion 14 on Sh. 5, C-106 will have supernatant from AY-101 via AY-102.
- Curies/ L = (Bq/ L) / (3.7E + 10)

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Table B-2. Release Curies (data from Table B-1)
Scenarios 1a and 1c (Supply Line)

Isotope	Half-life Yrs.	Total Release, Curies	
		2 hr (Scenario 1a, See Note 1)	Seismic shutoff (Scenario 1c, See Note 2)
Am-241	432.2	7.38E+1	2.35E+0
C-14	5730	1.63E-2	5.20E-4
Co-60	5.27	6.00E+1	1.91E+0
Cs-137	30.17	3.58E+4	1.14E+3
I-129	1.57E+7	5.71E-3	1.82E-4
Pu-239/240	2.41E+4	1.80E+2	6.01E+0
Sr-90	28.6	1.40E+5	4.45E+3
Tc-99	2.13E+5	1.54E+1	4.92E-1
U-238	4.47E+9	2.04E-2	6.48E-4

Notes:

- Based on Release Volume of $1.64E+5$ Liters ($4.33E+4$ gallons, from Scenario 1a and Criterion 15)
- Based on Release Volume of $5.22E+3$ Liters ($1.38E+3$ gallons, from Scenario 1c and Criterion 15)
- Example Calc. for total release in curies: Cs-137 and 2 hour release
 From Table B-1, $(4.73E-1) \times 0.30 \times (1.64E+5) + (1.09E-1) \times 0.70 \times (1.64E+5) = 3.58E+4$ Curies

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Table B-3. Release Curies (Data from Table B-1)
Scenarios 1b and 1d (Return Line)

Isotope	Half-life Yrs.	Total Release, curies	
		2 hr (Scenario 1b, See Note 1)	Seismic shutoff (Scenario 1d, See Note 2)
Am-241	432.2	2.46E+1	7.83E-1
C-14	5730	5.44E-3	1.73E-4
Co-60	5.27	2.00E+1	6.37E-1
Cs-137	30.17	2.38E+4	7.59E+2
I-129	1.57E+7	1.90E-3	6.06E-5
Pu-239/240	2.41E+4	6.30E+1	2.00E+0
Sr-90	28.6	4.66E+4	1.48E+3
Tc-99	2.13E+5	5.15E+0	1.64E-1
U-238	4.47E+9	6.79E-3	2.16E-4

Notes:

- Based on Release Volume of 1.64E+5 Liters (4.33E+4 gallons, from Scenario 1b and Criterion 15)
- Based on Release Volume of 5.22E+3 Liters (1.38E+3 gallons, from Scenario 1d and Criterion 15)
- Example Calc. for total release in curies: Cs-137 and 2 hour release
From Table B-1, $(4.73E-1) \times 0.10 \times (1.64E+5) + (1.09E-1) \times 0.90 \times (1.64E+5)$
= 2.38E+4 Curies

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3. EHSC CALCULATIONS FOR SCENARIO NO. 2, UNFILTERED RELEASE
DUE TO HVAC RECIRCULATION LINE BREAK

From Tables B-2 and B-3, the curies per liter values are low for 10 % C-106 solids and 90 % AY-102 supernatant (from AY-101). When these numbers are multiplied by the release volume of (0.073 X 4) liter, release curies will be insignificant and do not warrant any further calculations. Therefore, HVAC should be classified as SC-3, as determined per the dose calculations under Scenario 2.

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PROJECT <u>241-C-106 Waste Retrieval</u>		PRELIM. 	FINAL X	VOID 	0	<u>NDV</u> DATE 4-22-94	<u>NDV</u> DATE 4-22-94
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SUMMARY AND CONCLUSIONS

Scenario 1: WASTE TRANSFER SUPPLY AND RETURN LINES BREAK

Scenarios 1a and 1b: Waste Transfer Supply and Return Lines Break, 2 hour Release Time

The calculated offsite EDE dose value for both supply and return lines break, Scenarios 1a and 1b, is 8.4E-1 mrem. This is within the allowable limit of 500 mrem of Ref. 9. Similarly, the calculated onsite EDE dose value is 8.1E-1 rem which is within the allowable limit of 5 rem per Ref. 9. SC-3 classification is required based on the dose calculations per Criterion 3 of Ref. 9

EHSC calculation was performed in Step B.1, and the calculated EHSC value of 1,100,000 is greater than the value of 1,000,000. SC-1 classification is required to be specified for both supply and return transfer lines per Criterion 10 of Ref. 9. Project W320 decided to install a seismic switch in order to reduce the impact on the personnel safety and health hazards due to dose consequences, and the environment via soil contamination due to large quantity of waste spill on to the ground during transfer lines break, especially if the system continues to operate for a longer periods of time for lack of necessary instrumentation before the system is shutdown. See summary and conclusions for Scenarios 1c and 1d for line breaks with seismic shutoff.

Scenarios 1c and 1d: Waste Transfer Supply and Return Lines Break, 10 Seconds Release Time With Seismic Shutoff

The calculated offsite EDE dose value for both supply and return lines break, Scenarios 1c and 1d, is 4.2E-2 mrem. This is within the allowable limit of 500 mrem of Ref. 9. Similarly, the calculated onsite EDE dose value is 8.0E-2 rem which is within the allowable limit of 5 rem per Ref. 9. SC-3 classification is required based on the dose calculations per Criterion 3 of Ref. 9

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EHSC calculation was performed in Step B.2, and the calculated EHSC value of 20,000 is less than the value of 500,000. SC-3 classification is acceptable based on EHSC calculations per Criterion 13 of Ref. 9

Based on the onsite EDE values (using release time of 10 seconds), the time to reach the SC-1, Criterion 1 of Ref. 9 (500 mrem for offsite individual), will be equal to 33 hours [$500 \text{ mrem} / (4.2\text{E-}5 \text{ rem} \times 1000 \text{ mrem/rem} \times (3600 \text{ sec/hr} / 10 \text{ sec}))$], and the time to reach the SC-2, Criterion 2 of Ref. 9 (5 rem for onsite individual), will be equal to 10.4 minutes [$5 \text{ rem} / (8.02\text{E-}2 \text{ rem} \times (60 \text{ sec/min} / 10 \text{ sec}))$]. Since the estimated time (10.4 minutes) to reach SC-2 criterion does not provide sufficient time to implement recovery actions, the waste transfer lines should be classified as SC-2.

Scenario 2: UNFILTERED RELEASE DUE TO HVAC RECIRCULATION DUCT BREAK

The calculated offsite EDE dose value of $4.4\text{E-}1 \text{ mrem}$ is within the allowable limit of 500 mrem, and the calculated onsite EDE dose value of $4.2\text{E-}1 \text{ rem}$ is also within the allowable limit of 5 rem per Ref. 9 for a 2 hour air borne release to the environment. EHSC calculation was not performed as noted in Step B-3 due to very low release volume to the environment. Based on the onsite EDE values, the time to reach SC-2 per Criterion 2 of Ref. 2, will be equal to 24 hours.

Therefore, the new ventilation system should be classified as SC-3 per Ref. 8 and Ref. 9.

Scenario 3: SPRAY LEAK FROM A JUMPER OR CONNECTOR INSIDE THE PIT

Scenario No. 3a: Spray leak from a Jumper or Connector inside the pit with cover block off, and a release time of 2 hours

The calculated offsite EDE dose value of $3.1\text{E}+3 \text{ mrem}$ exceeds the allowable limit of 500 mrem, and also, the calculated onsite EDE dose value of $2.9\text{E}+3 \text{ rem}$ exceeds the allowable limit of 5 rem per Ref. 9. This calculation confirms that waste transfer pit covers provide confinement of aerosol release to the

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environment, and therefore, should remain in place prior to transfer operation. See evaluation for Scenarios 3b and 3c.

Scenario No. 3b: Spray leak from a Jumper or Connector inside the pit/box with cover block on with all the openings closed, and a release time of 8 hours

The calculated offsite EDE dose value of 1.6E-4 mrem is well within the allowable limit of 500 mrem, and also, the calculated onsite EDE dose value of 1.5E-4 rem is well within the allowable limit of 5 rem per Ref. 9.

Therefore, the pit covers should always be installed prior to any transfer operation via the jumpers/connectors within the pit.

Scenario No. 3c: Spray leak from a Jumper or Connector inside the pit/box with cover block on and 2 inch valve handle hole open, and a release time of 8 hours

The calculated offsite EDE dose value of 6.5E-2 mrem is well within the allowable limit of 500 mrem, and also, the calculated onsite EDE dose value of 6.2E-2 rem is less than the allowable limit of 5 rem per Ref. 9.

However, in order to be within the ALARA guidelines of Ref. 6, it is recommended that the valve handle hole shall be opened with necessary precautions and proper procedures, and pits/boxes monitored for leaks.