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7. Abstract

Part of the effort to decommission the Hot Semiworks facility included filling Tank 241-CX-72 with grout in 1986. Later evaluation of the tank contents and sampling of a rod accidentally pulled out of the tank concluded that removal of the grout and sludge requires a 3 phase effort. This report provides a safety assessment of the Phase 1 activities which includes developing drilling equipment and procedures, removing and disposing of the upper grout layers from the tank. Phase 1 activities have a low hazard classification with the equipment classification no higher than Safety Class 3. Operational Safety Limits ensure safe drilling operations.

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LIST OF TERMS

ALARA	as low as reasonably achievable
DOE	U.S. Department of Energy
DWP	Decommissioning Work Procedure
HEPA	high-efficiency particulate air (filter)
HWOP	Hazardous Waste Operations Permit
JSA	Job Safety Analysis
OSL	operational safety limits
PUREX	Plutonium/Uranium Extraction (Facility)
REDOX	Reduction/Oxidation Plant
RWP	Radiation Work Permit
SARP	Safety Analysis Report, Packaging
TRU	transuranic (waste)

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PHASE 1 GROUT REMOVAL FROM TANK 241-CX-72 SAFETY ASSESSMENT

1.0 INTRODUCTION AND SUMMARY

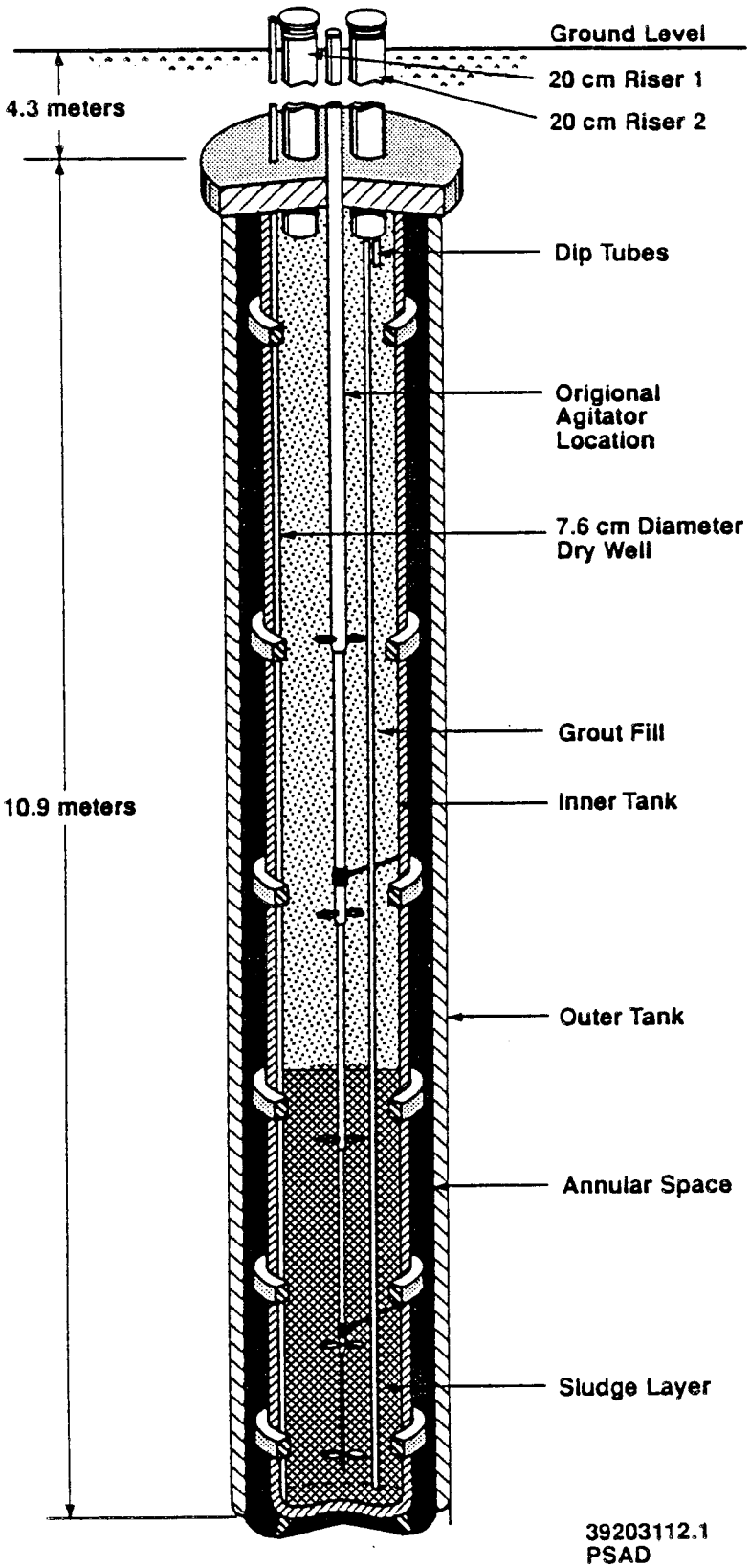
Part of the effort to decommission the Hot Semiworks facility included filling Tank 241-CX-72 with grout in 1986. The grout eliminated voids in the tank, providing one more step toward eventual entombment of the system. In 1988 a piece of heavy equipment accidentally pulled a 4.6-m-long agitator rod out of the tank. Examination of the rod showed it to be contaminated, requiring burial as low-level waste. The contaminated rod prompted further investigation of the tank contents including researching prior reports and performing nondestructive analysis of the tank contents by lowering various flux detectors into the 7.6-cm dry well located on the inside of the tank. The research and analysis efforts (WHC 1989a) concluded that the tank may contain high levels of transuranic (TRU) materials. Therefore, the grout and sludge in Tank 241-CX-72 require removal and sampling activities. The sample results will determine how much of the retrieved wastes require classification as TRU material.

Tank 241-CX-72, a 101.6-cm-diameter, 10.87-m-tall carbon steel tank (Figure 1) installed below ground level in 1956, received liquid wastes from simulated Plutonium/Uranium Extraction (Facility) (PUREX) operations at the Hot Semiworks facility. The wastes remained in the tank and were concentrated. During concentration, operators used a hand operated agitator to assess sludge characteristics. A level measurement in 1974 reported a sludge level of 189 cm with 2.5 cm of liquid. In 1978 sludge level and visual inspection information indicated an empty tank. Based on the 1978 and later "records" of an empty tank, and a 1986 measurement, operations filled the tank in 1986 with grout.

Examination of the rod uncovered in 1988 provided evidence that Tank 241-CX-72 may contain TRU wastes. Subsequent nondestructive characterization of Tank 241-CX-72 contents identified three distinct layers existing in the tank (WHC 1990a). The bottom 3 m topped with 0.6 m of contaminated grout mixed with sludge contains the vast majority of the TRU material. The intermediate layer (approximately 5.8 m) consists of grout with small amounts of cesium mixed with grout (probably residual waste clinging to the tank sidewall). The upper 1.5 m of grout should contain little or no contamination.

An evaluation of decommissioning alternatives for Tank 241-CX-72 contents concluded that removal of the grout and sludge requires a 3-phase effort (WHC 1990b). Phase 1 includes developing drilling equipment and removing and disposing of the uncontaminated and slightly contaminated upper grout layers. Phase 1 activities end upon removal of these layers and upon completion of the preparation of Tank 241-CX-72 for Phase 2 activities. Phase 2 includes core sampling of the remaining tank material, analyzing and characterizing the remaining wastes, and developing the equipment and strategies for removal of the remaining material. Phase 3, assuming that the material is TRU, provides for removal and disposal of the TRU material and tank stabilization.

Figure 1. Simplified Cross Sectional View of Tank 241-CX-72.



This report only pertains to the safety assessment for Phase 1. Phases 2 and 3 will require a separate safety assessment. This safety assessment records the hazards and operational limitations required to ensure safe operation of the drilling equipment, vacuum equipment, and ancillary equipment for grout removal and packaging. This report includes an assessment of the effectiveness of the equipment and processes for safe removal of the grout. Completion of this safety assessment satisfies the requirements of WHC 1988 and DOE 5481.1B (DOE 1986).

1.1 ASSESSMENT SUMMARY

The radiological and toxicological dose consequences for this nonreactor nuclear activity satisfy the criteria for low hazard classification (WHC 1988, 1990b) while drilling through the upper 4.7 m of grout (Section 3.3). The operations have a potential for exceeding low hazard classification limits when drilling at depths greater than 4.7 m (Section 3.3). The radiological evaluation document (WHC 1989a) and source term internal letter (Bendixsen 1992, Appendix A) provide the technical bases for the radiological and toxicological calculations which support the hazard classifications. Nondestructive data, the PUREX flowsheet, and prior tank data provide the estimated source term for the radionuclide content in Tank 241-CX-72 (Section 2.6). The concentrations expected from Phase 1 activities should not result in hazardous exposures to facility workers and should be below regulatory requirement limits at drill depths less than 4.7 m. Calculations show the estimated potential airborne concentrations are below the risk acceptance criteria for onsite and offsite individuals. In accordance with the low hazard classification for Phase 1 activities, the equipment is no higher than Safety Class 3 for components, equipment, or systems. For the purposes of Phase 1 activities, the exhauster is considered to be Safety Class 3.

Drilling operations below a depth of 4.7 m can continue with continuous beta and gamma monitoring of the powder transfer line and the powder sampler together with radiation levels below 1.0 rem/h (Speer 1992, Appendix C). Drilling operations below a depth of 4.7 m shall proceed only with continuous beta and gamma monitoring of the powder removed from Tank 241-CX-72. A detected radiation level above 1.0 rem/h for the sample filter shall cause shutdown of Phase 1 operations. At this time, an onsite assessment by safety, operations, and engineering will determine if Phase 1 operations should cease. No drilling below a depth of 7.6 m will be allowed until another safety assessment can be completed (WHC 1990a)

The insignificant amount of fissile material negates criticality as a potential safety concern. The source term bounds the retrieval, packaging, and transporting activities for Phase 1 operations.

Normal jobsite worker safety requirements contained in the Hazardous Waste Operations Permit (HWOP), Job Safety Analysis (JSA), and the Radiation Work Permit (RWP) will provide adequate occupational safety, respiratory, and skin protection for the workers. A prudent action provides for Independent Safety approval of these three worker safety documents before work commences. The readiness review process verifies conformance to safety practices and procedures before startup.

1.2 SUMMARY OF LIMITS AND PRUDENT ACTIONS

No unacceptable impacts are anticipated from Phase 1 activities while drilling the top 4.7 m of grout. However, application of controls apply for the described activities to minimize environmental impact and reduce exposures to as low as reasonably achievable (ALARA). These controls pertain to three receptor groups of concern: the operations worker, the uninvolved onsite person, and the offsite person. Three operational safety limits (OSL) ensure, conformance to regulatory requirements and for ALARA purposes. The OSLs minimize the potential for operations above the limits for a facility having a low hazard classification, minimize the potential for dust releases, and provide for storage and disposal of the retrieved grout in accordance with regulatory requirements. Also, prudent actions performed in the spirit of ALARA, further reduce potential hazardous material exposures to ALARA. The following briefly describes each OSL. (Refer to Section 4.0 for more detail on the OSL safety controls and prudent actions.)

- Prevent operations that would potentially result in the system having a moderate or high hazard classification. Radiation monitoring of the sample filter and analysis of the powder samples will ensure safe drilling operations. This OSL provides for drilling operations shutdown and evaluation if the limits are reached.

- Minimize the potential of dust release during the retrieval activities. Confinement and containment of the radionuclides and hazardous materials removed from Tank 241-CX-72 require checking the equipment and processes using the approved Decommissioning Work Procedure (DWP) (DWP-C-026-00010). The OSL requires that work will stop if any part of the operations indicate release of radioactivity or hazardous materials to the environs.

- Storing, packaging, and transporting the retrieved wastes from Tank 241-CX-72 will comply with applicable regulations to minimize the potential for release to the environment.

The prudent actions adopted for Phase 1 activities are as follows:

1. Equipment and materials removed from the work location will be monitored to ensure they are free of contamination

2. A disposal plan will provide for prompt removal of the retrieved material to the appropriate waste repository site

3. Activities will be conducted in compliance with HWOP, JSA, and RWP instructions.

2.0 BACKGROUND

2.1 HANFORD SITE/200 EAST AREA DESCRIPTION

The Hanford Site is a 1,450-km^2 (560-mi^2) tract of land located in Benton, Franklin, Adams, and Grant counties in southcentral Washington state.

The 200 East Area is located in the central part of the Hanford Site (Figure 2). The Hot Semiworks complex located in the eastcentral part of the 200 East Area was a chemical facility constructed to test existing and proposed separation processes and to evaluate full-scale equipment under actual conditions. The water table is approximately 88 m below the land surface (DOE 1987).

Hot Semiworks conducted process improvement and flow sheet refinement studies for the reduction-oxidation (REDOX) process and pilot plant investigations of PUREX recovery processes. Also, the facility conducted strontium recovery pilot plant studies and may have been used for americium and curium isotope recovery (WHC 1989b).

This safety assessment document discusses Tank 241-CX-72, a belowgrade tank, located in the southern portion of the Hot Semiworks complex (Figure 3). As shown in Figure 3, the Hot Semiworks complex includes buildings 2704-C (office building), 215-C (gas preparation building), and 276-C (solvent handling facility). Tank 241-CX-72 is southeast of the former Building 201-C, decommissioned in 1988. Tanks 241-CX-70 and 241-CX-71 are nearby tanks. Reference WHC 1990a contains detailed descriptions of the tanks, operating histories, locations, dimensions, layout, adjacent structures, and decommissioning status. This report only includes the description and background of Tank 241-CX-72.

2.2 TANK 241-CX-72 PHYSICAL DESCRIPTION AND BACKGROUND

Engineering drawings GE 1954a and GE 1954b locate the tank approximately 45 m southeast of Building 201-C and buried 4.3 m below grade. WHC 1990a and GE 1954b show the tank as an upright cylindrical vessel, with a diameter of 101.6 cm and length of 10.87 m. Five stiffener rings on the outside of the tank reinforce the ASTM A-7-52T carbon steel, 9.5-mm-thick vessel wall (Figure 1). Three rows of vertical guides connect the stiffener rings (WHC 1990c, GE 1954b) with cylindrical heaters mounted just above each stiffener ring.

Figure 2. Hanford Site Map Showing Public Highways.

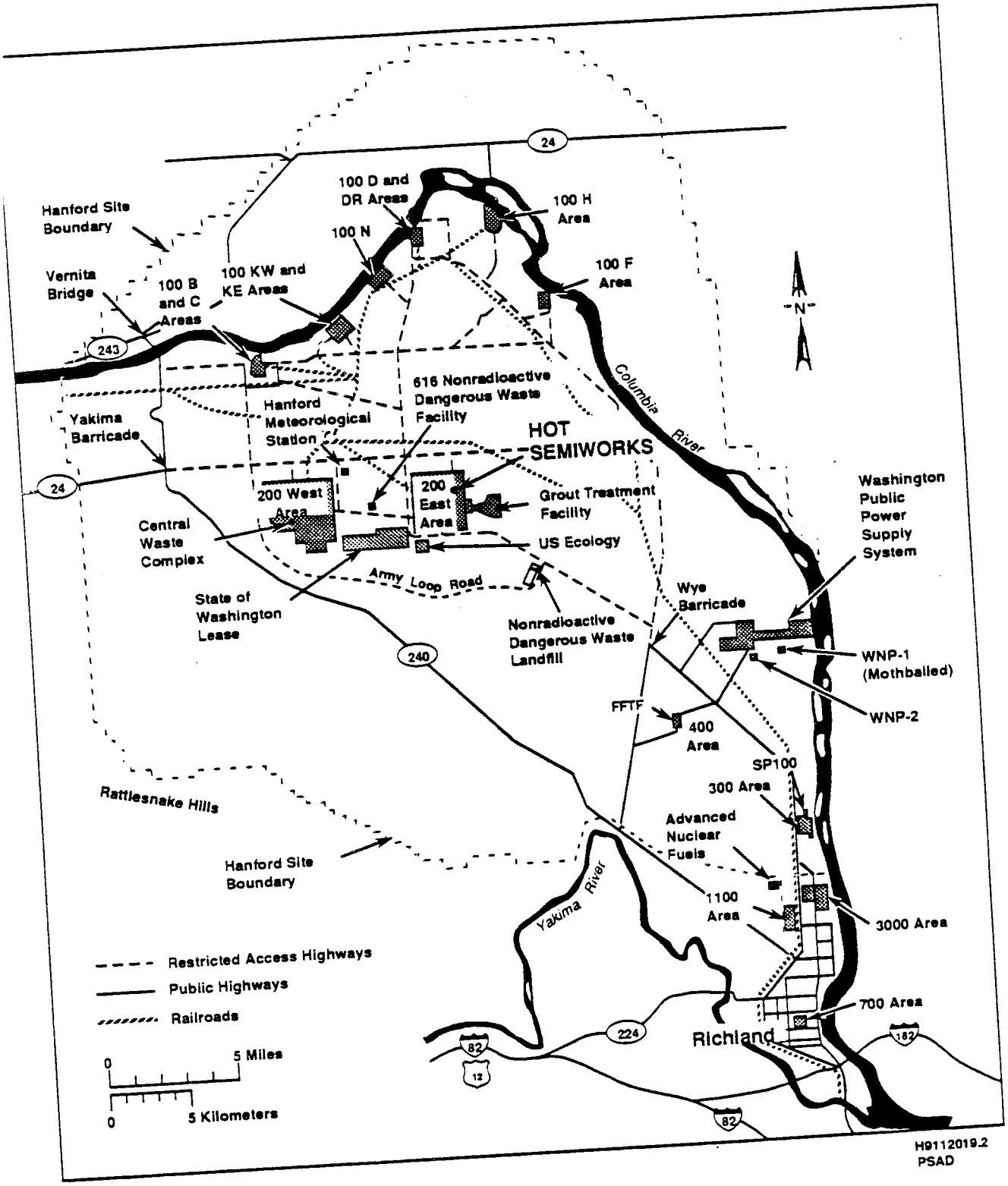
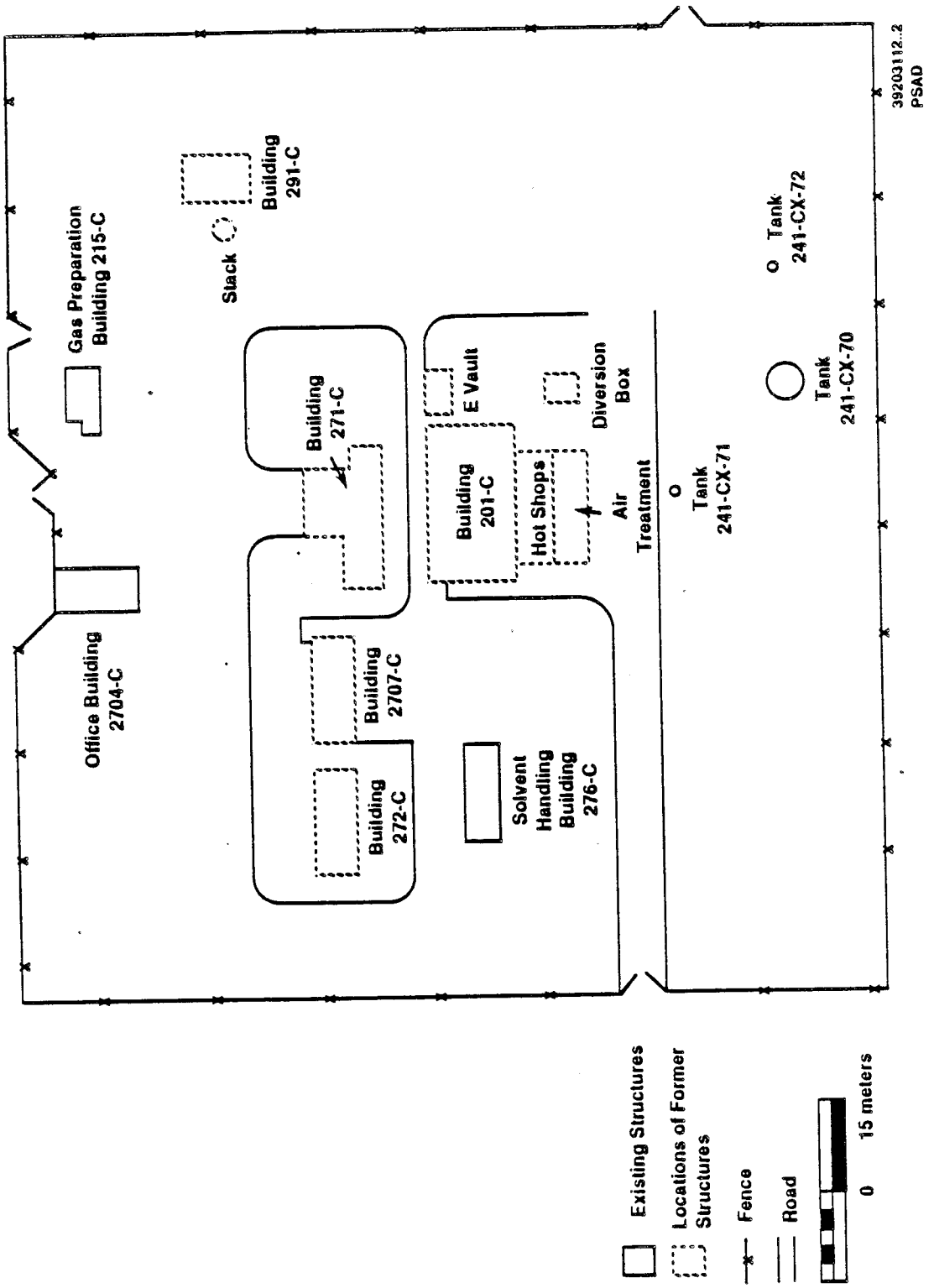


Figure 3. Hot Semiworks Site Facilities Map.



The tank wall extends beyond the bottom plate of the tank. Additional stiffeners reinforce and support the bottom plate. The tank internals include a 7.6-cm dry well pipe, a concentric paddle agitator rod assembly, a sparger pipe, and an instrument dip tube extending the length of the tank. The top plate contains, centrally located, two 20-cm-diameter access pipes, a 20-cm-diameter instrument pipe guide, a 7.6-cm-diameter vapor header pipe, and a 5-cm-diameter fill pipe.

Operators manually operated the concentric paddle assembly to assess sludge height and sludge stiffness properties within the tank. A sparger system near the bottom of the tank enhanced agitation of the tank contents (GE 1954a, GE 1954b, and WHC 1990c). The tank still contains the agitator assembly, the dry well, and sparger pipe.

A cylindrical caisson, 1.8 m in diameter and 10.9 m deep, fabricated from 13-mm carbon steel plate, encloses the Tank 241-CX-72 and stiffeners. A 30.5-cm-thick reinforced concrete pad supports both the caisson and tank. Reinforcing bars welded to the inside of the caisson tie the caisson to the base. Support pads welded to the bottom of the tank rest on the concrete base. A plate welded to the top of the tank extends over the caisson, acting as a seal for the caisson. A crane used two lifting lugs mounted on the tank to set the tank in place.

In 1986, decommissioning activities filled Tank 241-CX-72 with grout using inaccurate information that the tank was empty. Two years later, a trenching operation near the tank uncovered a 4.6-m rod that had been accidentally pulled out of the tank. Probably the earth movers adding the overburden earth layer had snagged the handle of the manual actuator rod and partially pulled it out of the tank. Examination of the manual agitator rod showed radioactive contamination on it. This prompted an investigation of the tank contents, which included finding relevant historical documents and starting nondestructive testing methods using the 7.6-cm dry well. These analyses indicated the tank contained TRU wastes. Radiation measurements taken inside of the dry well show three distinct regions existing in the tank. The bottom 3 m contain most of the dry sludge with approximately 0.6 m on top containing grout, which may be contaminated with TRU waste (WHC 1990b). The intermediate layer probably consists of grout mixed with residual cesium contamination that adhered to the tank walls and fill line. The top 1.5 m showed little or no radioactivity contamination levels.

Documents of past Hot Semiwork activities indicate Tank 241-CX-72 was installed in 1955. The tank received wastes for less than one year (probably 1956 or 1957) from pilot plant studies of the PUREX solvent extraction process. Pilot plant studies used Tank 241-CX-72 to study concentrating characteristics of aging waste.

2.3 PURPOSE

This report evaluates grout removal operations (Phase 1) designed for decontaminating and decommissioning of Tank 241-CX-72. This safety assessment demonstrates that dry mining operations are safe and satisfy the requirements of WHC 1988 and DOE 5481.1B, Safety Analysis and Review System (DOE 1986).

2.4 SCOPE

Phase 1 activities for Tank 241-CX-72 include installing a greenhouse over the top of the tank and installing vacuum and mining equipment. Phase 1 removes the uncontaminated and slightly contaminated grout layers from the tank and packages the grout for burial. Removing the grout overburden prepares the tank for Phase 2 and Phase 3 activities. Further decommissioning activities of the tank include core sampling and waste characterization (Phase 2), waste removal, packaging, disposal of grout and waste sludge, and tank decommissioning (Phase 3). The three phases for accomplishing decommissioning will use dry mining techniques (assuming that the sludge layer can be dry mined) for removing all waste and, thus, preclude wet sluicing operations (WHC 1990c and WHC 1990d). This report only covers Phase 1 activities and does not cover Phase 2 or Phase 3 activities or decommissioning activities for any of the other tanks or remaining structures at Hot Semiworks.

2.5 WORK DESCRIPTION

The key features of the Phase 1 grout removal operations include a heavy-duty, large diameter drill assembly and a heavy-duty industrial vacuum system for moving drill cuttings. The drill head assembly includes the tools for breaking the grout and fluidizing grout particles and pieces for conveyance to the vacuum loader equipment. The vacuum system consists of a blower, bag-house, hopper assembly, and a high-efficiency particulate air filter (HEPA) filtration system. The drill head assembly breaks the grout and waste into small pieces with simultaneous air fluidization for conveyance by the vacuum system.

Grout pieces and dust generated by the drill assembly impinge on a stationary plate inside of the baghouse assembly where a majority of the particulates fall into the hopper (i.e., the larger particles separate from the dust). The air continues around the stationary plate and flows through 27 bag filters. Dust accumulates on the exterior surface of the bag filter units. An automatic blow back system pulses pressurized air into 3 bag filters simultaneously to dislodge the accumulated dust, which then falls into the bottom hopper assembly. A sequencer and timer assembly pulses the bag filters 3 at a time in rotation providing for continuous flow through the remaining 24 bag filters. The baghouse filter assembly provides for greater than 99.9% removal efficiency for particles 1 micron or larger from the airflow. A HEPA filter, an integral part of the vacuum assembly downstream of the baghouse, provides for further separation of dust particulates before the exhausted air mixes with ventilation air flowing to the exhaust blower. Air from the vacuum assembly and greenhouse flows through the two-stage HEPA filtration unit past the stack monitoring system and up the exhaust stack.

2.5.1 Grout Retrieval

Phase 1 grout retrieval operations include loosening, grinding, and breaking grout and preparing the pieces, particles, and dust for conveyance to the vacuum system equipment. The current method uses a large diameter drill

to grind the grout into small pieces with air fluidization and conveying as the transport medium. The retrieval technique, while not fully developed, will include controls for confinement of the particles and dust, precluding contamination of the environment. Equipment for sampling and monitoring the retrieved grout ensures that burial containers meet packaging and burial criteria.

Grout retrieval will proceed smoothly and efficiently when operators, supervisors, and engineers operate the equipment within the design parameters established for the process. Control and monitoring of the equipment during Phase 1 grout removal operations requires instrumentation, equipment controls, procedures, and administrative controls. Redundancy of instruments and controls also helps the operators by providing backup instruments and controls if the primary fails in service. The minimum number of instruments, valves, and controls required for starting the drilling operations is listed in Table 1. These instruments and controls will provide for safe operation and reduce the potential for loss of confinement and contamination of the environment.

The amount of grout removed during Phase 1 will depend on sample analysis and radioactivity of the removed grout. Grout removal and packaging will continue until sample analysis indicates that grout TRU content is approaching the TRU waste limit or radioactivity of the sample increases above the limits of the grout burial box. Increasing sample TRU content or increasing radioactivity indicates that the drill bit is approaching the sludge layer. Grout retrieval will not exceed a depth of 7.6 m or 3 m from the inside bottom of Tank 241-CX-72 (WHC 1990a).

2.6 HAZARDS INVENTORY

This section contains a summary of the contents of Tank 241-CX-72 taken from various reports. The engineering report contains a comprehensive description of the tank and its contents (WHC 1990b). Previous reports record types and quantities of radionuclides in the tank using nondestructive assay techniques (Subrahmanyam 1989, WHC 1989b, WHC 1990b, WHC 1990e and WHC 1990f). These data indicate (1) a 3-m sludge layer at the bottom with a 0.6-m possible TRU mix of sludge and grout, (2) an intermediate grout layer (5.8-m thick) probably containing contamination from material adhering to the tank sidewall, and (3) an upper 1.5-m grout layer containing very little or no contamination. The Phase 3 activities will remove approximately 2.9 m^3 of TRU-contaminated material from Tank 241-CX-72 (WHC 1989c). This report covers removal of approximately 5.9 m^3 of grout from the tank.

This report includes the total amount of radionuclides estimated for Tank 241-CX-72. The total estimated amounts become the bounding source term for the tank. Phase 1 activities will remove only a tiny fraction of the total activity from the tank because the sludge layer and sludge-grout mix layer contain nearly all of the radionuclides. A recent letter (Bendixsen 1992, Appendix A) summarizes several methods employed to estimate the radioactive content of Tank 241-CX-72 and uses 3 new methods for estimating radionuclide content. Table 1 (Appendix A) estimates the ^{137}Cs content for the tank to range from 8,310 to 23,600 Ci and plutonium content to range from 20.9 to 100 g.

Table 1. Minimum Instruments and Controls for Safe Operations During Phase 1 Removal of Grout from Tank 241-CX-72. (2 sheets)

System or process feature	Function	Event mitigated or prevented
Drilling equipment		
Master key lock	Controls drill equipment control panel.	Prevents inadvertent operation during drill string add/remove activities.
Start/stop, jog, up/down switches	Controls up/down and on/off operations of the drill assembly.	Shuts down drilling operations during an emergency. Controls drill to prevent binding of the drill face with the grout.
Drill RPM control	Controls drill RPM.	Excessive RPM could damage the drill bit.
Vacuum equipment		
Start/stop switches, blowback timer and sequencer	Controls on/off operation of blower and bag filter blow back.	The switches are used to shut down the system if problems occur. The sequencer and blow back timer removes dust accumulation from the bag filters.
Filter differential pressure instruments	Indicates pressure drop across the bag filters.	Allows operators to adjust the blow back frequency and prevent plugging of the bag filters.
Vacuum loader flow meter	Measures flow of air through the vacuum system.	Provides a redundant indicator that the vacuum system is operating.
Caisson exhaust hose/HEPA filter and stack exhaust system		
HEPA filter differential pressure instruments	Measures differential pressure for the HEPA filters.	Indicates when the filters should be changed to prevent their clogging.
Start/stop switches	Provides startup and shutdown of the exhaust fan and exhaust monitoring equipment.	Shuts down the system if problems occur.

Table 1. Minimum Instruments and Controls for Safe Operations During Phase 1 Removal of Grout from Tank 241-CX-72. (2 sheets)

System or process feature	Function	Event mitigated or prevented
Exhauster flow indicator	Measures flow rate of exhaust gases.	Secondary indication that the ventilation exhaust system is operating.
Exhauster low-flow alarm	Alerts operators of low ventilation flow rate.	Provides warning that the ventilation system has shut down.
Continuous air monitors		
High alarms for caisson monitor and greenhouse monitor	Alarms when radioactivity levels become too high.	Alerts operators when either the caisson or the greenhouse airborne radioactivity exceeds the set point.
Powder sampler		
Switches and valves	Control sampling of powder.	Sample analysis shows TRU content and radioactive levels of the grout.
Radiation indicator	Measures radioactivity of powder.	Alerts operations when drill approaches the sludge layer.
Air compressor		
Air blow-down valve	Provides for periodic removal of accumulated water moisture.	Prevents caking of wet powder on the drill bit face and in the powder exhaust hose.
Automatic shutdown valve	Shuts when vacuum equipment is shut down or exhauster system is shut down.	Prevents pressurizing of the drill bit/grout interface on loss of vacuum.

HEPA = high-efficiency particulate air (filter).
RPM = revolutions per minute.
TRU = transuranic (waste).

In addition, an estimate based on a 1 to 2% plutonium loss during process development yielded 100 to 200 g of plutonium in the waste (Subrahmanyam 1989). Subrahmanyam also used the measured neutron flux, three possible isotopic compositions, and two chemical forms (oxide and fluoride) of the elements to estimate the plutonium content. These estimates "very strongly" suggested that the TRU compounds are present as fluoride compounds. Calculations of plutonium content using realistic and reasonable isotopic compositions yielded a conservative estimate of 150 to 200 g (WHC 1989c, WHC 1990f).

Based on these determinations, the upper bound estimate of waste in the tank includes 200 g of plutonium and 23,600 Ci of ^{137}Cs. Based on these values, the upper bounding concentrations of radionuclides in the waste are 5,770 nCi/g of plutonium and 15.4 Ci/L of ^{137}Cs (Appendix A).

3.0 HAZARD ASSESSMENT

3.1 INTRODUCTION

This hazard assessment of the dry mining operation on Tank 241-CX-72 uses the hazard classification approach (WHC 1988) to determine the consequences from a breach in the ventilation system during dry mining. Using the hazard classification technique does not necessarily reveal the mostly likely upset the project could face but merely tries to establish an upper bound on consequences from a postulated event. In theory, no other release will present consequences worse than those estimated using the hazard classification approach.

The results of the hazard classification analysis are presented as a function of drill depth from the top of Tank 241-CX-72. As the drill bit descends into the tank, it encounters increasing concentrations of ^{137}Cs and ^{239}Pu. The increasing concentrations, of course, have an effect on the consequences to the hypothetical onsite and Hanford Site boundary receptors. Based on the results, the project exceeds the criteria for a low hazard classification where the drill bit reaches 4.7 m from the top of the tank. Beyond the 4.7-m depth, the project has the potential for meeting moderate hazard or high hazard classification criteria.

3.2 ASSESSMENT

A hazard classification, also known as a facility hazard classification, is an analysis tool for the U.S. Department of Energy (DOE) to rank facilities as either high, moderate, or low hazard. Based on the facility's hazard classification, future safety documentation associated with the facility will receive varying degrees of attention all the way from review by the contractor for a low classification to DOE headquarter's review for a high classification. The hazard classification technique does not allow credit to be taken for any engineered or administrative barriers unless doing so increases the consequences of the postulated event. Using the hazard classification technique does not get at the mostly likely upset the project could face, but

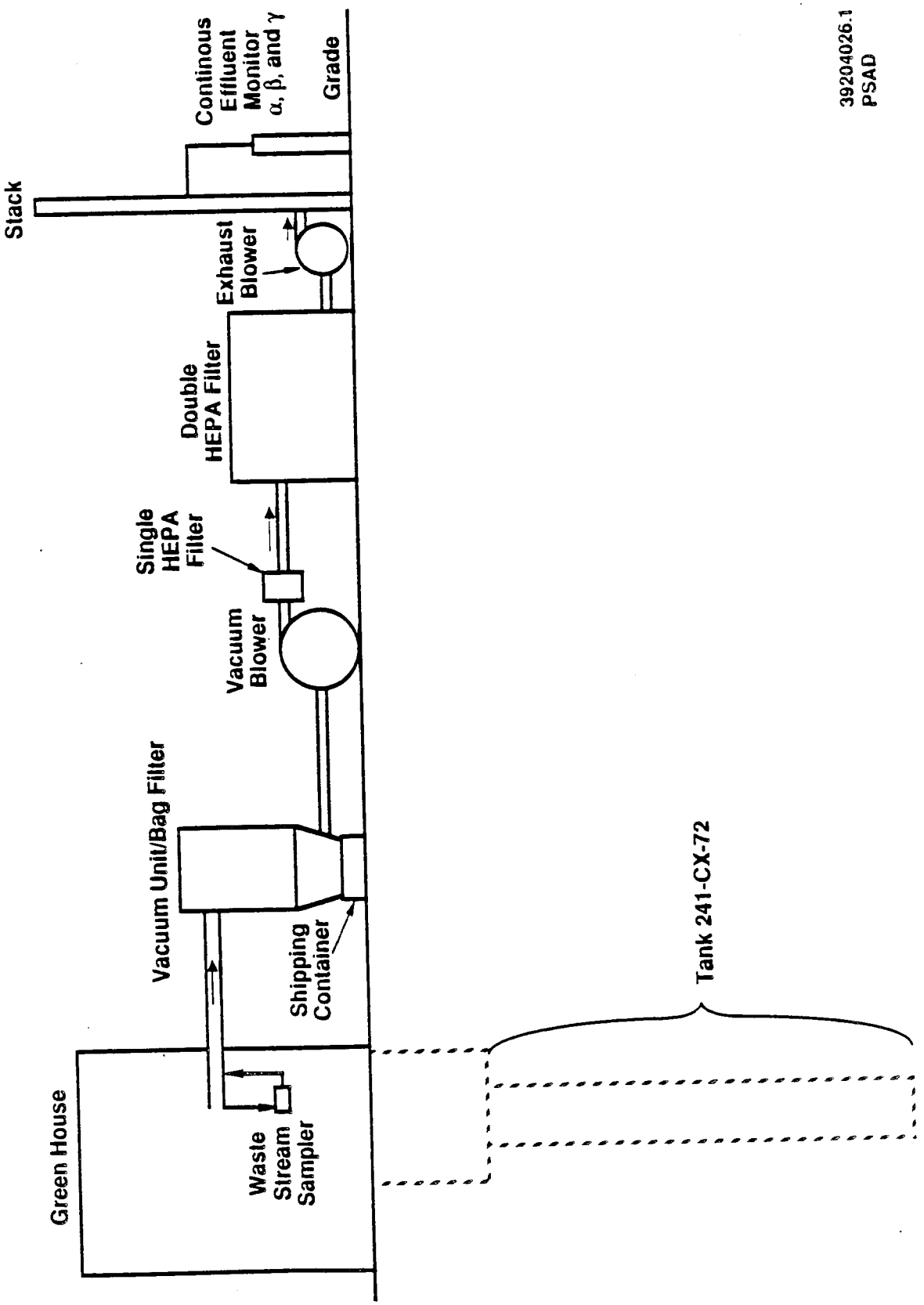
merely tries to establish an upper bound on consequences. In theory, no other release will present consequences worse than those estimated using the hazard classification approach.

To develop the hazard classification, a release event is postulated. Figure 4 shows a simple drawing of the system. Upstream of the fan, the system is at a negative pressure with respect to the atmosphere. A break in the piping system supplying the fan would not allow the material to escape to the environment. However, downstream of the fan the system is at a positive pressure with respect to the environment. A break in the system here would allow material to be injected into the atmosphere and become available for transport to receptors. Therefore, the postulated event is a break in the ventilation system between the fan and the single HEPA filter that allows cement fines to be introduced into the atmosphere.

Along with the postulated release, a source term is needed to determine radiation doses to the onsite and Hanford Site boundary receptors. As the drill bit descends into Tank 241-CX-72, it encounters increasing concentrations of radioactive material. This is illustrated in Subrahmanyam 1989. Subrahmanyam 1989 shows dose rate measurements taken inside the tank as a function of depth. This data also serves as a qualitative measure of the concentrations of the radioactive material as one goes deeper into the tank. Although the dose rate measurements may only represent radioactive material splashed onto the tank's wall during grout addition, this hazard classification analysis assumes that the contaminant's concentration is constant in the horizontal direction for any depth. The calculation in Appendix B uses the dose rate data and the best estimate of the curie concentrations in the tank from Section 2.6 to develop the inventory of ^{137}Cs and ^{239}Pu as a function of depth. A better estimate of the inventory with depth may be obtained by periodically collecting samples during the dry mining operation and performing analyses to determine the quantities and types of radioisotopes present in the cement fines.

Once the inventory of radioactive material is known, a release rate and a calculation of the receptor dose may be determined. It stands to reason that not all of the material released will reach the receptor. Most of the heavier cement particles will drop out of the air en route to the receptor. The determination of which particles will reach the receptor is made by assuming that only those in the respirable range will be transported any appreciable distance. It is generally accepted that the respirable range for particles is about 10 microns or less (Ayer et al. 1988). However, discussions with the Westinghouse Hanford Company Radiological Safety Assessment resulted in a respirable range of about 50 microns or less. This larger range is to account for particles that do not penetrate to the deep lung (10 microns or less) but are stopped within the upper areas of the respiratory system and potentially contribute to the receptor's dose. A sieve analysis performed on cement fines produced by the drill bit during testing reveal that about 6 mass percent are within the respirable range. This translates to only 6% of the material entering the atmosphere will make it downrange and give the receptor an inhalation dose. This approach is developed in the calculation in Appendix B.

Figure 4. Simple Drawing of the Waste Removal System.



The receptor locations used in WHC 1990a are used in this analysis. Although the receptor locations in WHC 1990a were determined with GENII version 1.436, the receptor locations, as well as the atmospheric dispersion factors, would be no different if determined by the current version of the GENII code. Knowing the release rate of material, the atmospheric dispersion factors, and assuming an 8-hour release time for the onsite receptor and a 24-hour release for the site boundary receptor, radiation dose values resulting from inhalation are determined. Radiation doses from two source terms are reported in Table 2 as a function of drill depth. From these values, Appendix B, Figures B-1 and B-2 are used to estimate drilling depths using the source term from Section 2.6

3.3 SUMMARY AND CONCLUSIONS

The facility hazard classification analysis is the technique of choice to determine the worst-case consequences of a postulated release. A hazard classification assumes no credit for any barriers unless doing so increases the consequences of the event. Using this approach, a hazard classification analysis is performed for the Tank 241-CX-72 dry mining project. Results are calculated with respect to drill depth as radiation doses to the hypothetical onsite and Hanford Site boundary receptors resulting from inhalation of the respirable, contaminated concrete fines. Based on these results, the project assumes a low hazard classification for drill depths between 0 and 4.7 m and a potential for a moderate hazard classification or high hazard classification for drill depths exceeding 4.7 m.

4.0 LIMITS AND PRUDENT ACTIONS

There are three administrative control OSLs applied to ensure the validity of the safety assessment and to minimize exposure and environmental impact to ALARA. They require minimizing the potential for operations above the limits for a facility having a low hazard classification, releasing contaminated dust, and disposing of retrieved wastes in accordance with regulatory requirements.

Operational Safety Limit 1

1.0 Title--Prevent operations that would potentially result in the system having a moderate or high hazard classification.

1.1 Applicability--This limit applies to the activities associated with the confinement equipment and structures used in removal and packaging of the material removed from Tank 241-CX-72 in Section 2.0 of this report.

1.2 Objective--Reduce the potential for onsite and offsite releases above the limits for operations having a low hazard classification.

Table 2. Dose Consequences of the Postulated Release.

Depth[a] (meters)	Inhalation dose (rem) (higher source values)		Inhalation dose (rem) (lower source values)	
	Onsite receptor (125 m WNW at Building 209-E)	Hanford boundary receptor (18.3 km SW)	Onsite receptor (125 m WNW at Building 209-E)	Hanford boundary receptor (18.3 km SW)
4.7	11.6[b]	0.013	0.89	0.001
5.0	19.3	0.021	1.48	0.002
5.3	19.9	0.021	1.52	0.002
5.6	26.9[c]	0.029	2.06	0.003
5.9	46.5	0.05	3.56	0.005
6.2	57.5	0.062	4.41	0.006
6.5	125.7	0.136	9.64[b]	0.012
6.8	237.1	0.256	18.18	0.023
7.1	513.9	0.555	39.4[c]	0.05
7.4	832	0.898	63.79	0.081

[a]Depth is measured from the top of Tank 241-CX-72, which is located approximately 4.3 m (14 ft) belowgrade.

[b]This value exceeds the 5-rem interface between a low and moderate hazard classification.

[c]This value exceeds the 25-rem interface between a moderate and high hazard classification.

1.3 Requirements

a. Drilling operations above the 7.6 m depth require routine (at least each shift) gamma monitoring on the powder sampler. A radiation limit of 1.0 rem/h at contact applies for the sample filter monitor. When the 1.0 rem/h limit is reached or exceeded, Phase 1 operations stop and the sample will be analyzed for specific isotopes to determine if the isotopic concentrations are sufficient to cause further waste removal operations to qualify as moderate or high hazard activity. Based upon the results of the isotopic analysis, Environmental Safety Assurance and Hanford Restoration Operations will determine if Phase 1 operations should cease or may continue based upon sample isotopic analysis.

b. Powder samples taken by the air sampler shall be analyzed daily for radioactivity (gross beta-gamma activity and gross alpha activity). The samples will be composited weekly (a maximum of 7 days) for analysis of TRU content. If the TRU content exceeds 100 nCi/g, Phase 1 operations stop and Environmental Safety Assurance and Hanford Restoration Operations management will determine if Phase 1 operations should cease permanently or should continue.

1.4 Surveillance--The DWP-C-026-00010 will require daily analysis of samples and continuous gamma monitoring on the sample filter for drilling operations at or below depths of 4.7 m. The DWP will confirm that sample data and radiation data taken and collected are assessed at least weekly and appropriate action taken by the person in charge.

1.5 Recovery--In the event that compliance with the 1.3 requirements of this OSL are observed by Environmental Safety Assurance, Hanford Restoration Operations, or Decommissioning Engineering to be inadequate, all operations at the work site will cease. Prompt action by Hanford Restoration Operations using approved operating procedures will be taken to determine the radiological status of the materials removed and to contain and confine the materials removed from Tank 241-CX-72. The deficiency will be reviewed with the Environmental Safety Assurance and Hanford Restoration Operations who will jointly determine recovery actions.

1.6 Audit Point--Documentation that demonstrates compliance with this OSL. The documentation will consist of the data sheets completed according to the DWP that is reviewed weekly and trended monthly by the cognizant decommissioning engineer.

1.7 Basis--The amount of radioactive/hazardous material is expected to be low in the grout down to 4.7 m. At drilling depths below 4.7 m, the potential for elevated concentrations of contaminates increases. Grout removal operations and confinement of the removed material must minimize the potential for the release of materials to the environment. Powder samples and radiation monitoring of the sample filter ensure that operations maintain a low hazard classification for the systems and the amount of contaminants released from this activity is ALARA.

The hazards assessment in Section 3.0 concluded that for drill depths down to 4.7 m, the dry mining operation has a low hazard classification. For mining operations down to 4.7 m, the radiation doses for an onsite receptor assuming the worst case accident are below the risk acceptance guidelines for a facility with a low hazard classification. For mining operations to continue below drill depths of 4.7 m, administrative controls (per DWP-C-026-00010) are required to analyze powder samples and monitor the radioactivity of the sample filter. The powder should be non-TRU material and the radiation level for the sample filter should be below 1.0 rem/h at contact. These controls measure the radioactivity of the material and ensure that the potential for releasing radioactivity above the limits for an onsite receptor is low.

Drilling needs to be controlled through the zone between 4.7 m and 7.6 m. As the transition area, this zone may experience rapid rather than gradual change in radioactive and/or hazardous material content. The purpose of the Phase 1 operation is to remove as much grout as possible without having the removed materials classified as other than radioactive low-level waste or waste requiring special handling, storage, or shipping.

Operational Safety Limit 2

2.0 Title--Minimize the potential for the release of contaminated dust.

2.1 Applicability--This limit applies to the activities associated with the confinement equipment and structures used in removal and packaging of the material removed from Tank 241-CX-72 in Section 2.0 of this report.

2.2 Objective--Reduce the potential for airborne release of radioactive/hazardous dusts and ensure that work is performed within the bounds of the safety assessment.

2.3 Requirements

a. The equipment, including instruments and controls, used for removal of material from Tank 241-CX-72 will be maintained to ensure that the potential for dust release is minimized. The filtering system for dust removal and the stack monitoring equipment shall be checked at least daily to verify containment and confinement of the material. Periodic checking (at least weekly) of the vacuum hoses for leaks will minimize material escape when the Ultra-Vac or exhauster operation stops.

b. During nonwork days the equipment will be checked at least daily to confirm containment and confinement of radionuclides or hazardous materials.

c. The packaged material will be maintained in a stabilized condition during collection and transportation.

d. The green house covering the drilling equipment shall be routinely inspected to ensure confinement integrity in accordance with original design configuration.

2.4 Surveillance--The DWP will specifically require that materials removed from Tank 241-CX-72, including dusts collected by the air filtration units, will be maintained in a stabilized condition that contains radioactive or hazardous materials. The DWP will require that material handling and other confinement related equipment or structures and samples taken and surveys/analysis performed are assessed at least weekly and appropriate action taken by the person in charge.

2.5 Recovery--In the event that compliance with the 2.3 requirements of this OSL are observed by Environmental Safety Assurance, Hanford Restoration Operations, or Decommissioning Engineering to be inadequate, all operations at the work site will cease. Prompt action by Hanford Restoration Operations using approved operating procedures will be taken to determine the radiological status of the materials removed and to contain and confine the materials from Tank 241-CX-72. The deficiency will be reviewed with Environmental Safety Assurance and Hanford Restoration Operations who will jointly determine recovery actions.

2.6 Audit Point--Documentation that demonstrates compliance with this OSL. The documentation will consist of the data sheets completed according to DWP-C-026-00010, which will be reviewed weekly and trended monthly by the cognizant decommissioning engineer.

2.7 Basis--Grout removal operations and confinement of the removed material must minimize the potential for the release of materials to the environment. Monitoring of the material handling equipment integrity, instruments and controls help to ensure the amount of contaminants released from this activity is ALARA.

Operational Safety Limit 3

3.0 Title--Minimize the potential for release to the environment from onsite storage of contaminated material from Tank 241-CX-72.

3.1 Applicability--This limit applies to the activities associated with the methods and equipment for removing, packaging, and storing contaminated/hazardous material from Tank 241-CX-72.

3.2 Objective--To minimize the potential for releasing contaminated/hazardous material to the environment.

3.3 Requirements--All waste containers shall be stored and shipped in accordance with the requirements of WHC-CM-2-14 (WHC 1991). A visual inspection of the containers shall verify that the container lid is positioned and secured, the container identification is properly affixed, and there is no evidence of penetrations or leakage. The container shall be loaded and maintained in accordance with the applicable Safety Analysis Report, Packaging (SARP).

3.4 Surveillance--The DWP will specifically require maintaining contaminated/hazardous materials in a condition that minimizes the potential for release to the environment. Documents will confirm that containment of the stored material is assessed at least weekly.

3.5 Recovery--In the event that compliance with 3.3 requirements of this OSL are observed by Environmental Safety Assurance, Hanford Restoration Operations, or Decommissioning Engineering to be inadequate, all operations at the work site will cease. Prompt action by Hanford Restoration Operations using approved operating procedures will be taken to determine the radiological status of the materials removed and to contain and confine the materials from Tank 241-CX-72. The deficiency will be reviewed with Environmental Safety Assurance and Hanford Restoration Operations who will jointly determine recovery actions.

3.6 Audit Point--Documentation that demonstrates compliance with this OSL. The documentation will consist of the data sheets completed according to DWP-C-026-00010, which will be reviewed weekly and trended monthly by the cognizant decommissioning engineer.

3.7 Basis--The release of contamination/hazardous materials to the environment must be minimized to reduce the potential impact to the environment, the facility workers, and people not involved in this work effort to ALARA.

Three prudent actions, safety practices performed in the spirit of ALARA, adopted by Decontamination and Decommissioning Engineering further ensure safe operation by maintaining control of contamination/hazardous materials, removing potential hazards, and meeting ALARA goals.

Function: Contaminated equipment.

Prudent Action 1: Even though radioactive contamination is expected to be minimal, monitor all equipment, filters, and packages removed from the work area for radioactivity to ensure they are free of contamination and can be released from the area.

Function: Disposal plan for stored packages.

Prudent Action 2: A disposal plan will be developed promptly and implemented to remove the contaminated material to a permanent waste repository. This will eliminate the potential hazard of the stored contaminated material.

Function: Operations.

Prudent Action 3: Conduct all operations in compliance with appropriate HWOP, JSA, and RWP requirements.

5.0 QUALITY ASSURANCE

Quality assurance shall be in accordance with the *Hanford Restoration Operation Quality Assurance Program Plan*, WHC-CM-6-7, (WHC 1989d).

6.0 REFERENCES

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APPENDIX A

TANK 241-CX-72 SOURCE TERM CALCULATIONS

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DON'T SAY IT --- Write it
17 Apr 1992

To: LE Johnson N1-75 From: RB Bendixsen N1-75

cc: DO Hess L6-57
JP Hinckley N1-24
DR Speer R2-77

Subject: Estimating Radionuclide Content of Tank 241-CX-72

Calculations of the radionuclide content of Tank 241-CX-72 using references 1 (Subrahmanyam 1989), 2 (Allen 1985) and 3 (Bendixsen 1990) provide higher values than previous estimates (Reference 4, Ludowise 1990). An error in the Reference 4 'Method B' calculations accounts for the lower Method 'B' radionuclide estimates. The author incorrectly assumed that a 24% waste volume reduction (WVR) for aging waste means that 76% of the waste is sludge. WVR means the fraction of supernate that can be removed through concentration methods and maintain the supernate within the range of liquid specifications. Table 1 shows a comparison of the values calculated in this report (note attached Appendix A showing the calculational methods) with Ludowise 1990 and the information in Subrahmanyam 1989. The estimated ^{137}Cs content of CX-72 ranges from 8,310 to 23,600 Ci and Pu content ranges from 20.9 to 200 g. The following paragraphs describe the assumptions used in this report.

The CX-72 tank history contained in Subrahmanyam 1989 can be summarized as follows: 1) Tank constructed in 1956 near Hot Semiworks in 200 East Area, 2) received 8723.9 L pilot scale PUREX type processed wastes from Hot Semiworks in 1957, received 61.2 lb (27,785 g) of U, received 48.49 g of Pu, a 1976 sludge measurement shows 74.5 in. of sludge with 1 in. of liquid, the tank has no outlet, and the tank received an unspecified amount of liquid fluoride wastes from "heel" recovery operations. The wastes in CX-72 probably self concentrated to the 1976 volume using radioactive decay heat.

Method 1 (Appendix A and Table 1) assumes the sludge in CX-72 to be similar to the PUREX sludge in the 241-A Tank Farm resulting from PUREX operations 1956 through 1964. Values of sludge accumulated in the A Tank Farm versus tons of fuel processed yield an average value of 65.2 l sludge/ton of processed fuel. This is a conservative estimate because the wastes collected in the A Tank Farm do not represent evaporating all of the supernate as is assumed for CX-72. The supernates for the A Tank Farm remained in place assisting in heat removal from the thermally hot sludge layer through the supernate layer to the ventilation system. Both the sluiced sludge layer and supernate were eventually pumped to B Plant for Sr and Cs separation and encapsulation operations. Therefore, Method 1 provides an upper bound for radionuclide concentrations in CX-72. This method estimates that CX-72 contains 14,800 Ci of ^{137}Cs with 19.3 g of Pu. The volume of waste discharged into CX-72 (7,050 l) appears to be consistent with the maximum 8,894 l volume for the tank.

Method 2 uses the 8,723.9 l data value and assumes that the Hot Semiworks PUREX pilot scale process is similar to the PUREX process described by R. L. Walser (Walser 1992). Usually, a pilot scale process will have more dilute wastes and higher Pu losses than the fully developed process. Therefore, we would expect the estimated 18,400 Ci ^{137}Cs to be higher and the 26.2 g Pu to be lower than the content of CX-72. This suggests that the Method 2 ^{137}Cs value is conservative but the Pu value might be too low.

Method 3 (Appendix A and Table 1) uses the 22% sludge value and assumes that the sludge in CX-72 is similar to the sludge produced by the PUREX process. Again, as with assumptions in Methods 1 and 2, slight differences probably exist between PUREX sludge and the sludge in CX-72 and the Hot Semiworks pilot scale process and the PUREX process. Evaporating the supernate and sludge in CX-72 to a 74.5 in. sludge layer with a 1 in. liquid layer on top of the sludge will produce a higher sludge layer when compared to the A Tank Farm sludge layers. Likewise, a pilot scale process will have higher concentrations of anions and cations in a larger volume waste stream which will cause a higher volume of sludge. Thus, the Method 3 ^{137}Cs value is also a conservative value and the Pu value is probably low. Table 1 shows estimated CX-72 values of 18,800 Ci ^{137}Cs and 23.2 g Pu.

Table 1
Radionuclide Content Estimates for Tank 241-CX-72

Estimates of Radionuclide Content						
Nuclide	Ref. 1	Ref. 4		Method		
		A	B	1	2	3
^{90}SrY (Ci)				14,800	18,400	14,600
$^{137}CsBa$ (Ci)		8,310	9,781	19,000	23,600	18,800
Pu (g)	48.49	200	200	21.1	26.2	20.9
U (g)	27,785			5,900	7,320	5,840
Volume (l)	8,723.9		3,902	7,050	8,723.9	6,970
Fuel (MTU)		2.9	3.4	21.4	29.1	23.2

References:

1. Subrahmanyam, V. B., 1989, Radiological Evaluation, Hot Semiworks Waste Tank 241-CX-72, WHC-SD-CP-TI-148, July 1989, Westinghouse Hanford Company, Richland, Washington

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3. Bendixsen, R. B., 1990, Sludge in Aging Waste Tanks and Maximum Observed Temperature, WHC-SD-WM-TA-019, April 1990, Westinghouse Hanford Company, Richland, Washington

4. Ludowise, J. D., 1990, An Estimate of the Radionuclide Content of Tank 241-CX-72, WHC-SD-DD-TI-051, April 1990, Westinghouse Hanford Company, Richland, Washington

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DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 4 of 5

(4) Building 241-CX-72 (5) Rev. _____ (6) Job No. _____

(7) Subject Radionuclide content of Tank CX-72

(8) Originator [illegible] Date 17 Apr 1992

(9) Checker _____ Date _____

(10)

The following calculations provide 3 estimates for radionuclide content of the sludge in Tank 241-CX-72. These estimates are based on the following information:

Measured sludge height in CX-72 74.5 in. (Ref. 1)

Volume added to CX-72 8,723.9 ℓ (Ref. 1)

From PUREX Flow sheet (Ref. 2)

Liquid waste volume from PUREX 300 ℓ/ton

sludge volume 22%

Radionuclide content

^{90}Sr Y	5 Ci/ℓ
^{137}Cs Ba	6 Ci/ℓ
U	.84 g/ℓ
Pu	.003 g/ℓ

All other radionuclides become inconsequential after 35y (1992-1957)

Sludge accumulation in 241-A Tank Farm (Ref. 3)

Tank	Tons fuel processed	in. of sludge
A-101	5791	26
A-102	7892	36
A-104	9323	60
A-105	11,176	65
A-106	8,940	83
TOTAL	43,112 tons	270 in.

$$\frac{270\ in.}{43{,}112\ tons} * \frac{2750\ gal}{in.} * \frac{3.785\ \ell}{gal} = 65.2\ \frac{\ell\ sludge}{ton\ fuel}$$

Sludge volume in CX-72 40 in. diameter cylinder

$$\frac{\pi}{4} * \left(\frac{40\ in.}{12}\right)^2 ft^2 * \frac{74.5}{12} ft * \frac{7.48\ gal}{ft^3} * \frac{3.785\ \ell}{gal} = 1534\ \ell\ Sludge$$

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 5 of 5

(4) Building 241-CX-72 (5) Rev. _____ (6) Job No. _____

(7) Subject Radionuclide Content of Tank 241-CX-72

(8) Originator [illegible] Date 17 Apr 1992

(9) Checker _____ Date _____

(10)

Calculate Radionuclide content of CX-72

Method 1 Assume sludge is similar to Aging waste sludge in 'A' Tank Farm

$\frac{1534\,l}{65.2\,l/TON} = 23.5$ Tons 23.5 TONS $* 300 \frac{l}{TON} = 7{,}050\,l$

$^{90}SrY = 5\,\frac{Ci}{l} * 300\,\frac{l}{TON} * 23.5$ TONS $* e^{\frac{-.693(35)}{28}} = 14{,}800$ Ci ^{90}SrY

$^{137}CsBa = 6\,Ci/l * 300\,\frac{l}{TON} * 23.5$ TONS $* e^{\frac{-.693(35)}{11041/365.25}} = 19{,}000$ Ci $^{137}CsBa$

DECAY FACTOR

Based on ^{90}Sr half life = 28 Years and ^{137}Cs half life = 11041 d

U $.84\,g/l * 300\,l/TON$ = 5,900 g U

Pu $.003\,g/l * 300\,l/TON$ = 21.1 g Pu

Method 2 Use data: CX-72 received 8723.9 l of waste

$\frac{8723.9\,l}{300\,l/TON} = 29.1$ TON $\frac{29.1}{23.5} = 1.24$ Fraction of Method 1

$14{,}800 * 1.24 = 18{,}400$ Ci ^{90}SrY

$19{,}000 * 1.24 = 23{,}600$ Ci $^{137}CsBa$

$21.1 * 1.24 = 26.2$ g Pu

$5{,}900 * 1.24 = 7{,}320$ g U

Method 3 Use 22% sludge value from PUREX flowsheet

$\frac{1534}{.22} = 6{,}970\,l$ added to CX-72 $\frac{6970\,l}{300\,l/TON} = 23.2$ Tons

$\frac{23.2}{23.5} = .99$ of Method 1

$14{,}800 * .99 = 14{,}600$ Ci ^{90}SrY

$19{,}000 * .99 = 18{,}800$ Ci $^{137}CsBa$

$21.1 * .99 = 20.9$ g Pu

$5{,}900 * .99 = 5{,}840$ g U

Maximum Values and sludge concentrations:

^{137}Cs (Method 2) $23{,}600$ Ci $* \frac{1}{1534\,l} = 15.4\,\frac{Ci}{l}$ ^{137}Cs

200g Pu (Subramanyam) $200g * 6.2E\text{-}02\,\frac{Ci}{g} * \frac{1}{1534\,l} * \frac{10^9\,nano\,Ci}{Ci} * \frac{1}{1400\,g/l} = 5770\,\frac{nCi}{g}$

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APPENDIX B

ONSITE AND OFFSITE DOSE CALCULATIONS

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APPENDIX B

The attached calculations performed by G. C. Henson provide both onsite receptor and site boundary receptor dose estimates for source values from R. B. Bendixsen (p. B-6 to B-17) and D. R. Speer (p. B-18 to B-29). These two sets of numbers yield dose estimates for various drilling depths when the respirable fraction is 6%. Figure B-1 shows a plot of these values for the onsite receptor located 100 m from the source. The values along the abscissa of Figure B-1 are ^{137}Cs values but the ordinate Rem value represents the combined dose from all of the radionuclides considered in the calculations (^{90}Sr, ^{90}Y, ^{137}Cs, ^{137}Ba, and ^{239}Pu).

Recent laboratory sieve analysis form the basis for the 6% respirable value. The laboratory respirable value is considerably above the literature respirable value of 0.1% (Ayer et al. 1988). Figure B-2 shows the onsite dose estimates at 100 m using the literature respirable value of 0.1%. The values in Figure B-2 were calculated by dividing the Figure 1 values by 60 (6% / 0.1% = 60).

These two figures give a range of dose estimates at the various drill depths. Comparing the estimated doses to established limits yields the following: For onsite receptors, a facility has a low hazard classification when the estimated dose is less than 5 rem. In Figure B-1, using the Appendix A maximum value of 15.4 Ci/l for ^{137}Cs, the 5 rem dose occurs at a drill depth of 4.7 m. In Figure B-2, the 5 rem dose occurs at 7.3 m of drill depth. Therefore, exceeding the 5 rem dose for a low hazard classification for Phase 1 drilling activities could occur somewhere between drilling depths of 4.7 and 7.3 m.

An inspection of the dose estimates for the offsite receptors and comparing these values to the low hazard classification limit of .5 rem shows that the drill depths exceed those for the onsite receptor. Therefore, the dose estimates for the onsite receptor are limiting, eliminating the need to calculate the maximum drill depths for the site boundary receptor.

Figure B-1. Dose to Onsite Receptor 8 Hour Exposure. (6% Respirable)

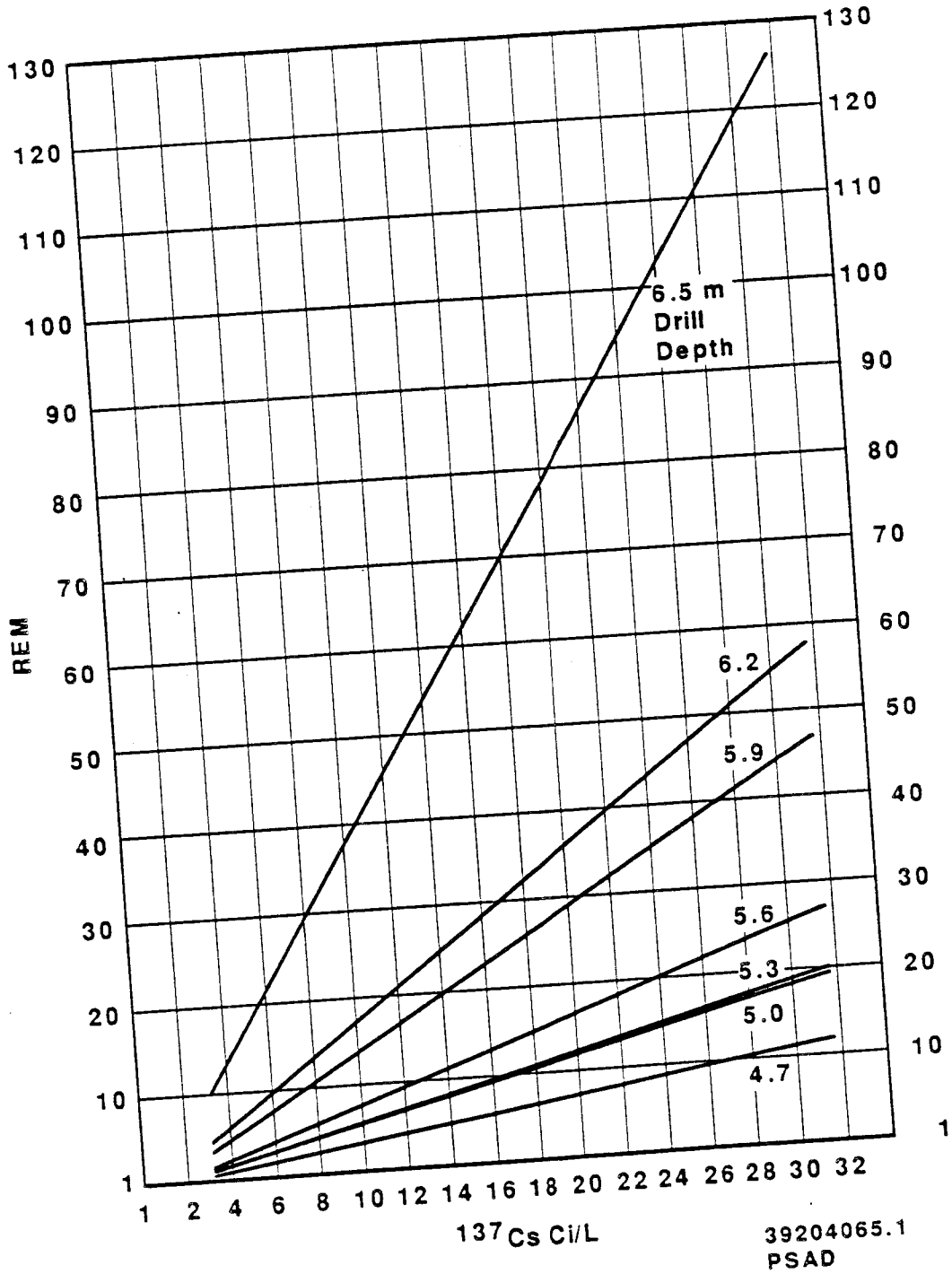
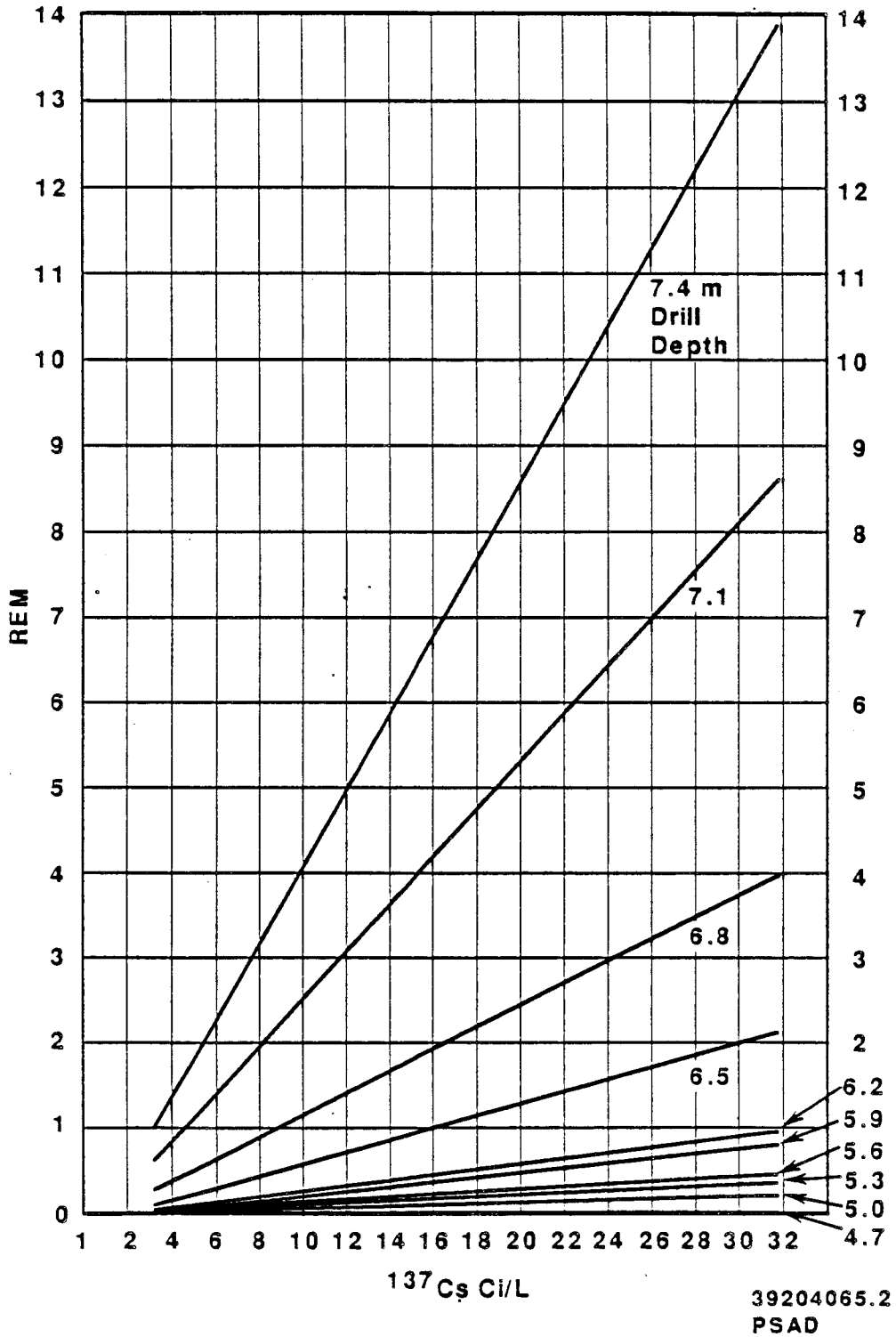


Figure B-2. Dose to Onsite Receptor 8 Hour Exposure. (0.1% Respirable)



USING REX BENDIXSEN'S INPUT

PURPOSE: This scoping calculation is to establish a relationship between receptor inhalation doses and drill depth into tank 241-CX-72. It is also necessary to establish at what depth during the drilling operation does the project assume a moderate and high hazard classification status.

GIVEN: It is postulated that during the drilling operation, the ventilation system is breached downstream of the fan (between the fan and the HEPA unit) allowing contaminated cement fines to be blown into the atmosphere. The Hazards Inventory (section 2.6) portion of this document estimates an upper bound to the radioactive inventory for tank 214-CX-72. This estimate is shown to be 5,770 nCi of plutonium per gram of grouted sludge and 66.5 Ci of cesium-137 per liter of grouted sludge. The rate of descent into the grout will be approximately 0.64 cm/hr. The diameter of 241-CX-72 is 101.6 cm. Dose rate data from Subrahmanyam 1989 shows the dose rate in the tank as a function of depth within regions 2 and 3 of 241-CX-72.

FIND:

a) Plot the dose rate data.
b) Determine the air concentration of the cement fines at the 100 m and site boundary receptors. Compare this to established limits.
c) Determine the air concentration of cesium-137 and Pu at the 100 m (328 ft) and site boundary receptors. Compare to the derived air concentration (DAC) for inhalation of cesium-137 and Pu.

SOLN:

BASIC EQUATION:

$$H = Ci * COQ$$

where H = concentration (Ci/m^3)
Ci = release rate (curies/sec)
COQ = atmospheric dispersion factor (sec/m^3)

ASSUMPTIONS:

1. This calculation relies upon the accuracy of the inventory specified in section 2.6 of this document. Changes to the quantity and type of radionuclide will have an affect on the results of this calculation. Therefore this calculation assumes the information provided is the best representation to date of what is actually in 241-CX-72. The calculation also assumes that the drilling operation is in a steady state condition. Therefore, the source term does not consider cement fines adhered to the ventilation ducting and components as being released during the postulated event.

2. Since the isotope of plutonium is not specified, it is assumed that the entire quantity is Pu-239. This isotope has the highest dose conversion factor of the plutonium isotopes typically encountered here at Hanford.

3. The inventory of radioactive material is representative of what is in the gouted-sludge layer of tank 241-CX-72. As the drill bit descends into the grout layers above, it will encounter increasing quantities of radionuclides. Dose rate measurements in the literature (Subrahmanyam 1989) provide a relative measure of the quantities of the radionuclides as a function of distance into the tank. This analysis requires knowledge of the curie concentrations as a function of distance into the tank. If one assumes that the dose rate data corresponds to say the cesium-137, then the dose rate data provides an understanding of the curie concentration of cesium as a function of distance. This calculation assumes that the given curie concentration for cesium corresponds to the dose rate data point at the 24.4 foot level in the tank (measured with respect to the top of 241-CX-72).

4. It has been assumed that the the vacuum unit/bag filter retains 85% of the concrete fines (Galgoul 1990). For the purposes of this scoping calculation, the vacuum unit/bag filter is assumed to allow 100% of the concrete fines to pass on to the fan and out to the atmosphere. It is required by WHC 1990 not to take into account any barriers when performing a hazard classification analysis.

5. The respirable range for the concrete fines is between about 44 microns and approximately 0. Discussions with WHC Radiological Safety Assessment reveal that the respirable range is about 50 microns to 0 micron. This range takes into account not only those particles which reach the deep portions of the lung (10 microns or less), but also those which are captured by the respiratory system's built in filtration and enter the digestive system. 44 microns was chosen as the cutoff for this calculation because it was the smallest range examined during Sieve analysis of the concrete fines and it closely follows the discussion with Radiological Safety Assessment.

VARIABLE DEFINITIONS:

$r := 20 \cdot in$ $r = 50.8 \cdot cm$	Radius of 241-CX-72 as taken from discussions with the project's field engineer.
$h := 1.0 \cdot cm$	Unit depth
$COQon := 2.3 \cdot 10^{-2} \cdot \frac{sec}{m^3}$	Atmospheric dispersion factor taken from Galgoul 1990 for the onsite receptor 125 m WNW of the drill site at the 209-E building.

$COQoff := 1.0 \cdot 10^{-5} \cdot \frac{sec}{m^3}$ — Atmospheric disperion factor from Galgoul 1990 for the Site Boundary receptor located 18.3 km SW of the drill site.

$DACpu := 2 \cdot 10^{-12} \cdot \frac{\mu Ci}{mL}$ — Derived air concentration for Pu-239 (WHC 1988)

$DACcs := 7 \cdot 10^{-8} \cdot \frac{\mu Ci}{mL}$ — Derived air concentration for Cs-137 (WHC 1988)

$DACsr := 2 \cdot 10^{-9} \cdot \frac{\mu Ci}{mL}$ — Derived air concentration for Sr-90 (WHC 1988)

$DACy90m := 5 \cdot 10^{-6} \cdot \frac{\mu Ci}{mL}$ — Derived air concentration for Y-90m (WHC 1988)

$DACy90 := 3 \cdot 10^{-7} \cdot \frac{\mu Ci}{mL}$ — Derived air concentration for Y-90 (WHC 1988)

$DCGcs := 4 \cdot 10^{-10} \cdot \frac{\mu Ci}{mL}$ — Derived concentration guide for Cs-137 (WHC 1991)

$DCGpu := 2 \cdot 10^{-14} \cdot \frac{\mu Ci}{mL}$ — Derived concentration guide for Pu-239 (WHC 1991)

$DCGsr := 9 \cdot 10^{-12} \cdot \frac{\mu Ci}{mL}$ — Derived concentration guide for Sr-90 (WHC 1991)

$DCGy90 := 1 \cdot 10^{-9} \cdot \frac{\mu Ci}{mL}$ — Derived concentration guide for Y-90 (WHC 1991)

$DCGy90m := 3 \cdot 10^{-8} \cdot \frac{\mu Ci}{mL}$ — Derived concentration guide for Y-90m (WHC 1991)

$CONCpu := 5770 \cdot \frac{nCi}{gm}$ — Upper bound concentration of Pu in 241-CX-72 waste

$CONCcs := 31.6 \cdot \frac{Ci}{liter}$ — Upper bound Cs-137 concentration in 241-CX-72 waste

$CONCsr := 27.2 \cdot \frac{Ci}{liter}$

$CONCy := CONCsr$

Upper bound Sr-90 and Y-90 concentration in 241-CX-72 waste.

$rho := 150 \cdot \frac{lb}{ft^3}$

Nominal density of concrete (This value is within the density ranges listed on page 6-230 of Baumeister 1967 -- Standard Handbook for Mechanical Engineers.)

$CONCpu := CONCpu \cdot rho$

$CONCpu = 0.0139 \cdot \frac{Ci}{liter}$

Make the units consistent for the Pu component.

a) Dose rate data from Subrahmanyam 1989

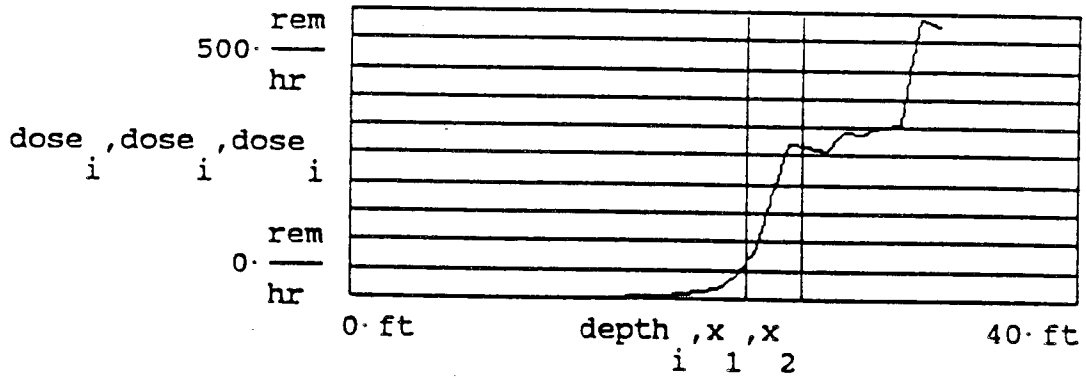
$$depth := \begin{bmatrix} 15.4 \\ 16.4 \\ 17.4 \\ 18.4 \\ 19.4 \\ 20.4 \\ 21.4 \\ 22.4 \\ 23.4 \\ 24.4 \\ 25.4 \\ 26.4 \\ 27.4 \\ 28.4 \\ 29.4 \\ 30.4 \\ 31.4 \\ 32.4 \end{bmatrix} \cdot ft \qquad dose := \begin{bmatrix} 3.8 \\ 6.3 \\ 6.5 \\ 8.8 \\ 15.2 \\ 18.8 \\ 41.1 \\ 77.5 \\ 168.0 \\ 272.0 \\ 265.0 \\ 258.0 \\ 293.0 \\ 288.0 \\ 301.0 \\ 304.0 \\ 491.0 \\ 476.0 \end{bmatrix} \cdot \frac{rem}{hr}$$

The variable depth is measured with respect to the top of 241-CX-72.

$i := 0 .. 17 \qquad x_1 := 22 \cdot ft \qquad x_2 := 25 \cdot ft$

Dose rate measurements versus depth in 241-CX-72

(where 0 ft corresponds to the top of 241-CX-72)



The depth values of 22 ft and 25 ft correspond to the depths to which the project would like to dry mine and the interface of the grouted sludge layer respectively.

b) Determine air concentration of cement fines at receptors:

The rate of descent for the dry mining operation is expressed by the variable Drillrat.

$$Drillrat := 0.25 \cdot \frac{in}{hr}$$

Not all of the particles released from the postulated incident will contribute to the inhalation dose of the hypothetical receptors. It is generally accepted that particles of aerodynamic equivalent diameters equal to or less than 10 microns are within the respirable range (Ayer et al. 1988). Considering assumption 5 and the fact that Sieve analyses on the concrete fines show that approximately 4 to 6 percent (by mass) of the fines were able to pass through a number 325 Sieve, it is assumed the respirable portion is that which can pass through the number 325 Sieve. Page 7-83 of Baumeister 1967 shows Taylor screen sizes for sieves. A number 325 sieve corrsponds to 44 microns. Therefore 4 to 6 percent (by mass) of the concrete fines are within the 44 to '0' micron range.

The respirable range relates to 6 percent, by mass, of the total quantity released to the atmosphere.

The quantity of concrete mined in one second and which is 44 microns or less in diameter is determined as follows:

$$Quant := \pi \cdot r^2 \cdot Drillrat$$

$$Quant = 0.0014 \cdot \frac{liter}{sec}$$

$$MASS := Quant \cdot rho$$

$$MASS = 3.4 \cdot \frac{gm}{sec}$$ Total mass released.

But the respirable mass released is 6 percent. Therefore,

$$MASSRESP := 0.06 \cdot MASS$$

$$MASSRESP = 0.2 \cdot \frac{gm}{sec}$$

$$VOLRESP := \frac{MASSRESP}{rho}$$

$$VOLRESP = 8.6 \cdot 10^{-5} \cdot \frac{liter}{sec}$$

The volume of concrete fines released within the defined respirable range.

The air concentration of concrete fines at the receptors is determined for the 125 meter receptor and the site boundary receptor.

For the 125 meter receptor,

$$Hon := MASSRESP \cdot COQon$$

$$Hon = 4.7 \cdot \frac{mg}{m^3}$$

NIOSH 1990 TLV-TWA for portland cement is 5 mg/m^3 for the respirible portion. 'No evidence could be found for the existence of an IDLH'.

For the site boundary receptor,

$$Hoff := MASSRESP \cdot COQoff$$

$$Hoff = 0.002 \cdot \frac{mg}{m^3}$$

c) Determine radionuclide air concentrations at the receptors.

The concentration of cesium-137 in the grouted sludge layer is given as

$$CONCcs = 31.6 \cdot \frac{Ci}{liter} .$$

Using assumption 3, the above value relates to the dose rate measurement made at the 24.4 foot level in the tank. Assuming a one to one relationship between the curie quantity and the dose rate data, a curie quantity is determined for the remaining dose rate data points between the 15.4 and 24.4 foot levels in the tank.

Taking just the first 10 values in the variable dose, the following shows the cesium-137 curie quantities at the various depths above 24.4 feet.

For $i := 0 .. 9$

$$CONCcs_i := dose_i \cdot \left[\frac{CONCcs}{dose_9}\right]$$

$$CONCcs = \begin{bmatrix} 0.4 \\ 0.7 \\ 0.8 \\ 1 \\ 1.8 \\ 2.2 \\ 4.8 \\ 9 \\ 19.5 \\ 31.6 \end{bmatrix} \cdot \frac{Ci}{liter}$$

This variable represents the curies of cesium-137 per liter of concrete at one foot intervals starting at 15.4 feet from the top of the tank to 24.4 feet from the top of the tank.

The same approach may be used to determine the Pu-239, Sr-90 and Y-90 concentrations as a function of depth. However, if we assume that for every X curies of Cs-137 there will be Y curies of Pu-239, we can use the Cs-137 concentration (CONCcs) to determine the Pu-239 concentration with depth as so on for the others. This is shown as follows:

$$CONCpu239_i := CONCcs_i \cdot \left[\frac{CONCpu}{CONCcs_9}\right]$$ For the plutonium

$$CONCsr90_i := CONCcs_i \cdot \left[\frac{CONCsr}{CONCcs_9}\right]$$ For the strontium

$$CONCy90_i := CONCsr90_i$$ For the yttrium

$$CONCy90m_i := CONCy90_i$$

The release rate of the radionulides is related to the release rate of the 'respirable' portion of the cement determined above.

$RATEcs := CONCcs \cdot VOLRESP$ $\qquad$ $RATEsr := CONCsr90 \cdot VOLRESP$

$RATEpu := CONCpu239 \cdot VOLRESP$ $\qquad$ $RATEy := RATEsr$

$RATEy90m := RATEy$

To determine the air concentrations for the 'respirable' portion of cesium to the receptors, the basic equation, defined earlier, is used.

For the onsite receptor (125 m):

$Honcs := RATEcs \cdot COQon$

$$Honcs = \begin{bmatrix} 8.7 \cdot 10^{-7} \\ 1.4 \cdot 10^{-6} \\ 1.5 \cdot 10^{-6} \\ 2 \cdot 10^{-6} \\ 3.5 \cdot 10^{-6} \\ 4.3 \cdot 10^{-6} \\ 9.4 \cdot 10^{-6} \\ 1.8 \cdot 10^{-5} \\ 3.9 \cdot 10^{-5} \\ 6.2 \cdot 10^{-5} \end{bmatrix} \cdot \frac{\mu Ci}{mL}$$

This variable is then compared to the the derived air concentration guide for cesium-137

The DAC for cesium-137 is

$$DACcs = 7 \cdot 10^{-8} \cdot \frac{\mu Ci}{mL}$$

$$Honpu := RATEpu \cdot COQon$$

$$Honpu = \begin{bmatrix} 3.8 \cdot 10^{-10} \\ 6.3 \cdot 10^{-10} \\ 6.5 \cdot 10^{-10} \\ 8.9 \cdot 10^{-10} \\ 1.5 \cdot 10^{-9} \\ 1.9 \cdot 10^{-9} \\ 4.1 \cdot 10^{-9} \\ 7.8 \cdot 10^{-9} \\ 1.7 \cdot 10^{-8} \\ 2.7 \cdot 10^{-8} \end{bmatrix} \cdot \frac{\mu Ci}{mL}$$

This variable is then compared to the the derived air concentration guide for plutonium-239

The DAC for plutonium-239 is

$$DACpu = 2 \cdot 10^{-12} \cdot \frac{\mu Ci}{mL}$$

And so on for the others...

$$Honsr := RATEsr \cdot COQon \qquad Hony := Honsr \qquad Hony90m := Hony$$

Since each radionuclide contributes to the hypothetical 125 m receptor's inhalation dose, we must compare the sum of fractions for each radionuclide.

$$FRACon := \left[\frac{Honcs}{DACcs}\right] + \left[\frac{Honpu}{DACpu}\right] + \left[\frac{Honsr}{DACsr}\right] + \left[\frac{Hony}{DACy90}\right] + \left[\frac{Hony90m}{DACy90m}\right]$$

$$FRACon = \begin{bmatrix} 581.2 \\ 963.5 \\ 994.1 \\ 1.3 \cdot 10^{3} \\ 2.3 \cdot 10^{3} \\ 2.9 \cdot 10^{3} \\ 6.3 \cdot 10^{3} \\ 1.2 \cdot 10^{4} \\ 2.6 \cdot 10^{4} \\ 4.2 \cdot 10^{4} \end{bmatrix}$$

This variable is unitless.

How does FRACon relate to a dose to the 125 meter receptor?

The DAC is the concentration of the radionuclide in air that if inhaled by an individual for 2000 hours (1 work year) would result in that individual receiving their annual limit on intake for the radionuclide.
The dose rate (on a per hour basis) then for the 100 meter individual is expressed as

$$DOSERATE100 := FRACon \cdot \left[\frac{5 \cdot rem}{2000 \cdot hr}\right]$$

$$DOSERATE100 = \begin{bmatrix} 1.5 \\ 2.4 \\ 2.5 \\ 3.4 \\ 5.8 \\ 7.2 \\ 15.7 \\ 29.6 \\ 64.2 \\ 104 \end{bmatrix} \cdot \frac{rem}{hr}$$

DOSERATE100 shows that as the drill bit descends into the grout, the potential dose to the receptor increases. It is assumed here that the value is constant in any given 1 foot slice of grout. Since the drilling rate is 0.25 in/hr, in 8 hours (a typical onsite release time for hazard classification analyses) the drill bit is still within the ith 1 foot slice.

The postulated release could take place at any 1-foot slice. So if we assume an 8 hour release time, the dose to the 100 meter receptor is given by:

$$DOSE100 := DOSERATE100 \cdot (8 \cdot hr)$$

$$DOSE100 = \begin{bmatrix} 11.6 \\ 19.3 \\ 19.9 \\ 26.9 \\ 46.5 \\ 57.5 \\ 125.7 \\ 237.1 \\ 513.9 \\ 832 \end{bmatrix} \cdot rem$$

At the onsite location

For the site boundary location, the same approach is used as for the onsite location. Note that the DAC is not appropriate to be used for nonoccupational workers. The DCG is the correct limit to use. If a member of the public is exposed to 1 DCG concentrations for 1 year, he will receive a dose of 0.1 rem.

$$Hoffcs := RATEcs \cdot COQoff \qquad Hoffsr := RATEsr \cdot COQoff$$

$$Hoffpu := RATEpu \cdot COQoff \qquad Hoffy := Hoffsr$$

$$Hoffy90m := Hoffy$$

$$FRACoff := \left[\frac{Hoffcs}{DCGcs}\right] + \left[\frac{Hoffpu}{DCGpu}\right] + \left[\frac{Hoffsr}{DCGsr}\right] + \left[\frac{Hoffy}{DCGy90}\right] + \left[\frac{Hoffy90m}{DCGy90m}\right]$$

$$DOSERATEsb := FRACoff \cdot \left[\frac{0.1 \cdot rem}{8766 \cdot hr}\right]$$

The DCG values correspond to a dose of 0.1 rem over a 8766 hour year.

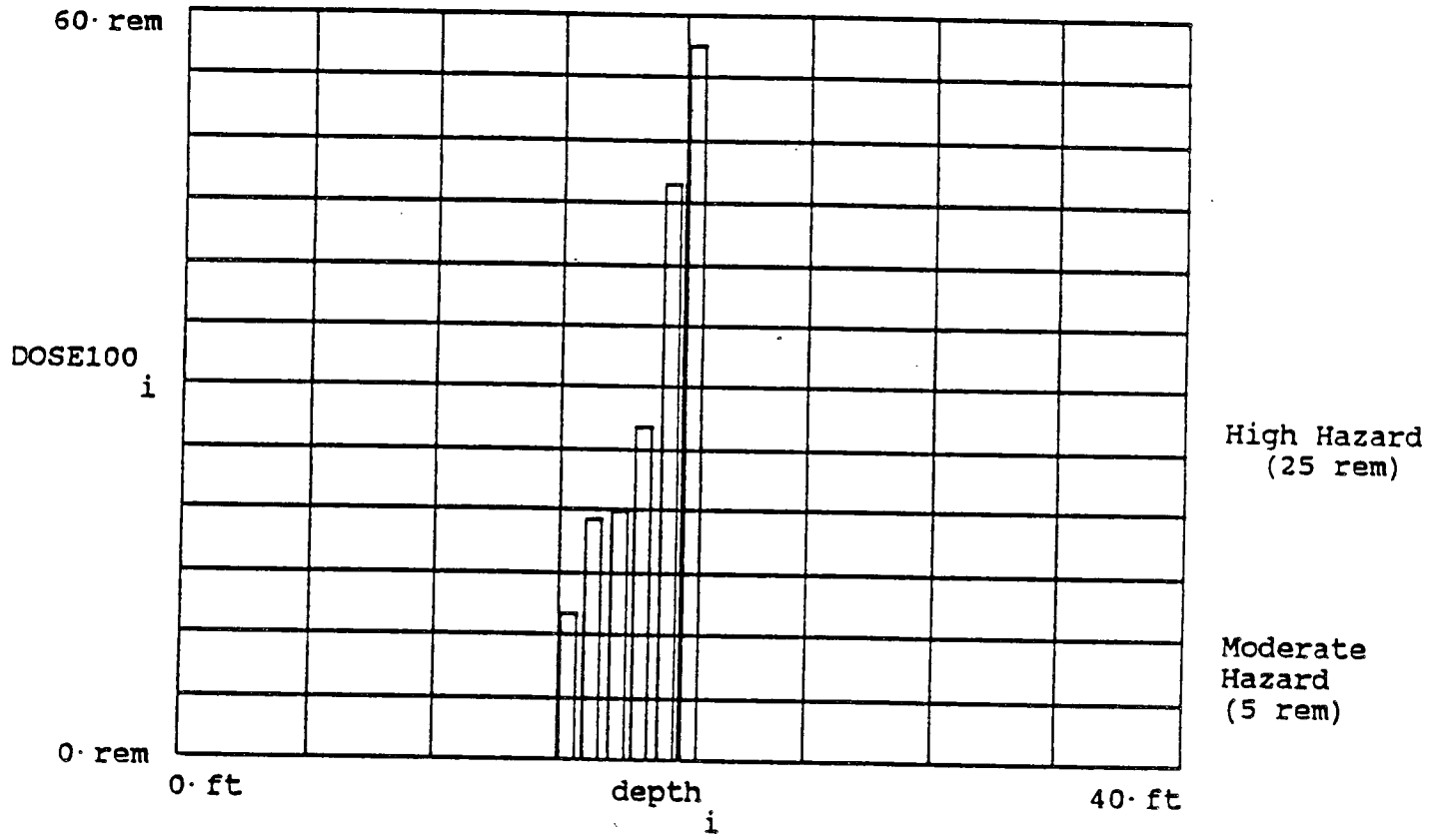
$$DOSEsb := DOSERATEsb \cdot (24 \cdot hr)$$

Typical release time for the site boundary dose determination is 24 hours for the hazard classification analysis.

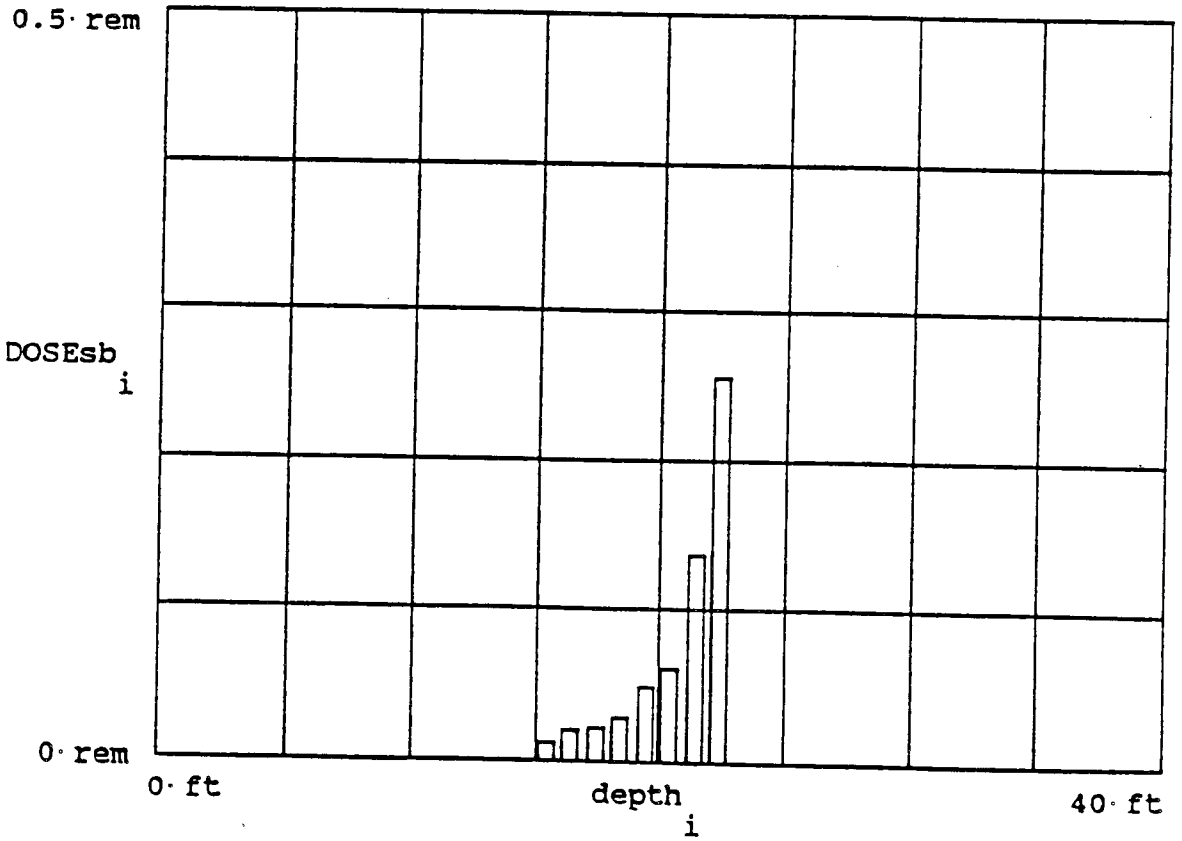
$$DOSEsb = \begin{bmatrix} 0.013 \\ 0.021 \\ 0.021 \\ 0.029 \\ 0.05 \\ 0.062 \\ 0.136 \\ 0.256 \\ 0.555 \\ 0.898 \end{bmatrix} \cdot rem$$

At the site boundary location

Dose to the Onsite Receptor Due to the Postulated Event



Dose to Site Boundary Receptor Due to the Postulated Event



USING DWAYNE SPEER'S INPUT

PURPOSE: This scoping calculation is to establish a relationship between receptor inhalation doses and drill depth into tank 241-CX-72. It is also necessary to establish at what depth during the drilling operation does the project assume a moderate and high hazard classification status.

GIVEN: It is postulated that during the drilling operation, the ventilation system is breached downstream of the fan (between the fan and the HEPA unit) allowing contaminated cement fines to be blown into the atmosphere. The Hazards Inventory (section 2.6) portion of this document estimates an upper bound to the radioactive inventory for tank 214-CX-72. This estimate is shown to be 5,770 nCi of plutonium per gram of grouted sludge and 66.5 Ci of cesium-137 per liter of grouted sludge. The rate of descent into the grout will be approximately 0.64 cm/hr. The diameter of 241-CX-72 is 101.6 cm. Dose rate data from Subrahmanyam 1989 shows the dose rate in the tank as a function of depth within regions 2 and 3 of 241-CX-72.

FIND:

a) Plot the dose rate data.
b) Determine the air concentration of the cement fines at the 100 m and site boundary receptors. Compare this to established limits.
c) Determine the air concentration of cesium-137 and Pu at the 100 m (328 ft) and site boundary receptors. Compare to the derived air concentration (DAC) for inhalation of cesium-137 and Pu.

SOLN:

BASIC EQUATION:

$$H = Ci * COQ$$

where H = concentration (Ci/m^3)
Ci = release rate (curies/sec)
COQ = atmospheric dispersion factor (sec/m^3)

ASSUMPTIONS:

1. This calculation relies upon the accuracy of the inventory specified in section 2.6 of this document. Changes to the quantity and type of radionuclide will have an affect on the results of this calculation. Therefore this calculation assumes the information provided is the best representation to date of what is actually in 241-CX-72. The calculation also assumes that the drilling operation is in a steady state condition. Therefore, the source term does not consider cement fines adhered to the ventilation ducting and components as being released during the postulated event.

2. Since the isotope of plutonium is not specified, it is assumed that the entire quantity is Pu-239. This isotope has the highest dose conversion factor of the plutonium isotopes typically encountered here at Hanford.

3. The inventory of radioactive material is representative of what is in the gouted-sludge layer of tank 241-CX-72. As the drill bit descends into the grout layers above, it will encounter increasing quantities of radionuclides. Dose rate measurements in the literature (Subrahmanyam 1989) provide a relative measure of the quantities of the radionuclides as a function of distance into the tank. This analysis requires knowledge of the curie concentrations as a function of distance into the tank. If one assumes that the dose rate data corresponds to say the cesium-137, then the dose rate data provides an understanding of the curie concentration of cesium as a function of distance. This calculation assumes that the given curie concentration for cesium corresponds to the dose rate data point at the 24.4 foot level in the tank (measured with respect to the top of 241-CX-72).

4. It has been assumed that the the vacuum unit/bag filter retains 85% of the concrete fines (Galgoul 1990). For the purposes of this scoping calculation, the vacuum unit/bag filter is assumed to allow 100% of the concrete fines to pass on to the fan and out to the atmosphere. It is required by WHC 1990 not to take into account any barriers when performing a hazard classification analysis.

5. The respirable range for the concrete fines is between about 44 microns and approximately 0. Discussions with WHC Radiological Safety Assessment reveal that the respirable range is about 50 microns to 0 micron. This range takes into account not only those particles which reach the deep portions of the lung (10 microns or less), but also those which are captured by the respiratory system's built in filtration and enter the digestive system. 44 microns was chosen as the cutoff for this calculation because it was the smallest range examined during Sieve analysis of the concrete fines and it closely follows the discussion with Radiological Safety Assessment.

VARIABLE DEFINITIONS:

$r := 20 \cdot in$ $r = 50.8 \cdot cm$	Radius of 241-CX-72 as taken from discussions with the project's field engineer.
$h := 1.0 \cdot cm$	Unit depth
$COQon := 2.3 \cdot 10^{-2} \cdot \frac{sec}{m^3}$	Atmospheric dispersion factor taken from Galgoul 1990 for the onsite receptor 125 m WNW of the drill site at the 209-E building.

$COQoff := 1.0 \cdot 10^{-5} \cdot \frac{sec}{m^3}$

Atmospheric disperion factor from Galgoul 1990 for the Site Boundary receptor located 18.3 km SW of the drill site.

$DACpu := 2 \cdot 10^{-12} \cdot \frac{\mu Ci}{mL}$

Derived air concentration for Pu-239 (WHC 1988)

$DACcs := 7 \cdot 10^{-8} \cdot \frac{\mu Ci}{mL}$

Derived air concentration for Cs-137 (WHC 1988)

$DACsr := 2 \cdot 10^{-9} \cdot \frac{\mu Ci}{mL}$

Derived air concentration for Sr-90 (WHC 1988)

$DACy90m := 5 \cdot 10^{-6} \cdot \frac{\mu Ci}{mL}$

Derived air concentration for Y-90m (WHC 1988)

$DACy90 := 3 \cdot 10^{-7} \cdot \frac{\mu Ci}{mL}$

Derived air concentration for Y-90 (WHC 1988)

$DCGcs := 4 \cdot 10^{-10} \cdot \frac{\mu Ci}{mL}$

Derived concentration guide for Cs-137 (WHC 1991)

$DCGpu := 2 \cdot 10^{-14} \cdot \frac{\mu Ci}{mL}$

Derived concentration guide for Pu-239 (WHC 1991)

$DCGsr := 9 \cdot 10^{-12} \cdot \frac{\mu Ci}{mL}$

Derived concentration guide for Sr-90 (WHC 1991)

$DCGy90 := 1 \cdot 10^{-9} \cdot \frac{\mu Ci}{mL}$

Derived concentration guide for Y-90 (WHC 1991)

$DCGy90m := 3 \cdot 10^{-8} \cdot \frac{\mu Ci}{mL}$

Derived concentration guide for Y-90m (WHC 1991)

$CONCpu := 78.5 \cdot \frac{nCi}{gm}$

Upper bound concentration of Pu in 241-CX-72 waste

$CONCcs := 3.26 \cdot \frac{Ci}{liter}$

Upper bound Cs-137 concentration in 241-CX-72 waste

$CONCsr := 2.93 \cdot \frac{Ci}{liter}$

$CONCy := CONCsr$

Upper bound Sr-90 and Y-90 concentration in 241-CX-72 waste.

$rho := 150 \cdot \frac{lb}{ft^3}$

Nominal density of concrete (This value is within the density ranges listed on page 6-230 of Baumeister 1967 -- Standard Handbook for Mechanical Engineers.)

$CONCpu := CONCpu \cdot rho$

$CONCpu = 1.8861 \cdot 10^{-4} \cdot \frac{Ci}{liter}$

Make the units consistent for the Pu component.

a) Dose rate data from Subrahmanyam 1989

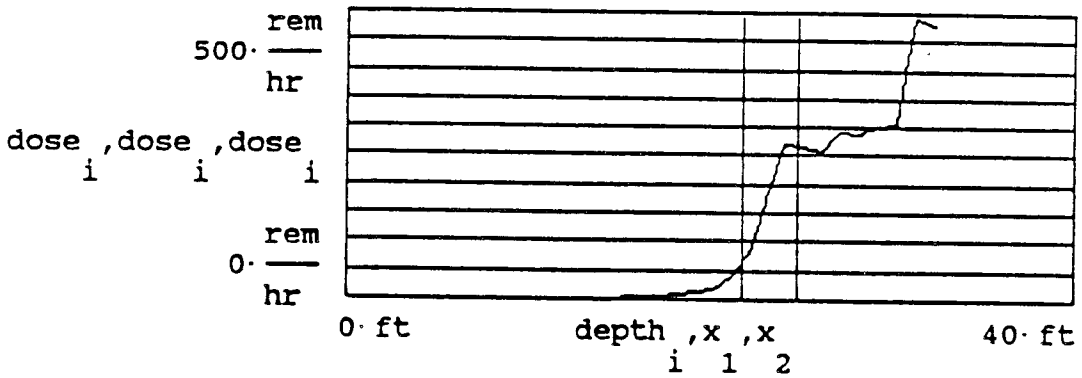
$$depth := \begin{bmatrix} 15.4 \\ 16.4 \\ 17.4 \\ 18.4 \\ 19.4 \\ 20.4 \\ 21.4 \\ 22.4 \\ 23.4 \\ 24.4 \\ 25.4 \\ 26.4 \\ 27.4 \\ 28.4 \\ 29.4 \\ 30.4 \\ 31.4 \\ 32.4 \end{bmatrix} \cdot ft \qquad dose := \begin{bmatrix} 3.8 \\ 6.3 \\ 6.5 \\ 8.8 \\ 15.2 \\ 18.8 \\ 41.1 \\ 77.5 \\ 168.0 \\ 272.0 \\ 265.0 \\ 258.0 \\ 293.0 \\ 288.0 \\ 301.0 \\ 304.0 \\ 491.0 \\ 476.0 \end{bmatrix} \cdot \frac{rem}{hr}$$

The variable depth is measured with respect to the top of 241-CX-72.

$i := 0 .. 17$ $\qquad x_1 := 22 \cdot ft$ $\qquad x_2 := 25 \cdot ft$

Dose rate measurements versus depth in 241-CX-72

(where 0 ft corresponds to the top of 241-CX-72)



The depth values of 22 ft and 25 ft correspond to the depths to which the project would like to dry mine and the interface of the grouted sludge layer respectively.

b) Determine air concentration of cement fines at receptors:

The rate of descent for the dry mining operation is expressed by the variable Drillrat.

$$Drillrat := 0.25 \cdot \frac{in}{hr}$$

Not all of the particles released from the postulated incident will contribute to the inhalation dose of the hypothetical receptors. It is generally accepted that particles of aerodynamic equivalent diameters equal to or less than 10 microns are within the respirable range (Ayer et al. 1988). Considering assumption 5 and the fact that Sieve analyses on the concrete fines show that approximately 4 to 6 percent (by mass) of the fines were able to pass through a number 325 Sieve, it is assumed the respirable portion is that which can pass through the number 325 Sieve. Page 7-83 of Baumeister 1967 shows Taylor screen sizes for sieves. A number 325 sieve corrsponds to 44 microns. Therefore 4 to 6 percent (by mass) of the concrete fines are within the 44 to '0' micron range.

The respirable range relates to 6 percent, by mass, of the total quantity released to the atmosphere.

The quantity of concrete mined in one second and which is 44 microns or less in diameter is determined as follows:

$$Quant := \pi \cdot r^2 \cdot Drillrat$$

$$Quant = 0.0014 \cdot \frac{liter}{sec}$$

$$MASS := Quant \cdot rho$$

$$MASS = 3.4 \cdot \frac{gm}{sec}$$ Total mass released.

But the respirable mass released is 6 percent. Therefore,

$$MASSRESP := 0.06 \cdot MASS$$

$$MASSRESP = 0.2 \cdot \frac{gm}{sec}$$

$$VOLRESP := \frac{MASSRESP}{rho}$$

$$VOLRESP = 8.6 \cdot 10^{-5} \cdot \frac{liter}{sec}$$

The volume of concrete fines released within the defined respirable range.

The air concentration of concrete fines at the receptors is determined for the 125 meter receptor and the site boundary receptor.

For the 125 meter receptor,

$$Hon := MASSRESP \cdot COQon$$

$$Hon = 4.7 \cdot \frac{mg}{m^3}$$

NIOSH 1990 TLV-TWA for portland cement is 5 mg/m^3 for the respirible portion. 'No evidence could be found for the existence of an IDLH'.

For the site boundary receptor,

$$Hoff := MASSRESP \cdot COQoff$$

$$Hoff = 0.002 \cdot \frac{mg}{m^3}$$

c) Determine radionuclide air concentrations at the receptors.

The concentration of cesium-137 in the grouted sludge layer is given as

$$CONCcs = 3.3 \cdot \frac{Ci}{liter}$$.

Using assumption 3, the above value relates to the dose rate measurement made at the 24.4 foot level in the tank. Assuming a one to one relationship between the curie quantity and the dose rate data, a curie quantity is determined for the remaining dose rate data points between the 15.4 and 24.4 foot levels in the tank.

Taking just the first 10 values in the variable dose, the following shows the cesium-137 curie quantities at the various depths above 24.4 feet.

For $i := 0 .. 9$

$$CONCcs_i := dose_i \cdot \left[\frac{CONCcs}{dose_9}\right]$$

$$CONCcs = \begin{bmatrix} 0.05 \\ 0.08 \\ 0.08 \\ 0.11 \\ 0.18 \\ 0.23 \\ 0.49 \\ 0.93 \\ 2.01 \\ 3.26 \end{bmatrix} \cdot \frac{Ci}{liter}$$

This variable represents the curies of cesium-137 per liter of concrete at one foot intervals starting at 15.4 feet from the top of the tank to 24.4 feet from the top of the tank.

The same approach may be used to determine the Pu-239, Sr-90 and Y-90 concentrations as a function of depth. However, if we assume that for every X curies of Cs-137 there will be Y curies of Pu-239, we can use the Cs-137 concentration (CONCcs) to determine the Pu-239 concentration with depth as so on for the others. This is shown as follows:

$$CONCpu239_i := CONCcs_i \cdot \left[\frac{CONCpu}{CONCcs_9}\right]$$ For the plutonium

$$CONCsr90_i := CONCcs_i \cdot \left[\frac{CONCsr}{CONCcs_9}\right]$$ For the strontium

$$CONCy90_i := CONCsr90_i$$ For the yttrium

$$CONCy90m_i := CONCy90_i$$

The release rate of the radionulides is related to the release rate of the 'respirable' portion of the cement determined above.

$RATEcs := CONCcs \cdot VOLRESP$ $\qquad$ $RATEsr := CONCsr90 \cdot VOLRESP$

$RATEpu := CONCpu239 \cdot VOLRESP$ $\qquad$ $RATEy := RATEsr$

$RATEy90m := RATEy$

To determine the air concentrations for the 'respirable' portion of cesium to the receptors, the basic equation, defined earlier, is used.

For the onsite receptor (125 m):

$Honcs := RATEcs \cdot COQon$

$$Honcs = \begin{bmatrix} 9 \cdot 10^{-8} \\ 1.5 \cdot 10^{-7} \\ 1.5 \cdot 10^{-7} \\ 2.1 \cdot 10^{-7} \\ 3.6 \cdot 10^{-7} \\ 4.4 \cdot 10^{-7} \\ 9.7 \cdot 10^{-7} \\ 1.8 \cdot 10^{-6} \\ 4 \cdot 10^{-6} \\ 6.4 \cdot 10^{-6} \end{bmatrix} \cdot \frac{\mu Ci}{mL}$$

This variable is then compared to the the derived air concentration guide for cesium-137

The DAC for cesium-137 is

$$DACcs = 7 \cdot 10^{-8} \cdot \frac{\mu Ci}{mL}$$

$$Honpu := RATEpu \cdot COQon$$

$$Honpu = \begin{bmatrix} 5.2 \cdot 10^{-12} \\ 8.6 \cdot 10^{-12} \\ 8.9 \cdot 10^{-12} \\ 1.2 \cdot 10^{-11} \\ 2.1 \cdot 10^{-11} \\ 2.6 \cdot 10^{-11} \\ 5.6 \cdot 10^{-11} \\ 1.1 \cdot 10^{-10} \\ 2.3 \cdot 10^{-10} \\ 3.7 \cdot 10^{-10} \end{bmatrix} \cdot \frac{\mu Ci}{mL}$$

This variable is then compared to the the derived air concentration guide for plutonium-239

The DAC for plutonium-239 is

$$DACpu = 2 \cdot 10^{-12} \cdot \frac{\mu Ci}{mL}$$

And so on for the others...

$$Honsr := RATEsr \cdot COQon \qquad Hony := Honsr \qquad Hony90m := Hony$$

Since each radionuclide contributes to the hypothetical 125 m receptor's inhalation dose, we must compare the sum of fractions for each radionuclide.

$$FRACon := \left[\frac{Honcs}{DACcs}\right] + \left[\frac{Honpu}{DACpu}\right] + \left[\frac{Honsr}{DACsr}\right] + \left[\frac{Hony}{DACy90}\right] + \left[\frac{Hony90m}{DACy90m}\right]$$

$$FRACon = \begin{bmatrix} 44.6 \\ 73.9 \\ 76.2 \\ 103.2 \\ 178.2 \\ 220.5 \\ 482 \\ 908.8 \\ 2 \cdot 10^{3} \\ 3.2 \cdot 10^{3} \end{bmatrix}$$

This variable is unitless.

How does FRACon relate to a dose to the 125 meter receptor?

The DAC is the concentration of the radionuclide in air that if inhaled by an individual for 2000 hours (1 work year) would result in that individual receiving their annual limit on intake for the radionuclide.
The dose rate (on a per hour basis) then for the 100 meter individual is expressed as

$$DOSERATE100 := FRACon \cdot \left[\frac{5 \cdot rem}{2000 \cdot hr}\right]$$

$$DOSERATE100 = \begin{bmatrix} 0.11 \\ 0.18 \\ 0.19 \\ 0.26 \\ 0.45 \\ 0.55 \\ 1.2 \\ 2.27 \\ 4.93 \\ 7.97 \end{bmatrix} \cdot \frac{rem}{hr}$$

DOSERATE100 shows that as the drill bit descends into the grout, the potential dose to the receptor increases. It is assumed here that the value is constant in any given 1 foot slice of grout. Since the drilling rate is 0.25 in/hr, in 8 hours (a typical onsite release time for hazard classification analyses) the drill bit is still within the ith 1 foot slice.

The postulated release could take place at any 1-foot slice. So if we assume an 8 hour release time, the dose to the 100 meter receptor is given by:

$$DOSE100 := DOSERATE100 \cdot (8 \cdot hr)$$

$$DOSE100 = \begin{bmatrix} 0.89 \\ 1.48 \\ 1.52 \\ 2.06 \\ 3.56 \\ 4.41 \\ 9.64 \\ 18.18 \\ 39.4 \\ 63.79 \end{bmatrix} \cdot rem$$

At the onsite location

For the site boundary location, the same approach is used as for the onsite location. Note that the DAC is not appropriate to be used for nonoccupational workers. The DCG is the correct limit to use. If a member of the public is exposed to 1 DCG concentrations for 1 year, he will receive a dose of 0.1 rem.

$Hoffcs := RATEcs \cdot COQoff$

$Hoffsr := RATEsr \cdot COQoff$

$Hoffpu := RATEpu \cdot COQoff$

$Hoffy := Hoffsr$

$Hoffy90m := Hoffy$

$$FRACoff := \left[\frac{Hoffcs}{DCGcs}\right] + \left[\frac{Hoffpu}{DCGpu}\right] + \left[\frac{Hoffsr}{DCGsr}\right] + \left[\frac{Hoffy}{DCGy90}\right] + \left[\frac{Hoffy90m}{DCGy90m}\right]$$

$$DOSERATEsb := FRACoff \cdot \left[\frac{0.1 \cdot rem}{8766 \cdot hr}\right]$$

The DCG values correspond to a dose of 0.1 rem over a 8766 hour year.

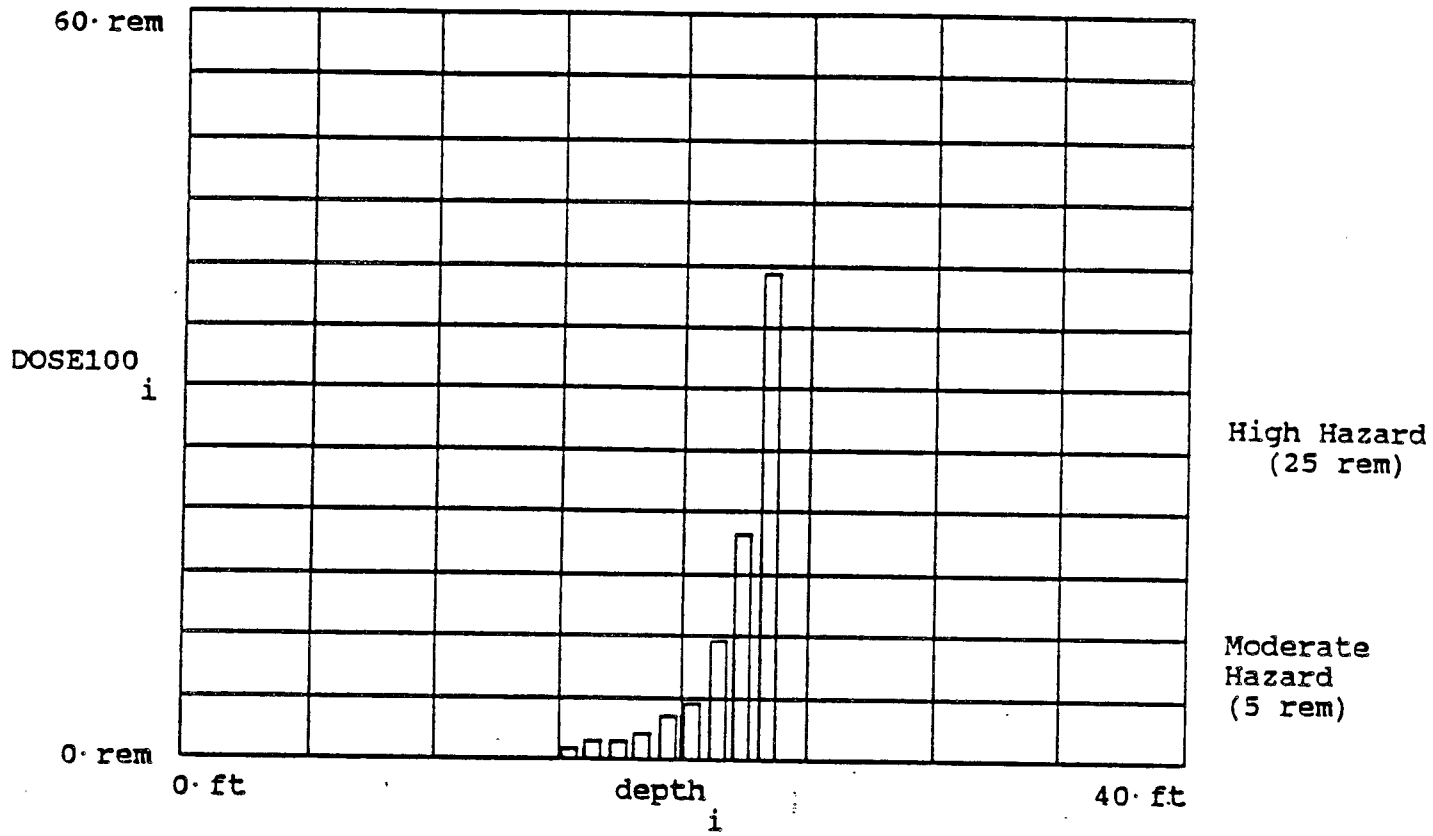
$$DOSEsb := DOSERATEsb \cdot (24 \cdot hr)$$

Typical release time for the site boundary dose determination is 24 hours for the hazard classification analysis.

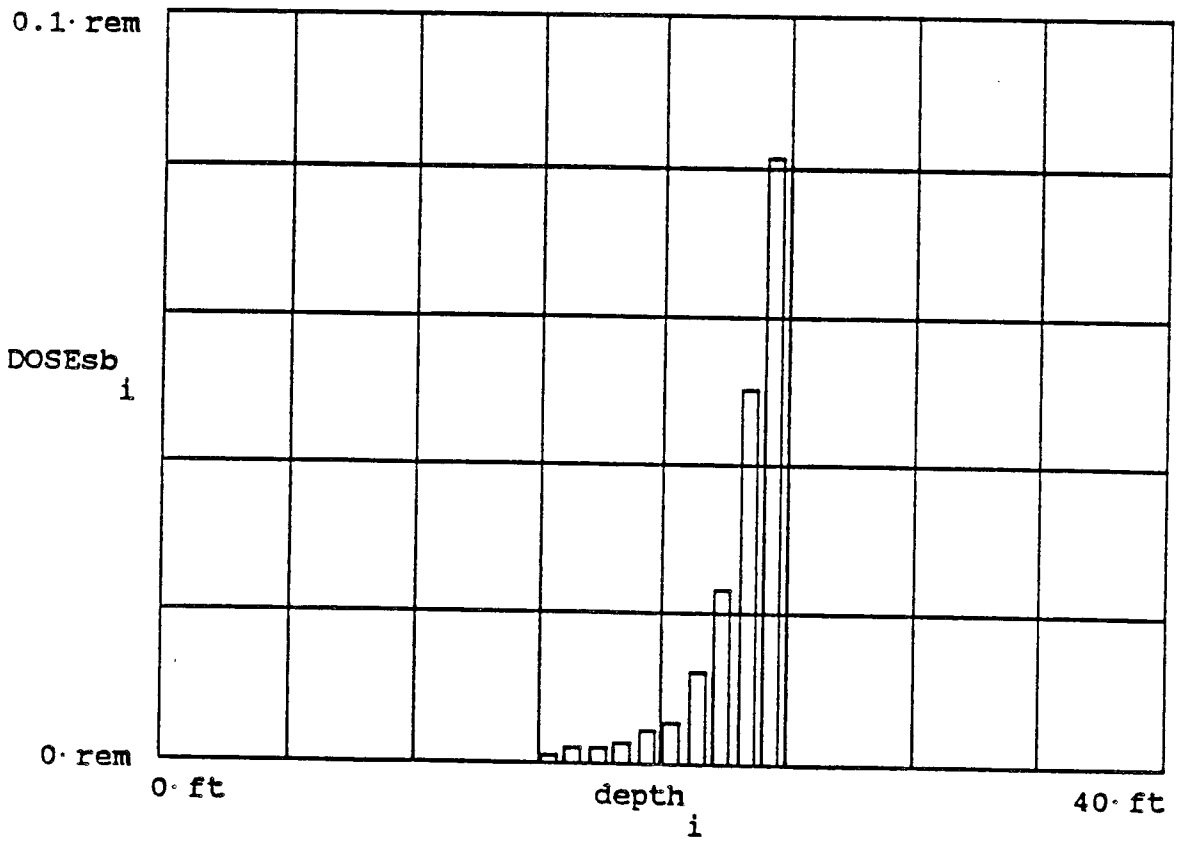
$$DOSEsb = \begin{bmatrix} 0.001 \\ 0.002 \\ 0.002 \\ 0.003 \\ 0.005 \\ 0.006 \\ 0.012 \\ 0.023 \\ 0.05 \\ 0.081 \end{bmatrix} \cdot rem$$

At the site boundary location

Dose to the Onsite Receptor Due to the Postulated Event



Dose to Site Boundary Receptor Due to the Postulated Event



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APPENDIX C

AIR SAMPLE ESTIMATED EXPOSURE

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Westinghouse Hanford Company

Internal Memo

From: Decommissioning Engineering
Phone: 3-1382 R2-77
Date: May 18, 1992
Subject: ESTIMATED EXPOSURE RATE FROM AIR SAMPLE OF GROUT REMOVAL FROM TANK 241-CX-72

81490-92-DRS-054

To: File C.11.3

cc: R. B. Bendixsen N1-75
S. G. Marske R2-77
K. S. Pedersen S4-67
R. G. Shuck S4-67
DRS File/LB

I have calculated the approximate gamma exposure rate from an air sample of the grout removal work to be down in tank 241-CX-72 based upon the isotopic concentration in the grout being great enough to cause higher than a low hazard activity, i.e., more than 5 rem to an onsite worker. These calculations are based on the results of the dose calculations shown in the recent safety assessment by Mr. Rex Bendixsen for grout removal.

The only isotopes that can be present in any significant quantities are cesium-137 and daughter (barium-137m), strontium-90 and daughter (yttrium-90), plutonium-239 and its daughters. This conclusion is based upon normal fission product production ratios and the age of the waste in the tank. Of these, only the barium-137m has a gamma of any significance.

The air sampler for grout removal is designed to remove a sample isokinetically at a rate of 2 cfm. This is one-five hundredths of the vacuum line air flow of 1,000 cfm from which it is sampling. Therefore the air sample should contain one-five hundredths of the waste removed.

The waste removal rate is currently designed to be 0.25 inches per hour from the tank. This amounts to 41.2 liters per 8 hours from the 40-inch diameter tank.

$$\frac{0.25 \text{ inches}}{\text{hour}} \bullet 8 \text{ hours} \bullet \pi \bullet (20 \text{ inches})^2 \bullet \frac{\text{liter}}{61 \text{ in}^3} = 41.2 \text{ liters}$$

According to the dose calculations given in the grout removal safety assessment by Mr. Bendixsen, the concentration of cesium-137 (barium-137m) in the grout required to get 5 rem to an onsite worker is approximately 0.189 Ci/liter as shown in the following table.

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Page 2
May 18, 1992

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Assumed Concentration in waste tank (Ci/l)		Conc required to get 5 rem onsite
Cs-137	31.6	1.89E-01 Ci/l
Sr-90	27.2	1.62E-01
Y-90	27.2	1.62E-01
Pu-239	0.139	8.30E-04

Depth in Tank (feet)	Exposure Rate (R/hr)	Cs-137 Conc. (Ci/l)	Onsite Dose (rem)	Ci/l per rem
15.4	3.8	0.4	11.6	0.0345
16.4	6.3	0.7	19.3	0.0363
17.4	6.5	0.8	19.9	0.0402
18.4	8.8	1.0	26.9	0.0372
19.4	15.2	1.8	46.5	0.0387
20.4	18.8	2.2	57.5	0.0383
21.4	41.1	4.8	125.7	0.0382
22.4	77.5	9.0	237.1	0.0380
23.4	168.0	19.5	513.9	0.0379
24.4	272.0	31.6	832.0	0.0380
			Ave=	0.037716
			Cs-137 conc required to get 5 rem	0.188584

Therefore, the amount of cesium-137 (barium-137m) on the air filter after 8 hours of waste removal would be 0.0156 curies.

$$\frac{0.189\ \text{Ci}}{\text{liter}} \cdot 41.2\ \text{liters} \cdot \frac{1}{500} = 0.0156\ \text{curies}$$

Page 131 of the Radiological Health Handbook (January 1970) gives a gamma constant of 0.33 R/hr at 1 m/Ci for cesium-137 (barium-137m). Using the inverse square law and multiplying by the estimated inventory on the filter gives 1.85 R/hr at 2 inches from the sample (the effective measurement point of a Cutie Pie held at contact with the sample).

$$\frac{0.33\ \text{R/hr}}{\text{Ci}} \cdot (1\ \text{m}/0.05\ \text{m})^2 \cdot 0.0156\ \text{Ci} = 2.06\ \text{R/hr}$$

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Page 3
May 18, 1992

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Applying a conservatism factor of 0.5 and rounding for simplicity yields a safety constraint of 1 R/hr from the sample to assure the grout removal operation does not exceed low hazard.

$$2.06 \text{ R/hr} \cdot 0.5 = 1.03 = 1.0 \text{ R/hr}$$

D. R. Speer, Manager
Decommissioning Engineering

cmj

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Title: Phase 1 Grout Removal From Tank 241-CX-72 Safety Assessment

Unclassified Category: UC-N/A

Impact Level: 3

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Author/Requestor (Printed/Signature): R. B. Bendixsen — Date: May 19, 1992

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