WCH-421
Rev. 0

River Corridor
Closure Contract

Final Hazard Categorization for Removal of 105-B Reactor Building Fuel Storage Basin Transfer Pits Sediment

February 2011

For Public Release

Washington Closure Hanford

Prepared for the U.S. Department of Energy, Richland Operations Office
Office of Assistant Manager for River Corridor

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ACRONYMS

AED	Aerodynamic Equivalent Diameter
ALARA	as low as reasonably achievable
ARF	airborne release fraction
CFR	*Code of Federal Regulations*
DOE	U.S. Department of Energy
ERDF	Environmental Restoration Disposal Facility
FHC	final hazard classification
FSB	fuel storage basin
JHA	job hazard analysis
RadCon	Radiological Control
RF	respirable fraction
RWP	radiological work permit
SSHASP	*Site-Specific Health and Safety Plan*
TQ	threshold quantity
WCH	Washington Closure Hanford

DEFINITIONS

Airborne Release Fraction (ARF): The emission factor used to estimate the portion of the radiological or non-radioactive hazardous material inventory that can be suspended in air and made available for airborne transport under a specific set of induced physical stresses. ARF values are applicable to release episodes (events) that are of limited duration as opposed to long-term continuous releases. The U.S. Department of Energy (DOE) handbook, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* (DOE 2000), provides ARF values for specific release mechanisms.

Airborne Release Rate: The coefficient used to estimate the amount of material that can be suspended in air and made available for airborne transport per unit time under a specific set of induced physical stresses. The DOE handbook (DOE 2000) provides Airborne Release Rate values for specific release mechanisms.

Material at Risk: The amount of radionuclides or non-radioactive hazardous materials available to be acted on by a given physical stress.

Respirable Fraction (RF): The fraction of airborne radionuclides present as particles that can be transported through air and inhaled into the human respiratory system. Particles with an aerodynamic equivalent diameter of 10 μm or less are typically considered respirable.

Segmentation: The division of radioactive or non-radioactive hazardous material inventories into areas within the facility that common cause events would be unlikely to result in an accidental release of these radioactive or non-radioactive hazardous materials from more than one designated segment. Segmentation is justified, provided that radioactive or non-radioactive hazardous materials in one facility segment cannot interact with radioactive or non-radioactive hazardous materials in another facility segment.

Threshold Quantity (TQ): Radionuclide thresholds published in *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports* (DOE 1997) that are used to establish the demarcation between hazard category 2, hazard category 3, and below hazard category 3 activities. Extremely hazardous substance thresholds published in 40 *Code of Federal Regulations* (CFR) 302 and 29 CFR 1910.119 are used to determine when a facility or activity is subject to emergency planning requirements (40 CFR 355) and the guidelines of process safety management (29 CFR 1910.119).

METRIC CONVERSION CHART

Into Metric Units		
If You Know	*Multiply By*	*To Get*
Length		
inches	25.4	millimeters
inches	2.54	centimeters
feet	0.305	meters
yards	0.914	meters
miles	1.609	kilometers
Area		
sq. inches	6.452	sq. centimeters
sq. feet	0.093	sq. meters
sq. yards	0.836	sq. meters
sq. miles	2.6	sq. kilometers
acres	0.405	hectares
Mass (weight)		
ounces	28.35	grams
pounds	0.454	kilograms
ton	0.907	metric ton
Volume		
teaspoons	5	milliliters
tablespoons	15	milliliters
fluid ounces	30	milliliters
cups	0.24	liters
pints	0.47	liters
quarts	0.95	liters
gallons	3.8	liters
cubic feet	0.028	cubic meters
cubic yards	0.765	cubic meters
Temperature		
Fahrenheit	subtract 32, then multiply by 5/9	Celsius
Radioactivity		
picocuries	37	millibecquerel

Out of Metric Units		
If You Know	*Multiply By*	*To Get*
Length		
millimeters	0.039	inches
centimeters	0.394	inches
meters	3.281	feet
meters	1.094	yards
kilometers	0.621	miles
Area		
sq. centimeters	0.155	sq. inches
sq. meters	10.76	sq. feet
sq. meters	1.196	sq. yards
sq. kilometers	0.4	sq. miles
hectares	2.47	acres
Mass (weight)		
grams	0.035	ounces
kilograms	2.205	pounds
metric ton	1.102	ton
Volume		
milliliters	0.033	fluid ounces
liters	2.1	pints
liters	1.057	quarts
liters	0.264	gallons
cubic meters	35.315	cubic feet
cubic meters	1.308	cubic yards
Temperature		
Celsius	multiply by 9/5, then add 32	Fahrenheit
Radioactivity		
millibecquerels	0.027	picocuries

1.0 PROJECT DESCRIPTION

This final hazard categorization (FHC) document provides the safety basis for removal of the sediment material and miscellaneous debris contained within the 105-B Reactor Building fuel storage basin (FSB) transfer pits. The sediment material was moved to the transfer pits during cleanup and stabilization activities of the fuel storage basin in 1985. The FHC document provides a hazard baseline, identifies and evaluates the associated hazards, identifies the appropriate controls, documents the FHC, and documents the controls necessary for surveillance and maintenance of the reactor buildings. In addition, this document provides a basis for evaluating proposed activities and discovered conditions.

This document includes the following:

- A description of the sediment removal activities to be performed
- An assessment of the inventory of radioactive and other hazardous materials within the transfer pits
- Identification of the hazards associated with the sediment removal activities for the transfer pits
- Identification of internally and externally initiated accidents having the potential to result in significant consequences for workers, the environment, or the public
- An assessment of the bounding radiological consequences of the potentially significant accidents
- Determination of the FHC based on the bounding material at risk
- Identification of the controls (including commitments) necessary to control the identified hazards and to ensure that the FHC remains valid.

Construction started on the 105-B Reactor Building in 1943. The reactor achieved start-up on September 26, 1944. Final shutdown (last date of critical operation) occurred February 13, 1968. (Note: The 105-B Reactor Building was shut down and held in standby from March 19, 1946, to July 1, 1948.)

1.1 FACILITY/SITE DESCRIPTION

This FHC document applies to the 105-B Reactor Building FSB and associated transfer pits. The rest of the 105-B Reactor Building is not addressed by this FHC document because it is covered by a separate safety basis. An Administrative Interface Agreement (WCH 2010a) identifies the interfaces between Mission Support Alliance and Washington Closure Hanford (WCH). The Administrative Interface Agreement establishes the following:

- Facility access and control requirements
- Work performance requirements within the 105-B Reactor Building FSB areas

- Safety, emergency notifications, and reporting requirements
- Hazardous material, environmental, radiological safety, training, and posting requirements including waste handling and disposal protocol.

The 105-B Reactor Building and associated FSB are described in greater detail in the following sections.

1.1.1 Reactor Building

The 105-B, 105-C, 105-D, 105-DR, 105-F, and 105-H Reactor Buildings were very similar in design and layout and had similar operational histories. These older reactor buildings are much different in design and size than the 105-KE and 105-KW Reactor Buildings and the 105-N Reactor Building. A description of the reactor building is provided below.

The 105-B Reactor Building was a plutonium production reactor. The reactor building housed the nuclear reactor and equipment directly associated with reactor operations. The reactor building is a non-airtight industrial structure of reinforced-concrete construction in the lower portions and of concrete block construction in the upper portion. Roof construction is of reinforced-concrete or precast concrete roof panels. The older reactor buildings that have undergone interim safe storage (all but the 105-B Reactor Building) have had the roofs replaced above the core areas, and significant portions of the buildings have been demolished.

The 105-B Reactor Building is the only standard reactor building that remains intact at the Hanford Site. It is currently maintained in a condition to allow controlled access for public tours that are arranged through and sponsored by the U.S. Department of Energy (DOE).

Damaged pre-cast concrete roof panels were repaired (circa 1994) in the "C" Elevator area, Corridor 1, Corridor 221, and valve pit room. Further repairs were performed (circa 2002) in the fuel storage viewing room, Corridor 221, Electrical Equipment Room, Corridor 227, valve pit room, and fan room.

Safety nets in the 105-B Reactor Building front-face work area ceiling area and a restraint for the existing vertical curtain were installed in 2002. The safety nets and curtain restraint will provide protection from potential failed roof panels and a potential failed vertical curtain retraction system. Expanded metal was installed in 2002 on the 105-B Reactor Building valve pit existing walkway guardrails. Three expanded metal lockable gates were installed on the walkway. The expanded metal partitions and lockable expanded metal gates provide protection from trips and falls and control access to the valve pit lower area.

A ventilation system was installed in 2003. This system normally does not provide supply air during occupancy but is used to sweep the area with fresh air during the night as a means to cool down the areas that may have been heated by the lights and from people, and reduce the radon levels as necessary. The air is supplied into the clean areas of the building, which slightly pressurizes the building as the air is released to the outside through the various cracks in this non-airtight industrial structure. The ventilation system is composed of the following three zones:

- Zone 1: Supply Fan Room 230a, Valve Pit Area #315, and Flow Lab #231a
- Zone 2: Corridors 5 and 227b; Corridor 227a; Electrical Equipment Room 223; Toilet 225a; Accumulator Room 222; Corridors 2, 3, and 211b; Corridors 4 and 211; Office/Storage 228a; and Viewing Room 414
- Zone 3: Control Room 220 and Office 219.

Non-tempered air is supplied for Zones 1 and 2 through roof-mounted ventilation units, while the Zone 3 ventilation unit is located inside the building. The units are provided with medium efficiency inlet filters that can handle differential pressure up to 1.5 in. The combined airflow from the units is approximately 637 m^3/min (22,500 ft^3/min) of air at 2.5 in. W.G.

Figure 1-1 shows the ground-level layout of an older reactor building. This figure is general in nature. The ventilation system was designed such that air flowed from office and service areas to potentially contaminated areas. The air was exhausted to the atmosphere through the 116 Reactor Building Exhaust Stack.

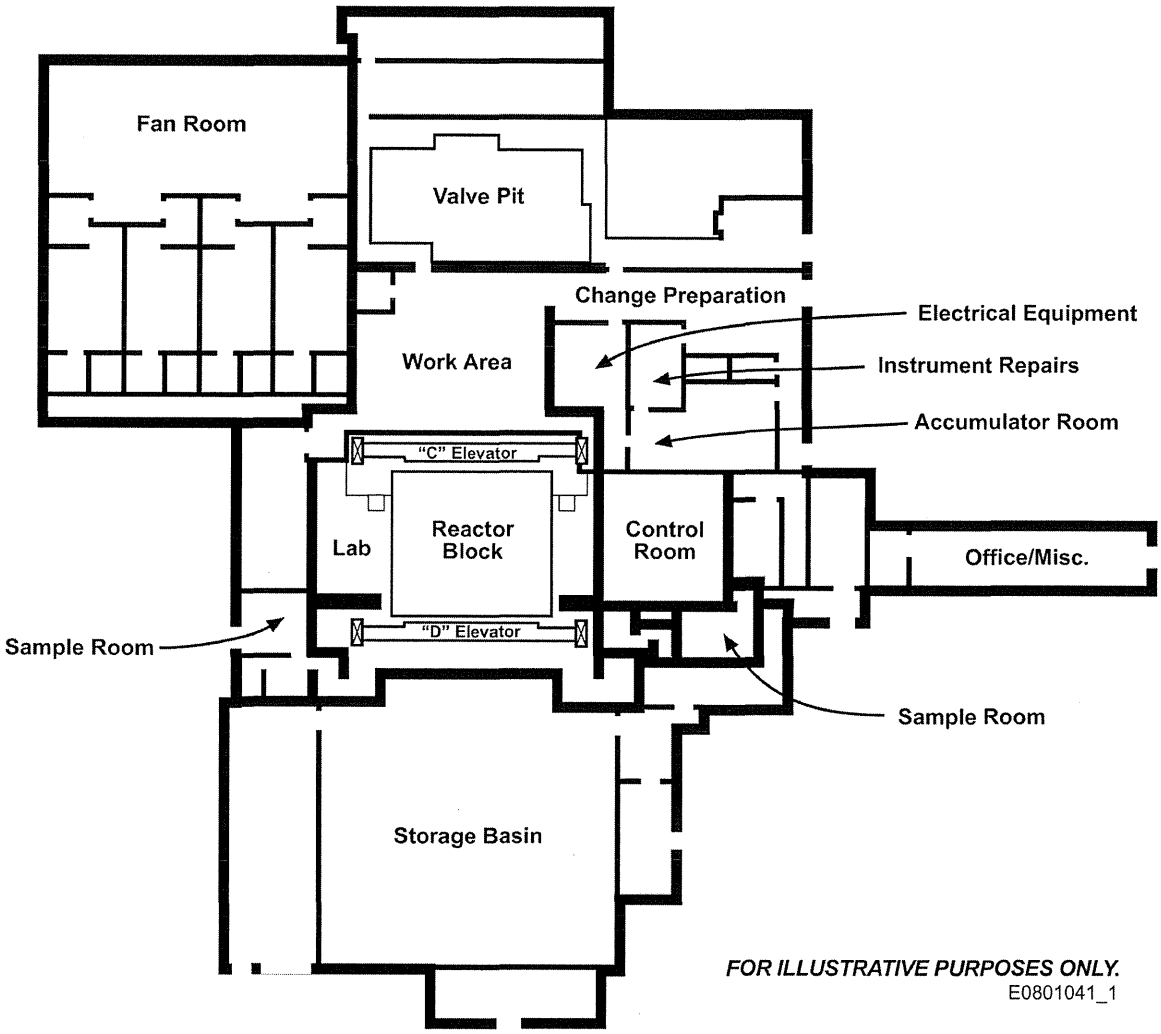
The reactor building can essentially be divided into the following three areas:

1. Areas within the massive, reinforced-concrete shield walls of the building surrounding the reactor process areas. This area includes the inner rod room, ball 3x system, laboratories, front-face work area, and the reactor block, which is composed of the graphite moderator stack, thermal and biological shields, process tubes, and the safety and control systems. Figure 1-1 provides a plan view of these areas. The standard reactor blocks are discussed in Section 1.1.1.1.

2. The reactor building FSB. The 105-B Reactor Building FSB is discussed in Section 1.1.1.2.

3. The general ancillary areas of the reactor building (excluding the FSB) located outside the shield walls. This area provided ancillary support areas during former reactor operations. The prior support areas included office areas, the reactor control room, tool storage rooms, restrooms, cooling water influent areas, change rooms, ventilation equipment areas, electrical system areas, and other infrastructure support. Figure 1-1 provides a plan view of these areas.

Figure 1-1. Older Reactor Building, Plan View, Ground Level.



1.1.1.1 Reactor Building Reactor Block. The reactor block is located near the center of the building. The reactor block consists of a graphite moderator stack (11 m [36 ft] high by 11 m [36 ft] wide by 8.5 m [28 ft] deep) encased in a cast iron thermal shield (20 to 25 cm [8 to 10 in.] thick) and a biological shield consisting of alternating layers of steel and Masonite® (132 cm [52 in.] thick). The entire block rests on a massive concrete foundation. Figure 1-2 shows the typical older reactor block construction.

All penetrations through the block outer shielding were provided with cover-gas seal designed tubes/sleeves/thimbles to maintain cover-gas containment.

During final shutdown of the reactor, a number of procedures were performed to contain contamination within the reactor block. The process used at the standard reactor buildings is

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assumed to be similar to the process used at the 105-KE and 105-KW Reactor Buildings. The procedures used at the 105-KE and 105-KW Reactor Buildings can be found in the reactor deactivation manuals (BHI 1998a, 1998b). Examples of the activities performed include installing process tube caps, draining cover-gas lines, closing drain valves, and draining all water lines.

1.1.1.2 Fuel Storage Basin. The FSB served as a collection, storage, and transfer facility for the irradiated fuel elements discharged from the reactor. It consists of a fuel element pickup area, storage area, and transfer area. Irradiated fuel elements were stacked in buckets under water in the pickup area, transported by an overhead monorail to the storage area, and temporarily stored underwater to allow decay of short-lived radionuclides before transporting for reprocessing. Figure 1-1 shows the location of the FSB, while the 105-B Reactor Building FSB layout is shown in Figure 1-3.

The 105-B Reactor Building FSB was drained and cleared of debris in late 1985, and a fixative was applied to radiologically contaminated surfaces. About 17 m^3 (600 ft^3) of sediment (radiologically contaminated iron oxides and silt) was moved to the transfer pits using a "washdown approach" where it remains covered by a wood deck. (Note: Some historical documents indicate that a layer of sand was placed on top of the sediment before it was covered with a wood deck, however, the existence of this sand layer has not been confirmed.) The washdown approach used fire hoses to move sediment from the floor of the storage area to the transfer pits. Water was supplied to the fire hoses by recirculating water out of the transfer pits, through a filter press, and then back to the basin. Dewatering manifolds consisting of a matrix of CUNO® filters were placed at the bottom of the transfer pits along with 46 cm (18 in.) of sand and gravel to dewater the sediment after transfer was complete. Use of a vacuum system for sediment removal before washdown was limited to identified, high-dose rate areas. Also, use of the dams that had been built as a contingency to maintain water levels for shielding from direct dose was not required. Figure 1-4 shows a conceptual drawing of the sediment transfer system.

Figure 1-2. Older Reactor Block Construction (Typical).

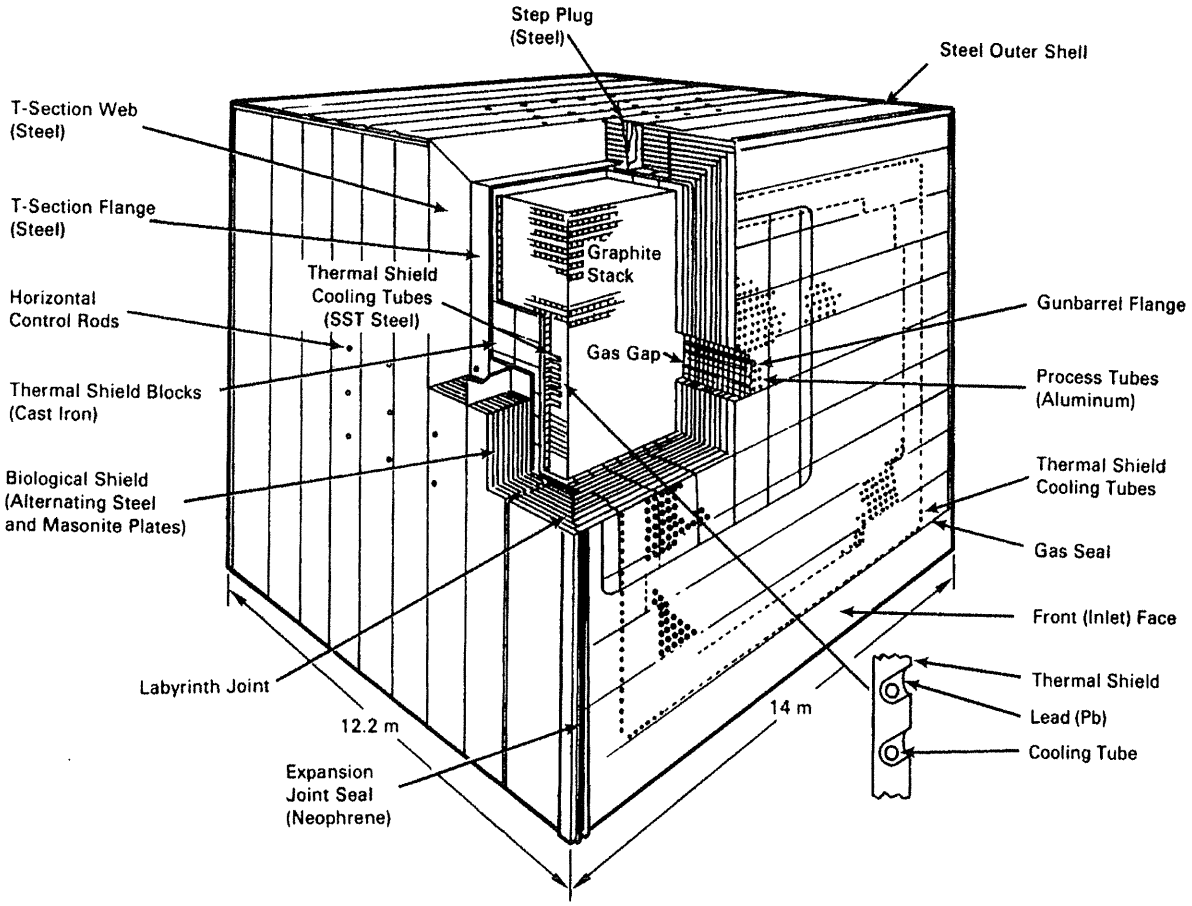


Figure 1-3. 105-B Fuel Storage Basin Layout.

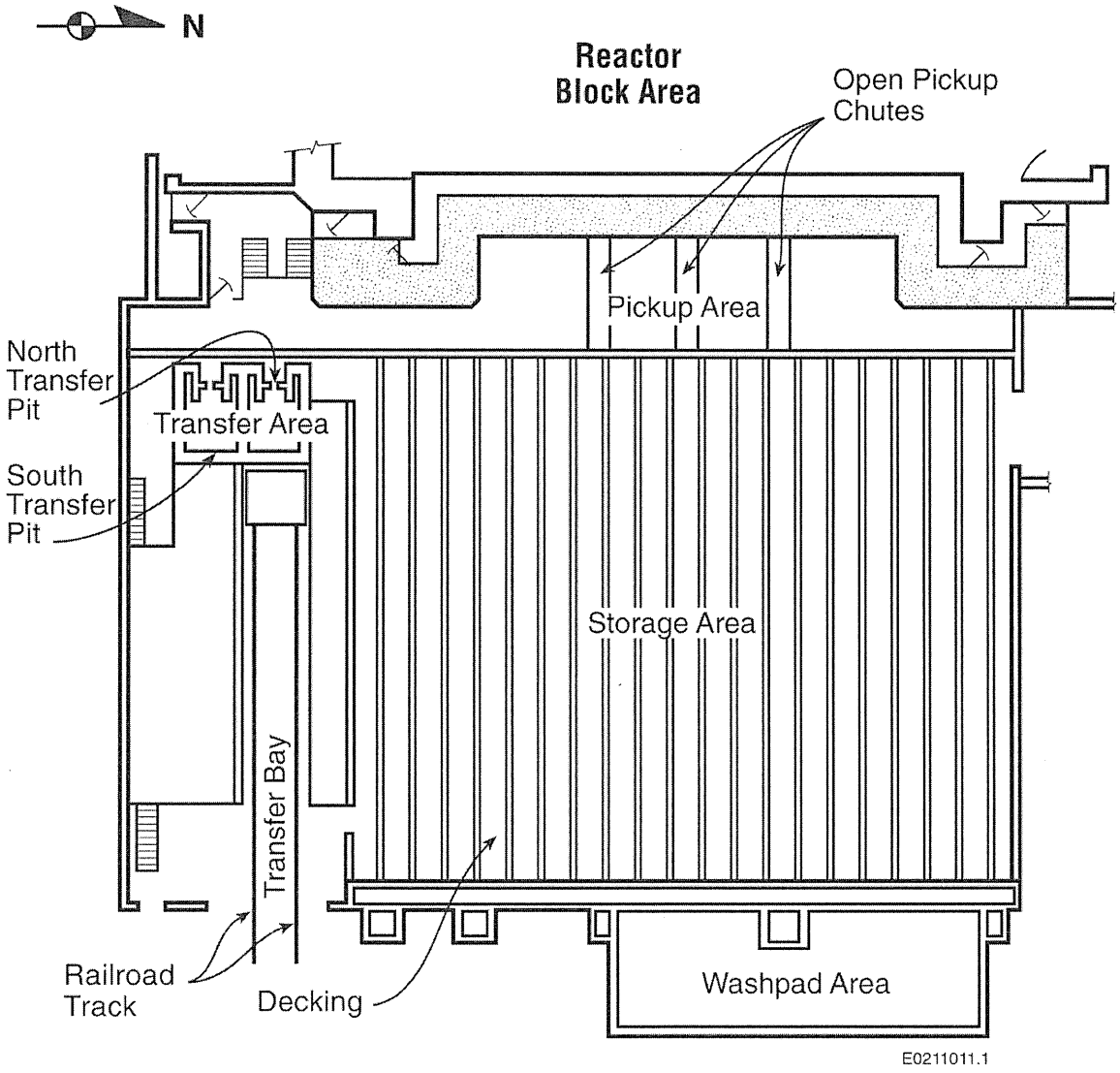
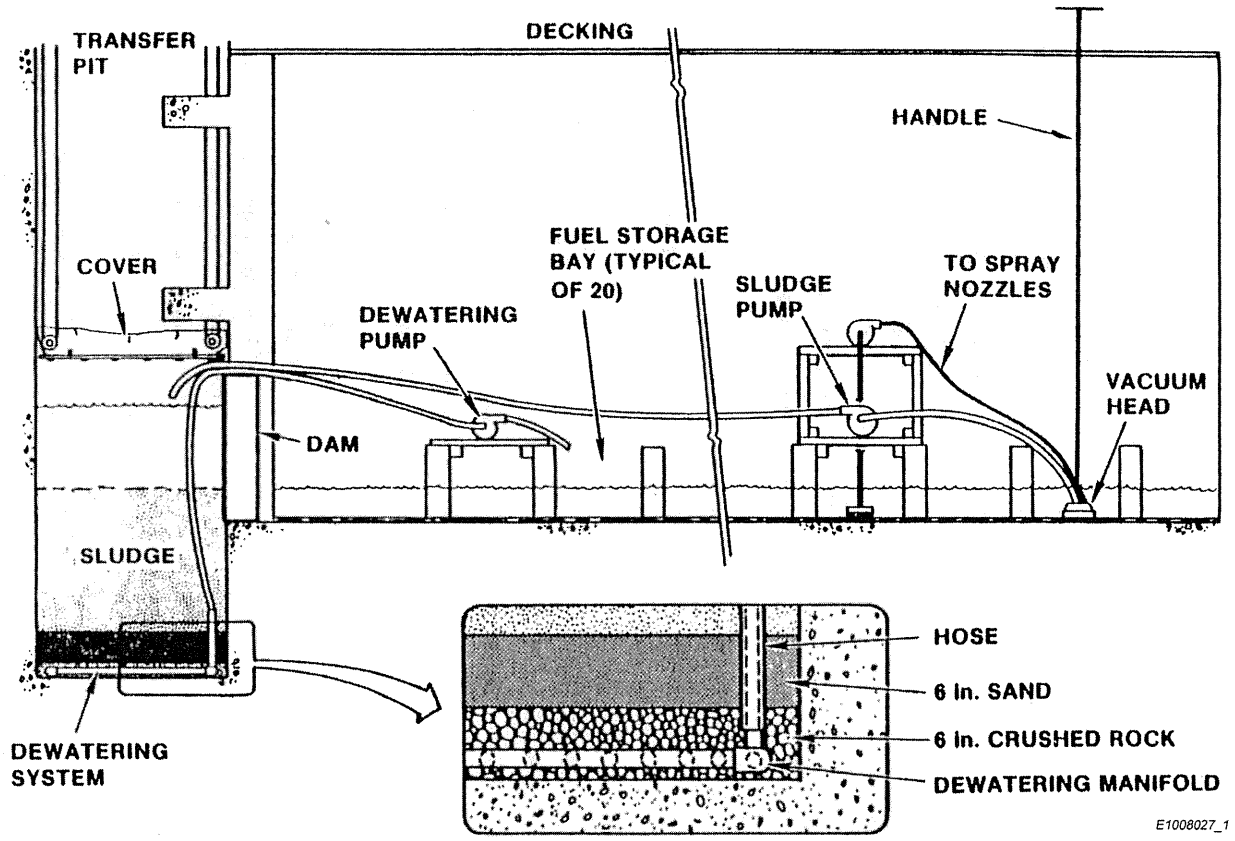


Figure 1-4. Conceptual Drawing of Sediment Transfer.



1.1.2 Active Systems

The only active operating systems associated with the 105-B Reactor Building are electrical power, emergency lighting and ventilation system, and a water system. Electrical power is currently used to supply interior lighting and convenience outlets in the 105-B Reactor Building. Electrical power is also used to supply space heaters in the 105-B Reactor Building. Electrical power will be used to supply equipment in the FSB during sediment removal activities. Hazards associated with the 105-B Reactor Building were evaluated in *Risk Management Study for the Retired Hanford Site Facilities: Qualitative Risk Evaluation for the Retired Hanford Site Facilities* (WHC 1993), and all of the hazards identified as moderate, serious, or critical have been mitigated.

The emergency lighting and space heaters at the 105-B Reactor Building are present because public tours are occasionally conducted, and to minimize deterioration of the roof as a result of freezing and thawing. The emergency lighting system automatically activates upon loss of power to the building or a loss of the lighting circuit. The space heaters are manually operated as needed.

The ventilation system at the 105-B Reactor Building normally does not provide supply air during occupancy, but is used to sweep the area with fresh air during the night. This ventilation

sweep serves as a means to cool down the areas that may have been heated by lights and visitors and reduce radon levels as necessary. The air is supplied into the clean areas of the building, which slightly pressurizes the building as the air is released to the outside through the various cracks in this non-airtight industrial structure.

1.2 PROCESS HISTORY

The 105-B Reactor Building is one of the nine water-cooled, graphite-moderated reactors constructed along the Columbia River by the U.S. Government. All of the reactors are situated within the 100 Areas on the Hanford Site. The reactors were all constructed to support the plutonium production effort initiated under the Manhattan Project.

The reactor building contained the reactor core and equipment directly associated with reactor operations. Cooling water for the reactor was withdrawn from the Columbia River, filtered and treated, pumped through the reactor block, and then returned to the river in a single-pass process. The 105-B Reactor Building was the first full-scale production nuclear reactor ever constructed. The 105-B Reactor Building was placed on the National Register of Historical Places on April 3, 1992, by the National Park Service of the U.S. Department of the Interior. The 105-B Reactor Building was also listed as a National Historic Mechanical Engineering Landmark in 1976 by the American Society of Mechanical Engineers' History and Heritage Committee. In October 1993, the American Society of Civil Engineers named the 105-B Reactor Building a National Civil Engineering Landmark. The American Nuclear Society presented the Nuclear Landmark Award to the 105-B Reactor Building in 1992. The *105-B Reactor Facility Museum Phase 1 Feasibility Study Report* (BHI 1995) was issued in October 1995.

Because of the length of time that has elapsed since the reactor building was operational, no information was obtained regarding significant, historical off-normal events that could potentially be relevant to understanding current conditions within the structure. The information relied upon to understand the current conditions have been gathered subsequent to final shutdown of the reactors (primarily within the last 25 years).

1.3 PROJECT OPERATIONS AND ACTIVITIES

The 105-B Reactor Building is currently undergoing routine surveillance and maintenance. Routine surveillance is primarily a directed walkdown, with procedural requirements to pay particular attention to several classes of problems. In addition to a walkdown inspection, the 105-B Area has requirements for routine fire extinguisher inspections and emergency lighting inspection/tests. The objective of routine surveillance is to ensure that any unfavorable conditions or trends are recognized and evaluated such that appropriate corrective actions can be initiated. Maintenance consists primarily of the correction of deficiencies identified during surveillance.

The planned activity for the 105-B Reactor Building FSB is to remove the sediment remaining in the transfer pits. Most of the sediment will be removed using a flexible mechanical auger system. The auger will be used to move sediment material from the bottom of the transfer pits to a waste container (i.e., Environmental Restoration Disposal Facility [ERDF] container) staged in the transfer bay (a.k.a., rail car) area of the FSB. The waste container will be lowered into the transfer bay area from the back of a truck, which will remain outside of the FSB. (Note: During

reactor operations, the transfer bay area was used to load irradiated fuel into rail cars for shipment to the 200 Areas for processing.) The container will then be loaded onto a truck and transported to the ERDF to allow disposal of the sediment material. Project planning indicates that a total of five ERDF containers (about 3 m^3 [120 ft^3]/container) will likely be used to transport the sediment. Small amounts of sediment and other materials (e.g., dewatering system, sand, gravel) present in the bottom of the transfer pits that cannot be removed with the auger system will be removed using other means such as hand tools (e.g., shovel) to load the material into containers (e.g., drums) for transport and disposal.

1.4 SUMMARY OF INVENTORY

The reactor block and FSB transfer pits are sufficiently separated within the reactor building to warrant different segments. In addition, Figure 1-5 shows that there are two transfer pits in the FSB that contain the sediment material. The two pits are separated by a thick concrete wall over their entire depth. Consequently, the WCH activities involving removal of sediment from the fuel storage basin transfer pits can be segmented from activities involving the rest of the reactor building, and activities involving each pit can be segmented.

Each segment of the reactor buildings within the scope of this document contains both radiological and non-radiological hazardous materials. Hazardous material inventories were developed independently by type and segment. Development of the hazardous material inventories is discussed in the following sections.

1.4.1 Reactor Block Radionuclide Inventory

The eight reactor blocks have different radionuclide inventories based on the different designs, irradiation histories, and construction materials used. The total radionuclide inventory of the reactor block will include all of the components shown in Tables 16-23 of *Radionuclide Inventory and Source Terms for the Production Reactors at Hanford* (UNC 1987). These components include the graphite stack, thermal shield, process tubes, control system, and biological shield. For each component, the maximum inventory for each nuclide was used from each of the eight reactors to construct a bounding inventory.

The bounding reactor block inventory is provided in Table 1-1. Inventory values from reactors other than 105-KE are footnoted in Table 1-1. The inventory is derived by decaying the inventory provided in UNC (1987) from March 1, 1985, to March 1, 2007.

1.4.2 Fuel Storage Basin Radionuclide Inventory

The 105-B Reactor Building FSB inventory is shown in Table 1-2. The current inventory is derived from data obtained from two sediment samples taken from the transfer pits in early 2010 as documented in *Validation of 105-B Final Hazard Categorization Against New Sample Data for Fuel Storage Basin Transfer Pit Sediment* (WCH 2010b). The total inventory exceeds the hazard category 3 threshold quantity (TQ) in DOE (1997). The "settled solids" average bulk density of the sediment was determined to be 1.50 g/cm^3. Based on an estimated sediment volume of 17 m^3 (600 ft^3) and this density, the mass of sediment is calculated as 2.55×10^7 g. This is only about 50% of the sediment mass (5.00×10^7 g) assumed in the *Final Hazard Categorization for Surveillance and Maintenance of 100 Area Reactor Facilities* (WCH 2008) for determination of the FHC for surveillance and maintenance.

Table 1-1. Reactor Block Radionuclide Inventory.

Isotope	March 1, 1985, Inventory (Ci)						March 1, 2007 Total Inventory (Ci)
	Graphite Stack	Process Tubes	Thermal Shield	Control System	Biological Shield	Total	
H-3	3.00E+04					3.00E+04	8.71E+03
C-14	7.00E+03					7.00E+03	6.98E+03
Cl-36	5.40E+01					5.40E+01	5.40E+01
Ca-41	1.90E+02[a]				1.50E+01	2.05E+02	2.05E+02
Ni-59	2.00E+00[c]	1.30E+01	9.00E+00			2.40E+01	2.40E+01
Ni-63	2.80E+02[c]	1.70E+03	1.20E+03			3.18E+03	2.73E+03
Co-60	1.00E+02[a]	3.50E+02[b]	1.75E+04	1.10E+02		1.81E+04	1.00E+03
Sr-90	1.00E+01	3.00E-01				1.03E+01	6.06E+00
Mo-93		2.00E-01	6.00E-02			2.60E-01	2.59E-01
Zr-93		1.10E+01				1.10E+01	1.10E+01
Nb-94	1.10E+00	6.00E-01	3.00E-02			1.73E+00	1.73E+00
Tc-99		3.00E-02	3.00E-03			3.30E-02	3.30E-02
Ag-108m[e]			4.00E-02			4.00E-02	3.86E-02
Ba-133	3.40E+01[d]					3.40E+01	7.97E+00
Cs-137	3.00E+01					3.00E+01	1.81E+01
Eu-152	4.00E+01	2.00E+00				4.20E+01	1.36E+01
Eu-154	2.00E+01	1.60E+00				2.16E+01	3.66E+00
Pu-239	1.00E+00					1.00E+00	9.99E-01
Am-241	3.00E-01					3.00E-01	2.90E-01

[a] Source: Table 16 of UNC 1987 (105-B).
[b] Source: Table 17 of UNC 1987 (105-C).
[c] Source: Table 18 of UNC 1987 (105-D).
[d] Source: Table 7 of UNC 1987 (105-D).
[e] Assumed long-lived Ag-108m intended in UNC 1987, not short-lived Ag-108.

Table 1-2. 105-B Fuel Storage Basin Inventory.

Isotope	Concentration[a], Ci/g	Transfer Pit[a] Inventory, Ci	FSB Wall and Floor Inventory, Ci	Total Inventory, Ci
C-14	2.40E-10	6.12E-03	6.12E-04	6.73E-03
Co-60	5.49E-09	1.40E-01	1.40E-02	1.54E-01
Ni-63	2.50E-06	6.37E+01	6.37E+00	7.01E+01
Sr-90	3.66E-07	9.33E+00	9.33E-01	1.03E+01
Tc-99	3.89E-10	9.91E-03	9.91E-04	1.09E-02
Cs-137	6.22E-07	1.59E+01	1.59E+00	1.75E+01
Eu-152	2.93E-08	7.47E-01	7.47E-02	8.22E-01
Eu-154	4.10E-09	1.04E-01	1.04E-02	1.14E-01
Th-232	8.33E-13	2.12E-05	2.12E-06	2.33E-05
U-233	2.00E-10	5.11E-03	5.11E-04	5.62E-03
U-234	5.14E-10	1.31E-02	1.31E-03	1.44E-02
U-235	2.27E-11	5.78E-04	5.78E-05	6.36E-04
U-236	1.28E-11	3.25E-04	3.25E-05	3.58E-04
U-238	5.48E-10	1.40E-02	1.40E-03	1.54E-02
Pu-239/240	4.24E-08	1.08E+00	1.08E-01	1.19E+00
Am-241	2.18E-08	5.56E-01	5.56E-02	6.12E-01

[a] Source: WCH 2010b

However, it is believed that a mass of 2.55 x 10^7 g is more appropriate for the 105-B Reactor Building Transfer Pits because (1) the actual sediment mass at the 105-C Reactor Building Transfer Pits, which also had as estimated volume of 17 m^3 (600 ft^3), was determined to be about 2.6 x 10^7 g and (2) the measured density of N Reactor Building fuel storage basin sediment was about 1.5 g/cm^3. It is assumed that the inventory is split equally between the two transfer pits.

Sample analysis included particle size determination to provide information on the amount of respirable material present in the sediment material. Particle size analysis initially determined that about 8% (average of two samples) passed through a sieve of mesh size 400. This means that 8% of the sample (sediment) material is smaller than 37 µm. Further analysis determined that only 50% of this material (<37 µm) was smaller than about 14 µm, and only 25% of this material (again <37 µm) was smaller than about 4.5 µm. The method used to determine the particle size distribution below 37 µm (sieving) approximates the geometric diameter of the particles. However, it is the Aerodynamic Equivalent Diameter (AED) that is of interest rather than the geometric diameter for determining the fraction of the sediment that is respirable. Section 1.2 of DOE (2000) provides an equation (1-2) that relates the AED to the geometric diameter through several factors, and then proceeds to indicate that the AED may be estimated from the geometric diameter by simply multiplying the geometric diameter by the square root of the particle density. This implies that the AED of the particles is equal to or greater than the geometric diameter if the particle density is ≥ 1 g/cm^3. Given (1) a measured bulk density of 1.5 g/cm^3 for the 105-B sediment, (2) a measured dry density of 1.47 g/cm^3 for similar N Basin sediment material (BHI 1998c), and (3) the sediment material was present on the bottom (i.e., floor) of the FSB below the water before cleanup and stabilization (implying a density greater than 1 g/cm^3), a particle density of ≥ 1 g/cm^3 is clearly a reasonable assumption. Therefore, it can be concluded from the particle size determination data that only about 5% of the sediment material is respirable.

The activity on the walls and floor of the 105-B Reactor Building FSB is conservatively assumed to be 20% of the activity of the sediment collected in the transfer pits in WCH (2008). The basis for this assumption is a calculation completed for the 105-C Reactor Building FSB titled *Estimate of 105-C Basin Fixed Contamination Source* (BHI 1996). Review of the calculation indicates that an upper bound estimate of the inventory in the basin concrete was derived from dose rate measurements in the basin and the assumption that the isotopic composition of the inventory in the basin concrete was the same as that of the sediment in the transfer pits. The dose rates predicted by the model used in BHI (1996), assuming 20% of the transfer pit inventory, were 107 mrem/hr and 27 mrem/hr at the bottom of the basin and above the wood planking, respectively. These dose rates were about a factor of 3 greater than the average measured dose rate at the bottom of the basin and about a factor of 9 greater than the maximum measured dose rate above the wood planking. Based on this review of BHI (1996), it is concluded that a more reasonably conservative estimate of the basin (walls and floor) inventory is 10% of the sediment inventory in the transfer pits. The reasonable conservatism of this estimate is supported by data presented in *Radiological Characterization of the Retired 100 Areas* (UNI 1978) for contaminated sludge and concrete in 100 Area retention basins. While the retention basins did not perform exactly the same function as the fuel storage basins, the contamination of the two structures can be considered analogous. Both concrete structures retained water that was in contact with irradiated fuel elements. They also contained some fuel element debris generated by fuel element failures. Table 2.6-4 of UNI (1978) shows that the contaminated concrete structure contained an average of 4% and 6% of the beta-gamma concentration and the plutonium-238 plutonium-239/240 concentration of the sludge, respectively.

1.4.3 Non-Radiological Hazardous Material Inventory

The 105-B Reactor Building contains a large volume of contaminated and non-contaminated lead. This lead is in the form of bricks, blankets, sheets, pipes, plates, shot, and paint. Table 1-3 summarizes the non-radiological hazardous material inventory.

Asbestos was widely used throughout the reactors. Most of the asbestos was used as piping insulation and in transite panels used as structural components. Previous deactivation activities in the 100 Area Reactor Buildings removed some of the asbestos piping insulation. Most of the remaining asbestos is considered friable, and some of the material is radiologically contaminated.

Mercury is found in electrical switches and thermometers throughout the reactor buildings, although much of the mercury has been removed during previous deactivation efforts. Polychlorinated biphenyls are anticipated to be found in the control room, valve pits, areas containing light ballast, the FSB, and in some areas containing oil. Cadmium is present as an alloy with lead (DOE 1989). Other heavy metals are also present in the FSB sediment.

Table 1-3. Non-Radiological Hazardous Material Inventory of the 105 Reactor Buildings.

Location	Hazardous Material
Effectively throughout the 105 Reactor Buildings. Lead-based paint in the exhaust tunnels.	Unknown quantities of lead-based paint and other lead objects. Various types of lead shielding in radiologically controlled area/surface contamination area locations.
	Minor quantities of chemical compounds left over from operation/partial deactivation activities.
Areas of the 105 Reactor Buildings containing switches and other devices.	Unknown quantities of mercury.
Various pieces of equipment located in all buildings; FSB.	Unknown quantities of various heavy metals.
Effectively throughout all buildings (particularly the 105 Reactor Buildings).	Unknown quantities of non-piping asbestos insulation (including transite wall coverings) that may be friable if degraded or damaged.
Control room, valve pit, areas containing light ballast, FSB and some areas containing oil in the 105-B Reactor Building.	Unknown quantities of polychlorinated biphenyls. Potential exists to find small quantities of sodium dichromate in the valve pits.

FSB = fuel storage basin

2.0 HAZARD EVALUATION

This section presents the hazard analysis and hazard categorization process performed for the 105-B Reactor Building FSB transfer pit sediment removal. The hazard analysis and categorization processes were based on guidance contained in NS-1, *Nuclear Safety*, NS-1-3.2, "Guide for Performing Hazard Analysis and Final Hazard Categorizations." The hazard analysis process consists of (1) defining the work activities to be authorized, (2) identifying the hazards associated with the work place (facility), and (3) evaluating the hazards associated with the facility. Given the results of the hazard analysis, an FHC was performed to assess the potential impacts of the sediment removal activity based on a bounding, unmitigated release of radioactive material.

2.1 HAZARD ANALYSIS

Hazards associated with sediment removal activities at the 105-B Reactor Building FSB transfer pits were identified in Appendix A. The hazard analysis in WCH (2008) identified two events that have the potential to significantly affect the sediment material contained within the transfer pits during surveillance and maintenance: a drop/impact and a fire. These two events are also applicable during sediment removal as material contained within the transfer pits could be affected by a fire event involving spilled diesel fuel or by a drop/impact event involving structural components or equipment falling into the pits.

Given the relatively simple processes involved in sediment removal, the release mechanisms that could potentially affect a significant amount of sediment in addition to these two events are readily identified. Removal of the sediment material from the transfer pits to a waste container staged in the transfer bay area generates the potential for a spill event to affect the sediment material. While a deflagration involving diesel fuel vapors may be credible under certain conditions (e.g., high temperature), such an event would be relatively low energy based on the limited amount of diesel fuel present and the low vapor pressure of diesel (about 1/100 that of gasoline). The sediment removal activities are scheduled to occur during the colder part of the year (October through March) when the public tours of the 105-B Reactor Building are not occurring, which reduces the amount of vaporization that could occur. Based on the above, the following three conditions were selected for quantitative evaluation for final hazard categorization determination: (1) a drop/impact event affecting the FSB sediment, (2) a fire event affecting the FSB sediment and wall contamination, and (3) a spill event involving the FSB sediment. These hazards are quantitatively evaluated in the next section.

2.2 HAZARD CATEGORIZATION

Hazard categorization was determined by comparing the bounding FSB inventory in Table 1-2 against the TQs in DOE (1997). The TQs in DOE (1997) are adjusted using the credible release fraction determined for each release mechanism prior to comparison. It is judged to be appropriate to use adjusted TQs because the sediment material, which is mostly contaminated iron oxides and silt, and the planned removal activities are very similar in form and circumstance, respectively, to that of contaminated soil remediation.

2.2.1 Drop/Impact Event Affecting FSB Sediment

The potential exists for sediment remaining in the 105-B Reactor Building FSB transfer pits (pits measure 193 by 274 cm [76 by 108 in.] each) to be impacted by falling building debris generated by a seismic or other external event. DOE (2000) does not evaluate an airborne release of radioactive material that can be directly correlated to the potential shock/impact to the sediment remaining within the transfer pits at about 6 m (20 ft) below grade.

The experiments evaluated in DOE (2000) to determine bounding ARF × RF values for shock/impact to powders involved kilogram quantities of relatively pure, dry, ceramic, or metallic oxides. The powders were contained in a small .95 L (1-qt) steel can without a lid that was resting on a plywood sheet. The sediment remaining in the 105-B Reactor Building transfer pits has an estimated volume and mass of about 17 m^3 (600 ft^3) and 2.6 $\times 10^4$ kg, respectively. (Note that this volume implies an average sediment depth of about 1.75 m [5.25 ft] in the two transfer pits based on the previously stated dimensions.)

The large volume and mass of sediment (in comparison to that used in the experiments) would provide significant self-shielding from impact stresses (i.e., impact stress would only affect a limited depth of sediment). Also, the sediment is contained in the transfer pits which are located about 6 to 7 m (20 to 25 ft) below-grade. Finally, as discussed in Section 1.4.2, the respirable fraction of the sediment material has been determined to be about 5%. Based on these identified differences, it is reasonably conservative to reduce the ARF × RF value of 1×10^{-4} given in Section 4.4.3.3.2 of DOE (2000) to a value of 5×10^{-5} for this event. This reduction accounts for the smaller respirable fraction of 5% for the sediment material compared to the value of 10% used in Section 4.4.3.3.2 of DOE (2000).

2.2.2 Fire Event Affecting FSB Sediment and Contamination

The above-grade portion of the 105-B Reactor Building FSB is constructed of concrete block walls and a precast concrete panel roof. The below-grade portion is constructed of reinforced concrete (DOE 1989). Although the FSB is constructed of non-combustible materials, wood planking is present and the asphalt fixative used on the FSB walls and floor is combustible. Originally, the wood planking was fire resistant; however, it is assumed that the fire resistance has been degraded due to age. Ignition of the wood planking during sediment removal activities is unlikely to occur because a wall separates most of the FSB storage area from the transfer bay area where the removal activities will occur (Figure 1-3). However, an opening in the wall between the two areas could allow propagation of a fire from the transfer bay area to the storage area. Ignition of the wood planking could result in some burning wood falling into the FSB itself and subsequent heating of the surface contamination underneath the asphalt fixative.

A spill of diesel fuel from the waste container transportation truck into the transfer bay area and subsequent ignition of the resultant pool could occur. The spilled diesel fuel would be prevented from entering the transfer pits by the concrete walls of the transfer pits that extend several feet above the floor level of the transfer bay area. The pool fire could realistically affect a single loaded waste container in the transfer bay area since the project plans to only load a single container at a time and to remove the loaded container before an empty container is lowered into the transfer bay area; however, the entire inventory was considered for the purpose of hazard categorization.

When the asphalt fixative was applied, the activity remaining in the FSB was bound to concrete surfaces. For the purposes of this analysis, it is assumed that the activity is present as a layer

of chemically non-reactive powder. It is also assumed that the sediment material is a chemically non-reactive powder material. This is judged to be conservative because a significant fraction of the material has migrated into the concrete and is not present as surface contamination, and the large volume and mass of sediment in a waste container (in comparison to that used in the experiments to develop bounding ARF × RF values) would provide significant self-shielding from thermal stresses. The ARF × RF value of 6×10^{-5} given in Section 4.4.1.1 of DOE (2000) for the thermal stress of chemically non-reactive compounds is judged to be reasonably conservative for this event.

2.2.3 Spill Event Affecting FSB Sediment

As discussed in Section 1.3, most of the sediment will be removed from the transfer pits using a flexible mechanical auger system. It is planned to use water to moisten the sediment before and during handling operations to decrease the potential for airborne material. The auger will be used to move sediment material from the bottom of the transfer pits to a waste container (i.e., ERDF container) staged in the transfer bay (a.k.a., rail car) area of the FSB. A flexible auger system typically consists of a rotating auger that is fully enclosed within a plastic or metal tube (i.e., housing). The systems are typically powered by small (≤ 3 hp) electric motors. The conveyed material is discharged to the receiving container (i.e., ERDF container) through a discharge head.

The auger will convey sediment from the bottom of the transfer pits into the waste container resting on the floor of the transfer bay area, which represents a total vertical distance of about 9.1 m (30 ft). Conveyance of the sediment material to a waste container is, in effect, a controlled spill. The potential fall distance of the sediment material must be evaluated in order to determine the appropriate ARF × RF value for this event since DOE (2000) has unique values for <3 m (9.8 ft) and >3 m (9.8 ft) fall distances. As discussed in the previous paragraph, the auger will be fully enclosed within a plastic or metal housing. A significant spill of sediment during conveyance, other than as designed through the discharge head, would require failure of the housing that does not affect the auger or drive system (motor). An event that could fail the housing while allowing the remainder of the auger system to remain functional is not postulated as a credible event during sediment removal activities. An impact event (i.e., falling debris in a seismic event, heavy equipment impact) or a fire event could fail the housing. However, these events would also cause the entire auger system to become non-functional. Therefore, it is believed that the credible fall distance for any significant amount of sediment does not exceed 3 m (9.8 ft).

Bounding ARF and RF values of 2×10^{-3} and 3×10^{-1} are given in Section 4.4.3.1.3 of DOE (2000) for the free-fall spill of powders from distances <3 m (9.8 ft). According to Section 4.4.3.1.2 of DOE (2000), the respirable fraction of the source material used in the experiments providing the basis for these values was greater than 95%. Based on the sediment respirable fraction information provided in Section 1.4.2, the RF value is reduced to 5×10^{-2} for a spill event. Therefore, the ARF × RF value is determined to be 1×10^{-4}.

The respirable release fractions for all three events are shown in Table 2-1.

Table 2-1. Respirable Release Fractions.

Accident	Airborne Release Fraction	Respirable Fraction	Total
Drop/Impact event affecting FSB sediment	1×10^{-3}	5×10^{-2}	5×10^{-5}
Fire event affecting FSB sediment	6×10^{-3}	1×10^{-2}	6×10^{-5}
Spill event affecting FSB sediment	2×10^{-3}	5×10^{-2}	1×10^{-4}

FSB = fuel storage basin

2.2.4 Final Hazard Categorization

The final hazard category for the sediment removal activities may be determined by using the respirable release fractions in Table 2-1 to revise the hazard category 3 TQs in DOE (1997) and then comparing the FSB inventory given in Table 1-2 to the revised TQs. Examination of the values in Table 2-1 shows that the spill event is the most limiting because it has the largest respirable release fraction value. The revised TQ values based on a respirable release fraction of 1×10^{-4} are shown in Table 2-2. Comparison of the total inventory (taken from Table 1-2) to the revised TQs shown in Table 2-2 to determine the total sum-of-the-fractions is provided in Table 2-3. The total sum-of-the-fractions value is determined to be 3.7×10^{-1}, which indicates that the sediment removal activity is below hazard category 3.

Table 2-2. Category 3 TQ Values Revised for Appropriate Release Value.

Isotope	R_{EPA}[a]	RV_{HA}	Food[b] Ingestion RV (Ci)	Adjusted[c] Food Ingestion RV (Ci)	Water[d] Ingestion RV (Ci)	Inhalation[e] RV (Ci)	Adjusted[c] Inhalation RV (Ci)	Direct[f] Exposure RV (Ci)	$TQ_{REVISED}$[g] (Ci)
C-14	5.E-01	1.E-04	--	--	1.5E+02	2.1E+01	1.0E+05	--	3.0E+03
Co-60[h]	1.E-03	1.E-04	6.0E+01	6.0E+02	v. lg.	1.6E+02	1.6.E+03	1.5E+01	2.8E+02
Ni-63	1.E-02	1.E-04	2.7E+02	2.7E+04	v. lg.	1.0E+03	1.0E+05	--	5.4E+05
Sr-90	1.E-02	1.E-04	8.2E-01	8.2E+01	v. lg.	2.1E+00	2.1E+02	--	1.6E+03
Tc-99	1.E-02	1.E-04	8.9E+01	8.9E+03	2.9E+02	3.6E+02	3.6E+04	--	5.8E+03
Cs-137	1.E-02	1.E-04	3.0E+00	3.0E+02	v. lg.	1.0E+02	1.0E+04	6.5E+01	1.3E+03
Eu-152	1.E-02	1.E-04	2.4E+01	2.4E+03	v. lg.	1.0E+01	1.0E+03	3.5E+01	7.0E+02
Eu-154	1.E-02	1.E-04	1.5E+01	1.5E+03	v. lg.	1.0E+01	1.0E+03	4.2E+01	8.4E+02
Th-232	1.E-03	1.E-04	2.1E-01	2.1E+00	v. lg.	5.2E-03	5.2E-02	7.3E+05	1.0E+00
U-233	1.E-03	1.E-04	3.0E+00	3.0E+01	v. lg.	2.1E-01	2.1E+00	1.9E+06	4.2E+01
U-234	1.E-03	1.E-04	3.0E+00	3.0E+01	v. lg.	2.1E-01	2.1E+00	7.5E+05	4.2E+01
U-235	1.E-03	1.E-04	3.0E+00	3.0E+01	v. lg.	2.1E-01	2.1E+00	2.7E+02	4.2E+01
U-236	1.E-03	1.E-04	3.0E+00	3.0E+01	v. lg.	2.1E-01	2.1E+00	1.6E+06	4.2E+01
U-238	1.E-03	1.E-04	3.0E+00	3.0E+01	v. lg.	2.1E-01	2.1E+00	--	4.2E+01
Pu-239/240	1.E-03	1.E-04	1.8E+00	1.8E+01	v. lg.	2.6E-02	2.6E-01	1.7E+06	5.2E+00
Am-241	1.E-03	1.E-04	3.0E-01	3.0E+00	v. lg.	2.6E-02	2.6E-01	--	5.2E+00

Notes: v. lg. Indicates that the sorption coefficient is greater than zero and the release value is much greater than that for other pathways (EPA 1989).—indicates that no gamma rays are emitted or the gamma rays which are emitted have gamma ray energies of less than 0.07 MeV and are strongly attenuated in air. No release value for the direct exposure pathway was calculated (EPA 1989).

[a] As reported in Appendix A of "Technical Background Document to Support Final Rulemaking Pursuant to Section 102 of the Comprehensive Environmental Response, Compensation, and Liability Act: Radionuclides", EPA Contract 68-03-3452, 02/89

[b] A release of RV to atmosphere produces a dose of 0.5 rem via the food ingestion pathway. Assumes deposition on crops 30 m (98 ft) from the point of release. Dispersion based on extrapolation of ground level data for stability class D and 1 m/sec windspeed (X/Q = 0.072 m^3/sec).

[c] Food ingestion and inhalation RVs adjusted by multiplying each original value by ratio of (R_{EPA} / RV_{HA}). See note 7 below.

[d] A release of RV to groundwater produces a dose of 0.5 rem via the water ingestion pathway. Assumes a well 30 m (98 ft) from the point of release. Contact time = 9 days. Independent of the airborne release fraction.

[e] A release of RV to atmosphere produces a dose of 0.5 rem via the inhalation pathway. Assumes a receptor 30 m (98 ft) from the point of release. Dispersion based on extrapolation of ground level data for stability class D and 1 m/sec windspeed (X/Q = 0.072 m^3/sec) and average breathing rate (2.7E-4 m^3/sec).

[f] A point source of RV produces a dose of 0.5 rem at 30 m (98 ft) in 24 hours. Independent of airborne release fraction.

[g] TQ = 20 x the minimum value of {(Food RV x R_{EPA}/R_{HA}), Water RV, (Inhalation RV x R_{EPA}/R_{HA}), or Direct Dose RV}. The value "20" results from the EPA RVs being based on an effective dose of 0.5 rem and the 1027 values being based on an effective dose of 10 rem (i.e., [0.5 rem x 20 = 10 rem]).

[h] The most restrictive value from EPA (1989) is direct exposure. 20 times this value is 300 Ci. The TQ listed in DOE 1997 is 280 Ci. The more restrictive value of 280 Ci is used.

EPA = U.S. Environmental Protection Agency
RV = release value
TQ = threshold quantity

Table 2-3. Sum of Fractions Using Revised TQ Values.

Isotope	Total Inventory, Ci	Revised TQ, Ci	Sum of Fractions
C-14	6.73E-03	3.0E+03	2.2E-06
Co-60	1.54E-01	2.8E+02	5.5E-04
Ni-63	7.01E+01	5.4E+05	1.3E-04
Sr-90	1.03E+01	1.6E+03	6.4E-03
Tc-99	1.09E-02	5.8E+03	1.9E-06
Cs-137	1.75E+01	1.3E+03	1.3E-02
Eu-152	8.22E-01	7.0E+02	1.2E-03
Eu-154	1.14E-01	8.4E+02	1.4E-04
Th-232	2.33E-05	1.0E+00	2.3E-05
U-233	5.62E-03	4.2E+01	1.3E-04
U-234	1.44E-02	4.2E+01	3.4E-04
U-235	6.36E-04	4.2E+01	1.5E-05
U-236	3.58E-04	4.2E+01	8.5E-06
U-238	1.54E-02	4.2E+01	3.7E-04
Pu-239/240	1.19E+00	5.2E+00	2.3E-01
Am-241	6.12E-01	5.2E+00	1.2E-01
		Total Sum of Fractions	**3.7E-01**

TQ = threshold quantity

3.0 HAZARD CONTROLS

3.1 SAFETY MANAGEMENT PROGRAMS

WCH safety management programs are routinely relied upon to ensure quality, worker safety, and environmental protection. The following sections discuss the WCH safety management programs.

3.2 CONDUCT OF OPERATIONS

Conduct of Operations is imposed to ensure that work is performed in a controlled and organized manner, and that the necessary documentation is maintained. The *Washington Closure Hanford Conduct of Operations Implementation Matrix* (WCH 2007) shows how Conduct of Operations is implemented.

Conduct of Operations strongly emphasizes technical competency, workplace discipline, and personal accountability to ensure a high level of performance during all activities. If conflict arises between instructions or directions, work shall be safely stopped until resolution is achieved. Safety is the first priority, and all planning shall include appropriate safety analyses to identify potential safety and health hazards and the methods needed to appropriately control these hazards. Workers will not start work until approved safety procedures, instructions, and directions are provided for non-routine operations.

Conduct of Operations requires workers to be alert and aware of conditions affecting the job site. Similarly, operators and workers shall notify the chain of command of any unexpected situations.

3.3 INTEGRATED WORK CONTROL PROGRAM

Worksite analysis is used to provide the means to successfully identify and control hazards in the workplace. This requires that a hazard analysis be performed for all new processes, material, or equipment before use. The Integrated Work Control Program also ensures that surveys, reviews, self-inspections, and/or regular examinations of processes and individual jobs are routinely conducted. Programmatic worksite controls include, but are not limited to, the following:

- Activities will be conducted in accordance with the *Site-Specific Health and Safety Plan (SSHASP) 100 Area S&M Project* (WCH 2006) and the WCH Radiological Control Program.
- The Radiological Control (RadCon) group completes comprehensive surveys and trends the results of monitoring activities in areas where employees work.
- Industrial hygiene technicians perform personal and area exposure sampling and monitoring.
- An integrated team with representation from affected craft(s), Field Engineering, Safety, Industrial Hygiene, RadCon, and other professionals conduct a job hazard analysis (JHA) of work in the planning stage. This is to ensure that all hazards are identified and that the

appropriate work methods are used to further reduce the risk to workers and the environment.

- Craft and supervision discuss the activities and ensure that the proper controls are in place prior to commencing work. If problems are found with the work control documents, the package is revised.

- Prior to start of work, a pre-job meeting is held to discuss those portions of the work control documents that will be performed during the work. Craft and supervisors carefully plan work in accordance with the work package, SSHASP, radiological work permit (RWP), and other work control documents.

Site-specific work controls will include the following elements:

- The JHA and RWP for specific work activities will identify the hazards and necessary controls for protection of the workers.

- Only trained equipment operators will be used.

- Containment will be provided, as appropriate.

- Dust-suppression techniques (e.g., fixatives) will be applied, as necessary, to control fugitive dust.

- Fixatives may be applied for contamination control.

- Field safety oversight personnel (Industrial Hygiene and RadCon safety specialists) will be provided during field activities.

3.4 RADIOLOGICAL PROTECTION PROGRAM

The radiological controls and protection program is defined in DOE-approved programs and WCH-approved procedures (RC-1, *Radiation Protection Procedures*). This program implements WCH policy to maintain radiological exposures to levels that are as low as reasonably achievable (ALARA) and to ensure adequate protection of workers. The WCH Radiological Protection Program meets the requirements of 10 CFR 835. Appropriate dosimetry, RWPs, personal protective equipment, ALARA planning, periodic surveys, and RadCon technical support will be provided.

Standard WCH controls for work in radiological areas are assessed as being adequate to control sediment removal activities. These controls support the planning that identifies the specific conditions and govern the specific requirements for an activity, periodic radiation and contamination surveys of the work area, radiological material handling, and periodic or continuous observation of the work by RadCon. The ALARA planning process will identify shielding requirements, contamination control requirements (including local ventilation controls), radiation monitoring requirements, and other RadCon requirements for the individual tasks conducted during 105-B Reactor Building FSB transfer pit sediment removal activities.

3.5 FIRE PROTECTION PROGRAM

The WCH Fire Protection Program (WCH procedures contained in SH-1, *Safety and Health*, Section 5, "Fire Protection") complies with the appropriate requirements of applicable CFR and National Fire Protection Association criteria, as well as the additional requirements of DOE Headquarters and the Richland Operations Office directives included in the WCH contract. The WCH Fire Protection Program was developed to the guidance of the *DOE Fire Protection Handbook* (DOE 1996).

Key Programmatic Element

- Combustible materials are controlled in accordance with Fire Protection Program procedural requirements.

3.6 OCCUPATIONAL SAFETY AND INDUSTRIAL HYGIENE

Activities will be conducted in accordance with the SSHASP (WCH 2006). The SSHASP addresses the health and safety hazards associated with sediment removal activities at the 105-B Reactor Building. In addition to the SSHASP, the JHA and RWP for specific work activities will identify the hazards and necessary controls for protection of the workers.

Before work begins, a pre-job briefing is held with the affected workers. This briefing will include reviews of the hazards that may be encountered and the associated requirements. The primary non-radiological hazards during sediment removal activities include the following:

- Asbestos
- Lead
- Polychlorinated biphenyls
- Biological (insect bites and snakes)
- Temperature extremes
- Working in close proximity to moving equipment
- Possible exposure to organic and inorganic chemicals
- Uneven working surfaces
- Electrical
- Beryllium
- Noise.

3.7 TRAINING

The experience and capabilities of the operating staff are extremely important in maintaining worker and environmental safety. Day-to-day knowledge of ongoing operations, month-to-month understanding of conditions encountered, and ongoing understanding of lessons learned are vital to continued safe operation.

Training requirements will ensure that personnel have been instructed in the technologies to work safely in and around radiological areas and to maintain their individual radiation exposure and the radiation exposures of others ALARA. Standardized core courses and training material

will be presented, and site specific information and technologies will be added to adequately train workers.

Radiological control technicians must complete and be current in qualification training. Radiological workers who are not radiological control technicians must meet the training (i.e., General Employee Radiological Training, RadWorker I, RadWorker II) requirements stipulated in applicable RadCon procedures. The level of qualification is based on areas to be entered and the types of activities performed. These training courses require the successful completion of examinations to demonstrate understanding of theoretical and classroom material.

Some sediment removal activities may require the acquisition of expert services, as opposed to project staff training. Subcontractors will be required to comply with the minimum requirements for performing work within the project worksite.

The WCH training program provides workers with the knowledge and skills necessary to safely execute assigned duties. A graded approach is used to ensure that workers receive a level of training commensurate with their responsibilities that complies with applicable requirements.

3.8 CONFIGURATION MANAGEMENT

Established configuration/change control processes ensure that proposed changes and discovered conditions are reviewed in relation to the authorized activities identified by this FHC. The only document expected to need configuration control for 105-B Reactor Building FSB sediment removal activities is this document. Configuration management will be performed in accordance with NS-1, *Nuclear Safety Manual*.

NS-1, *Nuclear Safety Manual*, defines the maintenance of configuration of hazard categorization documents in procedure NS-1-2.1, "Hazard Categorization." This procedure requires that hazard categorizations be maintained for facilities that have an FHC of below hazard category 3. The FHC evaluation process is used to evaluate proposed changes and discoveries for facilities and activities to determine if the changes are within the authorization basis.

The WCH work control procedures contain requirements regarding the incorporation of those controls resulting from the hazard analysis that are relied upon to ensure the hazard categorization remains valid into the work packages. This process ensures that these controls are appropriately implemented into the work control documents.

In the event that a finding indicates a breach of these commitments, appropriate stabilization and/or recovery actions will be identified and implemented as appropriate. SEM-3, *Incident Response and Investigation*, SEM-3-1.2, "Occurrence Categorization and Reporting," describes the reporting process and protocol applicable to such a discovery.

3.9 CRITICALITY SAFETY

The Criticality Safety Program is described in NS-1-1.1, "WCH Criticality Safety Program." The WCH Criticality Safety Program applies to fissionable materials that are discovered, held, produced, processed, stored, transferred, transported, disposed, or otherwise handled by WCH. In accordance with the Criticality Safety Program, a criticality safety review (WCH 2010c) was

prepared and determined that the fissionable material inventory for the reactor transfer pits does not exceed the subcritical mass limit based on the total inventory identified. Criticality Safety Requirements are not needed, since there are no normal or any credible abnormal conditions that could lead to criticality.

3.10 EMERGENCY PREPAREDNESS

The WCH Emergency Management Program (including preparedness, planning, and response) is described in detail in SEM-2, *Emergency Management Program*, and contains the administrative responsibilities for compliance with the *Hanford Emergency Management Plan* (DOE-RL 1999). The 100 Areas will be managed under the Operational Emergency Base Program.

All emergency planning and preparedness activities for the 105-B Reactor Building will be consistent with planning and preparedness actions undertaken by other Hanford Site contractors and similar projects. Activities will be in a manner that ensures the health and safety of workers and the public and the protection of the environment in the event of an abnormal incident or emergency.

Project response to any emergencies (i.e., project or neighboring project incident) will be to evacuate personnel to a safe location and initiate the required responsibilities of the building warden and other project personnel in support of the Incident Command System.

The WCH Emergency Management Program is based on a graded approach and is commensurate with the hazards and consequences associated with the projects/facilities and activities managed by WCH (involving radioactive and non-radioactive hazardous materials) and/or neighboring facilities.

3.11 FHC PRESERVATION CONTROLS

There are no preservation controls identified, as necessary, to maintain the assumptions of the final hazard categorization. Authorized activities are discussed in Section 1.3. The FHC evaluation process (as discussed in Section 3.8) will be used to evaluate proposed changes and discoveries to determine if the changes are within the safety basis.

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APPENDIX A

HAZARD IDENTIFICATION TABLE

APPENDIX A

HAZARD IDENTIFICATION TABLE

Table A-1 has six columns; the column headings and content are described as follows:

- **Column 1 – Hazard Type.** This column identifies the following types of hazards investigated: radiological (including radioactive material and direct radiation), fissile material, toxic hazards, carcinogenic hazards, biohazards, asphyxiates, flammable/ combustible material, reactive material, explosive material, electrical energy, thermal energy, kinetic energy, noise, seismic, and high wind and water intrusion.

- **Column 2 – Location.** This column identifies the location where these activities are to be performed.

- **Column 3 – Form.** This column specifies the form of the hazard type. This column is not intended to provide a detailed identification of the chemical (e.g., oxide) or physical form of the hazard type (e.g., crystalline). Such detail is not considered at the hazard identification stage of a safety analysis.

- **Column 4 – Quantity.** This column quantifies the hazard. Measured values are presented when relevant and available.

- **Column 5 – Remarks.** This column presents information that provides a better understanding of the hazard type, location, form, and quantity.

- **Column 6 – References.** This column lists the information sources used to identify the location, form, and quantity of a given hazard type.

Table A-1. Hazard Identification Table for Sediment Removal. (5 Pages)

Hazard Type	Location	Form	Quantity	Remarks	References
Radioactive Material	105-B FSB/transfer pit	Activation, fission products, and corrosion products.		Asphalt emulsion fixative in FSB, sludge has been moved into the transfer pit and covered with a layer of sand (this can't be visually confirmed but is believed to be the case) and a plywood cap.	UNI-3958, UNI-3714, Rev. 1 (UNC 1986, 1987)
Direct Radiation	105-B FSB/transfer pit	Activation, fission, and corrosion products.		Asphalt emulsion fixative in FSB, sludge has been moved into the transfer pit and covered with a layer of sand (this can't be visually confirmed but is believed to be the case) and a plywood cap. Measured dose rates from sediment are about 25 mrem/hr.	UNI-3714, Rev. 1 (UNC 1987)
Fissionable Material	105-B FSB/transfer pit	Pu-238, Pu-239, and Am-241 present in the FSB/transfer pits.	WCH 2010 shows criticality evaluation for the 105-B FSB/transfer pits and includes the fissionable quantities. As of 3/1/98: 0.0677 Ci of Pu-238 1.6 Ci of Pu-239 0.49 Ci of Am-241	Criticality evaluation concluded that no criticality controls or requirements were required.	UNI-3714, Rev. 1, and 0100B-CE-N0014, Rev. 0 (UNC 1987, WCH 2010)

Table A-1. Hazard Identification Table for Sediment Removal. (5 Pages)

Hazard Type	Location	Form	Quantity	Remarks	References
Hazardous Material (e.g., toxic, carcinogenic)	All areas	Asbestos in the form of piping insulation, transite wall board, ventilation components, and insulation (friable if degraded or damaged).	Unknown quantities.	Ongoing abatement program in place to remove damaged asbestos from facility. Majority of asbestos is in nonfriable form. In reactor facilities that have undergone ISS, most of the asbestos was removed with the exception of asbestos in inaccessible areas.	Field walkdowns
	All areas	Lead shielding in the form of lead shot, brick, and cast forms; also lead-based paint.	Quantities vary in each facility.	Ongoing abatement program in place to remove accessible and removable lead from facility. Majority of lead is in nondispersible form. Greater oxidation rates of lead occur than anticipated based on arid climatic conditions, resulting in dispersible lead oxide. Reactors that have undergone ISS have all accessible and hazardous materials removed from the facility.	Field walkdowns
	All areas	Mercury contained in switches and instruments.	Unknown quantities.	Not readily dispersible; not significantly impacted by S&M activities.	Historical knowledge
	All areas	PCBs contained in electrical and lighting equipment.	Unknown quantity.	Not readily dispersible; not significantly impacted by S&M activities.	Historical knowledge
	All areas	Miscellaneous.	Negligible.	Miscellaneous chemicals remaining have been removed during past S&M cleanup campaigns. Potential exists to discover old containers of chemicals.	Staff interview
	105-B Reactor Building (valve pit)	Odors from spills that occurred during operation of the facility. The odor is being emitted from the acid brick floor and grout around the bricks. Lime has been placed around the area to neutralize the acid residue.	Unknown quantity.	Area emitting strong odor. Odor is not coming from surrounding tanks in the valve pit area.	BHI-00221 (BHI 1994) Personnel interview

Table A-1. Hazard Identification Table for Sediment Removal. (5 Pages)

Hazard Type	Location	Form	Quantity	Remarks	References
Biohazard	All areas	Rodents, insects, snakes; bird and animal feces.	Greater activity than normally occupied facilities.	Because there is very little human activity in and around most parts of the reactor facilities, increased rodent, insect, and snake activity can be expected. Animal feces are considered a health hazard and may be radioactively contaminated.	Historical knowledge
Asphyxiant/ Confined Spaces	All areas	Unventilated areas of the reactor buildings.	Not quantified.	Unventilated below-grade areas of structures pose potential hazards. Numerous confined spaces under concrete cover blocks; in accessible water basins; and in drains, manways, and valve pits around the facilities. ISS activities left the remaining structure unventilated.	Historical knowledge
Flammable/ Combustible Material	Reactor block	Graphite blocks.	Varies with each reactor block.	While graphite can burn, there is no ignition source of sufficient energy present in the facility to ignite the graphite.	Staff interview
	105-B FSB/transfer pit	Wooden planking over basin.	29.8 m^3 (39 yd^3).	The wooden planking is fire resistant; however, the ability to resist combustion will have decreased due to the age.	BHI-00066, Rev. 4, and WHC-SP-0331, Rev. 1 (BHI 1997, WHC 1994)

Table A-1. Hazard Identification Table for Sediment Removal. (5 Pages)

Hazard Type	Location	Form	Quantity	Remarks	References
Flammable/ Combustible Material (continued)	105-B FSB	Hydraulic fluid in battery-powered scissor lift and in waste container transport truck.	About 1 to 2 gallons in the scissor lift and about 40 to 50 gallons in the transport truck.	Planned for scissor lift to be located on floor of FSB to facilitate personnel movement between deck and floor level. Waste container transport truck to remain outside of structure.	
Reactive Material	All areas	Residual process and deactivation chemicals.	Residual quantities.	Potential exists to discover old containers of chemicals that, if mixed, could generate heat/gas.	Historical knowledge
Explosive Material	All areas	Hydrogen gas.	Unknown quantity of lead-acid batteries used for backup power for building systems. Most of these systems resided exterior to the shield walls; therefore, reactor facilities that have undergone ISS would have removed this hazard.	Hydrogen gas can be generated during charging of batteries.	Historical knowledge
	All areas	Miscellaneous chemicals.	Negligible.	Miscellaneous chemicals remaining have been removed during S&M cleanup campaigns. Potential exists to discover old containers of chemicals.	Staff interview
Electrical Energy	All areas	120/240/480 V.	Not applicable.	General industrial hazard.	Historical knowledge
Thermal Energy	All areas	Portable space heaters.	Used as needed	None outside that routinely encountered in industry.	
Kinetic Energy	All areas	Structural components.	Not applicable.	Structure occupied occasionally during S&M activities and for tours (105-B); roof structures in fair-to-good condition. ISS structures will be occupied once every 5 years as part of preventive maintenance walkdowns.	BHI-00066, Rev. 4 (BHI 1997)
	All areas	Aircraft crash.	Not applicable.	Probability of event is extremely low based on the size of the 105-B Reactor Building structure.	WHC-SD-CP-SAR-021 (WHC 1996)

Table A-1. Hazard Identification Table for Sediment Removal. (5 Pages)

Hazard Type	Location	Form	Quantity	Remarks	References
Kinetic Energy (continued)	All areas	Vehicle crash.	Not applicable.	Vehicles are routinely used around the reactor facilities.	Historical knowledge
High Pressure	All areas	Not applicable.	None.	Available references provided no information regarding high pressure; however, based on the current phase of the building, it is anticipated that there is no high pressure.	Historical knowledge and facility walkdowns
Natural Phenomena	All areas	Earthquake of sufficient magnitude to result in failure of structural components.	Not applicable.	The potential exists for an earthquake to induce the failure of building structural members, which then can interact with the radioactive and hazardous material inventory.	None
	All areas	Flood of sufficient magnitude to result in structural intrusion.	Not applicable.	The potential exists for water to interact with radioactive and hazardous materials within the building as a result of flooding.	None
	All areas	Volcanic activity produces sufficient ash fall to result in the failure of structural members.	Not quantified.	Ash fall has the potential to result in structural members interacting with radioactive and hazardous materials.	None
	All areas	Precipitation infiltrates the building structure. Precipitation in the form of snow results in failure of structural members.	Not quantified.	Precipitation could infiltrate the structure and interact with radioactive and hazardous materials. Snow has the potential to result in structural members interacting with radioactive and hazardous materials.	None
	All areas	Lightning strike on the building structure.	Not applicable.	Lightning could potentially strike the building and interact with the radioactive and hazardous material inside.	None
	All areas	High winds of sufficient magnitude to result in the failure of structural members.	Not quantified.	High winds could potentially induce the failure of structural members and a resultant interaction with radioactive and hazardous materials.	None

FSB = fuel storage basin
ISS = Interim Safe Storage
PCB = polychlorinated biphenyl
S&M = surveillance and maintenance

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