AIIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910

301/587-8202

Centimeter

Inches

MANUFACTURED TO AIIM STANDARDS
BY APPLIED IMAGE, INC.

Application for Approval of Modification for the 105-KE Basin Encapsulation Activity

Date Published
March 1993

Prepared for the U.S. Department of Energy

United States
Department of Energy

P.O. Box 550
Richland, Washington 99352

Approved for Public Release

MASTER

dp
DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

LEGAL DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, nor any of their contractors, subcontractors or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or any third party's use or the results of such use of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof or its contractors or subcontractors. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

This report has been reproduced from the best available copy.
Available in paper copy and microfiche.

Available to the U.S. Department of Energy
and its contractors from
Office of Scientific and Technical Information
P.O. Box 62
Oak Ridge, TN 37831
(615) 576-8401

Available to the public from the U.S. Department of Commerce
National Technical Information Service
5285 Port Royal Road
Springfield, VA 22161
(703) 487-4650

Printed in the United States of America

DISCLM-1.CHP (1-91)

EXECUTIVE SUMMARY

This is an Application for Approval of Modification (hereafter referred to as the Application) for the 105-KE Basin Encapsulation Activity. This Application is being submitted to the U. S. Environmental Protection Agency pursuant to the Title 40, *Code of Federal Regulations*, Part 61.07, "Application for Approval of Construction and Modification," as amended. The encapsulation activity will consist of all of the activities described in Section 1.3 as necessary to complete encapsulation of the fuel elements and sludge contained in the 105-KE Basin.

The 105-KE Basin is a storage basin for irradiated N Reactor fuel in 100-K Area of the Hanford Site. The 105-KE Basin currently stores 1,150 MTU (metric tonnes uranium) of N Reactor irradiated fuel elements transferred to the basin from 1975 through 1989. The 105-KE Basin also stores sludge and debris, which has been generated during the storage of all irradiated nuclear fuel, that has passed through the basin since 1975. Because the ultimate disposition of the fuel will be determined by an Environmental Impact Statement and the resulting Record of Decision will likely have a significant implementation period, the 105-KE Basin will continue to store the fuel elements for an estimated additional 20 years.

The encapsulation activity will be carried out on the 105-KE Basin floor under a minimum of 3 m (10 ft) of water. The encapsulation activity includes emptying irradiated fuel elements from the existing canisters stored in the 105-KE Basin, packaging these fuel elements in new canisters, packaging sludge from previous fuel inventories and the current inventory in new canisters,

capping these canisters of fuel elements and sludge, and returning the new canisters to the storage racks in the basin. The empty canisters will be cleaned and crushed underwater. Packaging and disposal of the crushed canisters will take place above the water.

This Application is being submitted to the U. S. Environmental Protection Agency in accordance with verbal guidance provided in a February 18, 1993 conversation between representatives of the Agency and Westinghouse Hanford Company. This Application is a copy of the Notice of Construction submitted to the State of Washington Department of Health, pursuant to *Washington Administrative Code*, 246-247, "Radiation Protection -- Air Emissions," modified to identify the Notice of Construction as meeting the requirements of Title 40 *Code of Federal Regulations*, Part 61.07, "Application for Approval of Construction and Modification," as amended.

The Application presents the chemical and physical processes relating to the encapsulation activity, source term, expected annual emissions, radionuclide control and monitoring equipment, and projected dose to the maximally exposed individual.

CONTENTS

1.0	APPLICANT	1-1
1.1	PURPOSE OF APPLICATION	1-1
1.2	LOCATION	1-1
1.3	DESCRIPTION OF THE CHEMICAL AND PHYSICAL PROCESSES	1-4
	1.3.1 105-KE Basin	1-4
	1.3.2 Encapsulation Activity	1-4
2.0	DESCRIBE THE SOURCE TERM	2-1
2.1	BACKGROUND	2-1
2.2	SOURCE TERM DESCRIPTION	2-1
	2.2.1 Fuel Elements	2-1
	2.2.2 Sludge	2-2
	2.2.3 Basin Water	2-3
3.0	PROVIDE DRAWINGS OF THE EMISSION UNIT	3-1
3.1	DESCRIBE THE RADIONUCLIDE CONTROL EQUIPMENT	3-1
3.2	RADIONUCLIDE CONTROL EFFICIENCIES	3-5
3.3	WATER TREATMENT SYSTEMS COMPONENT RADIONUCLIDE REMOVAL EFFICIENCIES	3-5
3.4	OPERATIONAL CONTROL	3-6
4.0	PROVIDE EXPECTED ANNUAL EMISSIONS	4-1
4.1	PROJECTED EMISSIONS BASED ON APPENDIX D	4-1
4.2	PROJECTED EMISSIONS BASED ON GOOD ENGINEERING JUDGMENT	4-1
5.0	DESCRIBE THE MONITORING EQUIPMENT	5-1
6.0	PROJECTED DOSE TO THE MAXIMALLY EXPOSED INDIVIDUAL	6-1
6.1	PROJECTED DOSE TO THE MAXIMALLY EXPOSED INDIVIDUAL USING 40 CFR 61, APPENDIX D, PROJECTED EMISSIONS AND CAP 88	6-1
6.2	PROJECTED DOSE TO THE MAXIMALLY EXPOSED INDIVIDUAL USING GOOD ENGINEERING JUDGMENT PROJECTED EMISSIONS AND CAP 88	6-1
6.3	PROJECTED DOSE TO THE MAXIMALLY EXPOSED INDIVIDUAL USING GOOD ENGINEERING JUDGMENT PROJECTED EMISSIONS AND GENII	6-1
7.0	BEST AVAILABLE RADIONUCLIDE CONTROL TECHNOLOGY ASSESSMENT	7-1
8.0	REFERENCES	8-1

LIST OF FIGURES

1-1	Regional Location of the 100-K Area	1-2
1-2	Location of the 105-KE Basin in the 100-K Area	1-3
1-3	Diagram of the 105-KE Basin	1-5
1-4	Diagram of the Discharge Pickup Chute Area	1-6
1-5	Hoist and Monorail System	1-7
1-6	Process Flow Diagram	1-9
1-7	Canisters in the 105-KE and 105-KW Basins.	1-10
3-1	Location of Roof Vents and Fixed Head Samplers in the 105-KE Basin	3-2
3-2	Schematic Arrangements of the Water Treatment Systems	3-3

LIST OF TABLES

2-1 ^a	Calculated 105-KE Basin Fuel Element and Sludge Activity	2-4
3-1	Average Radionuclide Removal Efficiencies of Components of the Water Treatment Systems at 105-KE Basin	3-6
4-1 ^a	Projected Annual Airborne Emissions Using 40 CFR 61, Appendix D	4-2
4-2 ^a	Projected Annual Airborne Emissions Using Fixed Head Sampler (RVII) Data	4-3
5-1	Particulate Radionuclides Contributing 10 Percent or More of the Committed Effective Dose Equivalent	5-2
6-1 ^a	Committed Effective Dose Equivalent to the Maximally Exposed Individual Using Projected Emissions Based on 40 CFR 61, Appendix D, Methodology (Table 4-1) and CAP 88 Dose Conversion Factors	6-2
6-2 ^a	Committed Effective Dose Equivalent to the Maximally Exposed Individual Using Projected Emissions Based on the Fixed Head Sampler (Table 4-2) and CAP 88 Dose Conversion Factors	6-3
6-3 ^a	Committed Effective Dose Equivalent to the Maximally Exposed Individual Using Projected Emissions Based on the Fixed Head Sampler (Table 4-2) and GENII Dose Conversion Factors	6-4

LIST OF TERMS

BARCT	best available radionuclide control technology
CEDE	committed effective dose equivalent
EPA	U.S. Environmental Protection Agency
IXM	Ion exchange modules
MEI	maximally exposed individual

This page intentionally left blank.

1.0 APPLICANT

Facility Owner: U.S. Department of Energy
Richland Field Office
P.O. Box 550
Richland, Washington 99352

1.1 PURPOSE OF APPLICATION

This Application for Approval of Modification (hereafter referred to as the Application) for the encapsulation of irradiated reactor fuel stored in the 105-KE Basin is being submitted to the U. S. Environmental Protection Agency (EPA) pursuant to Title 40 *Code of Federal Regulations*, Part 61.07, "Application for Approval of Construction and Modification," as amended.

The encapsulation activity will be carried out on the 105-KE Basin floor under a minimum of 3 m (10 ft) of water. The encapsulation activity will include emptying irradiated fuel elements from the existing canisters stored in the 105-KE Basin, packaging these fuel elements in new canisters, packaging sludge from previous fuel inventories and the current inventory in new canisters, capping the canisters of fuel elements and sludge, cleaning and crushing the empty canisters, and returning the new canisters of fuel elements and sludge to the storage racks in the basin. Packaging and disposal of all the crushed canisters will take place above water.

Encapsulation of the 105-KE Basin fuel inventory will provide primary containment for the irradiated fuel, some of which has had its zirconium cladding breached, exposing the uranium fuel to the 105-KE Basin water. This will dramatically reduce environmental concerns associated with the 105-KE Basin. The encapsulation activity will minimize continued release of radionuclides from the damaged fuel elements to the basin water. The lower radionuclide releases to the basin water will reduce basin work place radionuclide concentrations and air emissions from the facility. Lower releases will also reduce worker exposure and minimize waste generated from operation of the water treatment systems. Because the 105-KE Basin is located in close proximity to the Columbia River, the encapsulation activity will greatly reduce environmental risks associated with a leak from the basin.

1.2 LOCATION

The 100-K Area is located on the Hanford Site approximately 25 mi (40 km) northwest of the city of Richland, Washington (Figure 1-1). The location of the 105-KE Basin in the 100-K Area is shown in Figure 1-2.

Figure 1-1. Regional Location of the 100-K Area.

BUILDING NUMBER	DESCRIPTION
100-B-W	REACTOR BUILDING
100-B-W	WATER RETENTION BASIN
100-B-W	GAS STORAGE
110-B-W	GAS RECIRCULATION BUILDING
110-B-W	REACTOR COOL
117-B-W	STRAIGHT AIR FILTER BUILDING
110-B-W	STRAIGHT AIR FILTER BUILDING
150-B-W	HEAT RECOVERY FACILITY
151-B	SOUTHING STATION
151-B-W	SUBSTATION BUILDING
150-B-W	POWER CONTROL BUILDING
160-B-W	ON SUMMER
167-B	CROCK POT AND TUMBLER BUILDING
161-B-W	SUVER PUMP STATION
162-B	EMERGENCY WATER PUMP HOUSE
163-B-W	FILTER PLANT
163-B-W	CHLORINE VAULT
163-B-W	HEAD HOUSE
160-B-W	MAIN PUMP HOUSE
170-B-E	BADGE HOUSE
170-B-EA	EXCLUSION AREA BADGE HOUSE
1702-B-W	BADGE HOUSE
171-B-W	WAREHOUSE
1716-B-W	ON STORAGE
1717-B	MAINTENANCE & TRANSPORTATION
1720-B	OFFICES & TELEPHONE EXCHANGE
1721-B	OFFICE TRAILER
1722-B	OFFICE TRAILER
166-B-E	REACTOR FACILITY
107-B	WATER RETENTION BASIN
110-B-E	GAS STORAGE
110-B-E	GAS RECIRCULATION BUILDING
110-B-E	REACTOR STACK
110-B-E	STRAIGHT AIR FILTER BUILDING
110-B-E	MAINTENANCE SHOP
150-B	HEAT RECOVERY FACILITY
163-B-E	POWER CONTROL BUILDING
166-B-E	ON HOUSE
166-B-E	ON SUMMER
161-B-E	SUVER PUMP STATION
163-B-E	CHLORINE VAULT
163-B-E	HEAD HOUSE
1702-B-E	BADGE HOUSE
1702-B	FILLMENT WATER TREATMENT PLANT PLANT
1702-B-E	WATER STORAGE RECIRCULATION BUILDING
1712-B	SHOP BUILDING
1712-B-E	ON-SUMMER
1702-B-E	ON & PUMP STORAGE
RADIOLOGICAL UNDERGROUND BITE INSTRUMENTS	
170-B	SOLID WASTE BURN GARAGE
160-B-1	EMERGENCY CRIB
160-B-2	SOLID WASTE BURN
160-B-3	ATTENTION BURN TANKS
160-B-W	ATTENTION BURN TANKS
160-B-2	115-BE CRIB
160-B-2	115-BE CRIB
160-B-3	1700 OUTFALL

1.3 DESCRIPTION OF THE CHEMICAL AND PHYSICAL PROCESSES

Describe the chemical and physical processes related to the emission unit.

The following paragraphs describe the 105-KE Basin and the encapsulation activities.

1.3.1 105-KE Basin

The 105-KE Basin is a rectangular (38-m [125-ft]-long, 20-m [67-ft]-wide, and 6-m [21-ft]-deep) reinforced concrete basin, and is housed within a portion of building 105-KE. The 105-KE Basin is divided into three bays by concrete partitions. The partitions are open at each end. The basin floor is covered with fuel storage racks that currently hold approximately 3,700 canisters of irradiated fuel elements. Each existing canister consists of two upright cylinders approximately 8-in. in diameter, made of either aluminum or stainless steel, and joined by trunions to facilitate handling. The existing canisters are open-top with closed- or wire-meshed bottoms. A canister can hold a maximum of 14 fuel elements.

The 105-KE Basin is filled with water to a nominal depth of 5 m (16 ft) to cool the fuel and to provide a radiological shield for the basin workers. The basin water is drawn from one end of each basin section and circulated through two closed water treatment systems. The water treatment systems are included as part of the radionuclide control equipment. Chapter 3.0 describes both of the water treatment systems in detail. One of the two water treatment systems is original which provides 2,500 kW of heat removal capability accomplished by a water-cooled chiller. The second water treatment system, a sand filter and ion exchange system, was added in 1985 to provide supplemental radionuclide removal capabilities. A diagram of the basin is shown in Figure 1-3.

Encapsulation of the fuel elements will occur in the discharge pickup chute, which is located directly behind the KE Reactor on the south side of the 105-KE Basin. The discharge pickup chute originally served as a fuel element discharge and packaging area for the reactor during operation. This discharge pickup chute has been converted to a work area for encapsulation activities (Figure 1-4). During installation of encapsulation equipment, sludge will be vacuum-pumped underwater from the discharge pickup chute to the Weasel Pit equipped with a screen partition (Figure 1-3). A grating suspended over the entire 105-KE Basin provides a working surface from which workers move fuel canisters underwater via a hoist and monorail system (Figure 1-5).

1.3.2 Encapsulation Activity

The encapsulation activity will be carried out in close proximity to the basin floor such that a minimum of 3 m (10 ft) of water shielding between the worker and the fuel is maintained. The existing hoist and the new

1-5

Figure 1-3. Diagram of the 105-KE Basin.

DOE/RL-93-14

Figure 1-4. Diagram of the Discharge Pickup Chute Area.

Notes: Sketch is not scale. Station locations are only approximate.

29101033.5

Figure 1-5. Hoist and Monorail System.

29101033.4

encapsulation equipment will be used to accomplish encapsulation of the fuel elements and sludge. The encapsulation activity will include:

- Emptying fuel elements from the existing canisters stored in the 105-KE Basin onto a dump table
- Filling new canisters with the fuel elements
- Relocating sludge to support encapsulation of the fuel elements
- Filling new canisters with sludge
- Capping the new canisters of fuel elements or sludge
- Cleaning and crushing empty canisters
- Returning the canisters of fuel elements or sludge to the storage racks in the basin
- Lifting the crushed canisters out of the water and packaging them for disposal.

1.3.2.1 Encapsulation of the Fuel Elements. The 105-KE Basin currently contains approximately 3,700 open-top canisters filled with N Reactor irradiated fuel elements and two baskets (the equivalent of five canisters) filled with irradiated aluminum-clad fuel assemblies from retired single-pass reactors. The open-top canisters, containing fuel elements, will be moved, one at a time, from the basin to the discharge pickup chute. The canisters are moved underwater using the existing hoist. The hoist operates on monorails suspended above the basin. The fuel elements will be emptied onto a dump table, manually tonged to the repackaging station, and hydraulically inserted into stainless steel canisters.

After the fuel elements are placed into the new stainless steel canisters the lids will be installed and corrosion inhibitor added. The canisters then will be purged with nitrogen. The lid of the stainless steel canisters incorporates a GRAFOIL¹ sealing surface and a mechanism that applies pressure to affect a lid-to-canister seal. The corrosion inhibitor consists of demineralized water and sodium nitrite. The corrosion inhibitor is designed to minimize fuel degradation rates within the canister. The nitrogen purge establishes an inert gas blanket at the top of canister. The nitrogen isolates the water within the canister from the basin water, preventing the escape of dissolved radionuclides. This, in effect, isolates the canister radionuclide inventory from the basin water.

The new canisters will then be moved back to the basin storage racks via the hoist and monorail system. A process flow diagram of emptying the fuel elements from the old canisters, repackaging the fuel elements into new canisters, and cleaning and crushing the old canisters is shown in Figure 1-6. Figure 1-7 shows the open-top canisters as they are currently stored in the 105-KE Basin and capped canisters, as they are stored in the 105-KW Basin.

¹Tradename of the Union Carbide Corporation.

Encapsulation

- 1 Fuel elements are emptied onto table.

Old canister

- 2 Each fuel element is manually transported to repackaging station using counter-flotation fuel tongs.

New Canister (Mark II)

- 3 Once 14 fuel elements are stacked (seven to each side) on the repackaging table, they are hydraulically inserted into the new canister.

- 4 The newly packaged Mark II canister is encapsulated by installing seals and lids. An inhibitor is injected, and the canister is purged with nitrogen.

Lids and seals

Gas trap

- 5 Canister is ready to be moved to its basin storage rack.

Emptied canister is cleaned and crushed. Crushed canister is removed from basin and packaged for disposal.

Figure 1-6. Process Flow Diagram.

Figure 1-7. Canisters in the 105-KE and 105-KW Basins.

irradiated in Reactor Fuel

105-KW Basin

105-KE Basin

The encapsulation of the fuel elements is estimated to take approximately 2 years. During periods of encapsulation, the work will be accomplished by operating two shifts per day, five days per week, and encapsulating eight canisters per shift.

1.3.2.2 Encapsulation of Sludge. Sludge and debris from basin operations has accumulated on the basin floor. The sludge resulted from corrosion of breached fuel elements, which consists of uranium, fission and activation products, along with dirt and sand introduced from the outside environment, and corrosion products of basin structural components. The debris consists of structural parts from fuel elements and miscellaneous tools and materials that were left behind. The sludge, which is often finely divided, is distributed loosely and unevenly on the basin floor. Activities performed near the floor of the basin cause the sludge to become resuspended near the activity. The sludge eventually settles to the basin floor within hours or days after the activity has ended.

Sludge deposited on the basin storage racks and canisters will likely be resuspended along the path of canisters being transported by the hoist and monorail system. Additional sludge suspension may occur as canisters with wire mesh bottoms, containing breached cladding, disperse some of their contents. The discharge pickup chute is expected to become clouded during encapsulation, and the basin in general will show an increase in turbidity. Water from the water treatment systems will be introduced into the discharge pickup chute to improve the visibility for operators performing the encapsulation work. Periodic underwater vacuum pumping and water sparging of the discharge pickup chute will be necessary to remove accumulated sludge that will settle to the bottom. The sludge will be vacuum pumped to the Weasel Pit, which is equipped with a fine mesh screen partition. This screen retains the majority of the sludge in the Weasel Pit and allows the liquid to return to the basin.

After the sludge is vacuum pumped, any solid debris greater than 0.64 cm (0.25-in.) in diameter remaining in the discharge pickup chute will be transferred to open-top canisters. These open-top canisters will be retrieved to the dump table and the contents sorted to segregate any fuel element fragments for encapsulation. The remaining debris will be packaged for disposal as low-level radioactive waste in accordance with WHC-EP-0063-3, "Hanford Site Solid Waste Acceptance Criteria" (WHC 1988a).

The sludge accumulated on the basin floor will be vacuum pumped to the Weasel Pit for consolidation. The sludge in the Weasel Pit will be encapsulated underwater. Encapsulation of sludge will employ operations such as (1) vacuum pumping sludge using a suction nozzle, (2) settling or centrifuging the pumped sludge for removing the suspended solids, and (3) mechanical filtration. Any remaining suspended solids from this process will be discharged to (1) one or both of the basin water treatment systems at the sand filter or cartridge filter or (2) the basin. Cartridge filter changeouts and backwashing of the sand filter to the North Load-Out Pit are routine basin activities. The spent cartridge filters are disposed of as low level radioactive waste in accordance with WHC-EP-0063-3 (WHC 1988a). The settled solids from the sand filter backwash will continue to be accumulated and will also be encapsulated.

Because the above process also places the sludge in suspension underwater, but adds the feature of filtering equipment, the sludge encapsulation is expected to result in less sludge suspension in the basin water than that experienced either during the 1984 fuel segregation or the more recent sludge transfers from the discharge pickup chute to the Weasel Pit prior to encapsulation equipment placement.

1.3.2.3 Crushing of Canisters. The emptied canisters will be moved underwater to the discharge pickup chute area and cleaned by using a high-pressure water jet and wire brushes. Each canister will then be crushed underwater, raised above the water and promptly inserted into a plastic bag. It is estimated that 1,400 aluminum canisters and 2,300 stainless steel canisters will be crushed during encapsulation. Approximately 2,000 empty canisters from basin fuel inventories, previously shipped for processing, also will be cleaned, crushed, bagged, and disposed of. The packaged, crushed canisters will be placed in burial boxes. These burial boxes will be sealed in accordance with WHC-EP-0063-3 (WHC 1988a). It is estimated that one burial box will be loaded and shipped for burial each week.

2.0 DESCRIBE THE SOURCE TERM

Describe the physical form of each radionuclide used (or created) during the process.

2.1 BACKGROUND

The 105-KE Basin contains 1,150 MTU of N Reactor fuel (approximately 3,700 canisters) and two baskets (the equivalent of five canisters) filled with residual aluminum-clad fuel assemblies from retired single-pass reactors. The N Reactor fuel consists of slightly enriched metallic uranium completely enclosed and bonded to a layer of zirconium alloy (Zircaloy-2), also known as the cladding. The cladding is designed to provide a barrier against the escape of fission products and fissile materials.

The fuel was discharged from N Reactor between 4 and 21 years ago. The fuel has decayed sufficiently to essentially eliminate ^{131}I , as well as other short half-life radionuclides. Following discharge of the fuel from the N Reactor, the fuel was allowed to cool for a minimum of 150 days in the N Basin. Then, the fuel was placed into open-top canisters, loaded into rail cask cars, and transported to the 105-KE Basin for storage.

Fuel elements with undamaged cladding retain all of their respective radionuclide source term, which is not released to the environment. However, the cladding integrity was occasionally breached during reactor defueling and subsequent handling operations. The loss of cladding integrity ranged from a crack to the complete separation of an element into two, and sometimes more, parts. Once the cladding integrity was lost, the water gained access to the metallic uranium. Even on fuel with damaged cladding, the radionuclides remain bound by the uranium metal as long as it remains in the metallic form. However, when aqueous corrosion of the uranium metal occurs, the radionuclides with high solubilities such as cesium and strontium can dissolve in the basin water and less soluble radionuclides are converted to an oxide form enabling them to be released from the metallic matrix and dispersed in the basin. The radionuclide-containing oxide so formed settles to the basin floor and is incorporated into the sludge.

2.2 SOURCE TERM DESCRIPTION

2.2.1 Fuel Elements

The radionuclide inventory of the irradiated fuel is 9.3 MCi (WHC 1993a).

The majority of the fuel stored in the 105-KE Basin is 0.95 wt% enriched with the ^{235}U isotope. A small amount of natural uranium fuel (0.71 wt% ^{235}U) is also stored in the 105-KE Basin. Because more highly enriched fuel results in higher radionuclide levels for given irradiation and decay times, radionuclide levels were conservatively based on 0.95 wt% ^{235}U enriched fuel.

Approximately 850 MTU of the inventory was irradiated to fuel grade assay levels where the ^{240}Pu content was allowed to increase to 12 percent of the total plutonium produced. The remaining 300 MTU of fuel was irradiated to levels where the ^{240}Pu build up was limited to 6 percent. Longer irradiation times increase the total radionuclide content of the fuel, so for conservatism, the radionuclide content was based on the longer irradiation time for the fuel grade assay.

A 10-year average decay time was used in establishing the present radionuclide content of the fuel. This is conservative because the average age of the fuel after discharge is slightly greater than 10 years. The radionuclides removed by the water treatment systems have not been removed from the source term.

2.2.2 Sludge

The radionuclide source term in the sludge has been determined conservatively to be 0.70 MCi. This value has been calculated by conservatively estimating the total sludge quantity, which consists of the following.

1. Sludge from present and previous inventories existing on all basin floor areas exclusive of the discharge pickup chute area floor.

Sludge on the basin floor, exclusive of the discharge pickup chute area, has been estimated to be 0.64-cm (0.25-in.) deep over the entire basin floor. This estimate is based on comparing the sludge layer to the canister storage racks and canister dimensions.

2. Sludge on the discharge pickup chute area floor from fuel segregation--a prior activity similar to encapsulation.

The depth of sludge on the discharge pickup chute area floor (3-m [10 ft] by 14-m [45-ft]) was estimated at an average depth of 20.32 cm (8 in.) before the sludge was transferred to the Weasel Pit. The depth of sludge (20.32 cm [8 in.]) in the discharge pickup chute (42 m² [450 ft²]) would correspond to 1.09 cm (0.43 in.) if distributed over the entire basin (811 m² [8,375 ft²]). This sludge originated from the fuel segregation effort conducted in 1984 and 1985, involving more than 5,000 canisters of stored fuel.

3. Sludge generated during the encapsulation of existing inventory of fuel elements that are now stored in open canisters.

The projected volume of sludge to be generated during encapsulation is estimated to be 1.09 cm (0.43 in.) evenly distributed over the basin floor. This is a conservative estimate based upon fuel segregation sludge deposition from 5,000 canisters as compared to 3,700 canisters.

The estimated total depth of sludge on the basin floor is:

$$0.64 \text{ cm (0.25 in.) (1)} + 1.09 \text{ cm (0.43 in.) (2)} + 1.09 \text{ cm (0.43 in.) (3)} = 2.82 \text{ cm (1.11 in.)}$$

Using a density of 4 g/cm^3 and the basin dimensions, a total uranium sludge depth of 2.82 cm (1.11 in.) converts to a mass of UO_2 of 87.7 MTU. The amount of oxide formed from corrosion converts to a mass of 77.3 MTU of unoxidized metallic uranium. Based on the fuel radionuclide inventory of 9.3 MCi in 1,150 MTU of fuel (WHC 1993a), the 77.3 MTU is equivalent to 0.62 MCi. For additional conservatism, the sludge can be estimated to contain 0.70 MCi, equivalent to approximately 86.6 MTU of fuel.

In summary, the source term evaluation is conservative for the following reasons:

- This Ci content accounts for no removal of radionuclides by the water treatment systems
- Radionuclide content was based on 0.95 wt% ^{235}U enriched fuel
- The radionuclide content was based on the longer irradiation time for fuel grade assay levels
- A 10-year average decay time was used in establishing the present radionuclide content of the fuel
- The sludge source term represents approximately 86.6 MTU of fuel, larger than that calculated (77.3 MTU) to be present in the volume of the sludge if assumed to be UO_2 .

As discussed previously, the total radionuclide inventory of the fuel (9.3 MCi) can be added to that of the sludge (0.70 MCi), to obtain a total of 10 MCi. The 105-KE Basin fuel element and sludge activity for various radionuclides is presented in Table 2-1 (WHC 1993a).

2.2.3 Basin Water

The sludge and fuel is covered by the $3.86 \times 10^6 \text{ L}$ (1.02 Mgal) of water in the 105-KE Basin (WHC 1988b). Only a small fraction of the sludge source term is present in the water at any given time (WHC 1993a).

The radionuclides of significance in the airborne emissions are known to be particulates that originate from the basin water. The primary mechanisms responsible for airborne contamination are transport at the water line of the basin, the "bathtub ring", and suspension of surface contamination on basin floors, gratings and tools (WHC 1993a).

Table 2-1.^a Calculated 105-KE Basin Fuel Element and Sludge Activity.

Radionuclide	Inventory (Ci)
⁸⁵ Kr	2.45 E+05
⁸⁹ Sr	2.59 E-05
⁹⁰ Sr	2.58 E+06
⁹⁵ Zr	2.66 E-02
⁹⁵ Nb	9.27 E-11
¹⁰³ Ru	5.02 E-09
¹⁰⁶ Ru	1.29 E+05
¹³⁴ Cs	1.73 E+04
¹³⁷ Cs	2.67 E+06
¹⁴⁴ Ce	4.45 E+05
²³⁴ U	5.22 E+02
²³⁵ U	9.45 E+01
²³⁸ U	4.24 E+02
²³⁸ Pu	2.86 E+04
²³⁹ Pu ^b	1.75 E+05
²⁴¹ Pu	3.71 E+06
Total	1.00 E+07

^aThis table does not include the 565 Ci of tritium (³H) contained in the fuel. This was calculated as follows:

4.91 E-01 Ci/MTU x 1,150 MTU = 565 Ci ³H
 (ORIGEN II Prediction of N Reactor Fuel Fission Product Composition, SD-CP-TI-094, Rockwell Hanford Company, Richland, Washington, 1986).

^bThis quantity represents all actinides.

3.0 PROVIDE DRAWINGS OF THE EMISSION UNIT

Provide drawings of the emission unit from point of origin of the source to emission to the environment. If final drawings are not available at the time of submittal, conceptual drawings are acceptable. The Department may request to see the actual drawings after completion of construction.

Radioactive airborne emissions from the 105-KE Basin originate from the basin water. Figure 3-1 presents the location of the vents in the roof of the basin. The primary mechanisms responsible for airborne contamination are transport at the water line of particulates from the basin "bathtub ring" and suspension of surface contamination on floors, gratings, and tools (WHC 1993a). The radionuclides are emitted through the vents to the environment.

3.1 DESCRIBE THE RADIONUCLIDE CONTROL EQUIPMENT

Describe the radionuclide control equipment: Describe the efficiency of each piece of radionuclide control equipment for each radionuclide that could contribute 10 percent or more of the committed effective dose equivalent (CEDE) to the maximally exposed individual (MEI).

The radionuclide control equipment, other than the fuel cladding itself, consists of both passive and active systems directed at controlling the source term. The application of these radionuclide controls are in accordance with ANS-57.7 (ANS 1988).

The passive system consists of the 1.02 Mgal of water, which covers the irradiated fuel and sludge. This water not only precludes the radionuclide source term from the irradiated fuel and sludge from becoming directly airborne, but also provides radiation shielding.

The active system consists of two separate water treatment systems shown in Figure 3-2 to reduce radionuclide concentrations in the basin water. The first water treatment system is composed of a cartridge filter cell, a chiller, and three ion exchange columns. The second water treatment system is composed of skimmers, a sand filter, and two auxiliary ion exchange modules.

The first system uses a recirculation pump to draw the basin water through underwater (2.4-m- [8-ft] deep) header pipes in each basin bay. The water passes through a cartridge filter cell at a rate of 1,893 L (500 gal)/min. The cartridge filter cell is composed of 80 individual filter elements and is designed to remove small (i.e., down to 10-micron in diameter) particulates. A backup filter cell and backup pump are available for use as alternates if necessary. Water is then routed through a chiller designed to remove residual decay heat while maintaining the temperature of the basin water at about 10 °C (50 °F). Water temperature in the basin is required by Process Standards to be between 5.5 °C (42 °F) and 32.2 °C (90 °F); in practice the temperature is kept near 10 °C (50 °F) to reduce evaporation and corrosion. After chilling, the water was normally routed

105 KE Basin Roof Vent (RV), Fixed Head ⊕, and Record ☒ Sampler Locations

Figure 3-1. Location of Roof Vents and Fixed Head Samplers in the 105-KE Basin.

Figure 3-2. Schematic Arrangements of the Water Treatment Systems.

through one of three ion exchange columns receiving flow in parallel operation (at a flow rate of approximately 189 L [50 gal]/min] each) before being returned to the basin. At present, the ion exchange columns are used only sparingly in the 105-KE Basin to minimize the generation of transuranic waste. The water is returned by piping to the basin.

Cartridge filters are expected to be changed quarterly during encapsulation activities based on an increase in differential pressure across the filters, normally about 25 lb/in². Based on past operations, filter cartridges typically remove about 30 Ci of radioactivity before they are changed out.

The second system draws basin water from skimmer weirs along the north wall of the basin and pumps the water at 1,514 L (400 gal)/min. through a large sand filter containing approximately 3,946 kg (8,700 lb) of sand. This water is routinely routed to at least one of two auxiliary ion exchange modules (at approximately 606 L [160 gal]/min each) with the balance bypassing the ion exchange modules (IXM) and returned to the basin. After ion exchange treatment, that portion of the water is discharged to the South Load-Out Pit.

The sand filter is required to be backwashed when an increase in differential pressure occurs. In practice, the pressure limit is not reached and a backwash is performed quarterly in conjunction with sand sampling requirements. The backwash process is performed by first lowering the North Load-Out Pit level by approximately 1.8 m (6 ft). This is accomplished by valving off the skimmers and allowing the skimmer pump to draw directly from the North Load-Out Pit. The North Load-Out Pit (also known as the Sand Filter Backwash Pit) is segregated from the basin by means of a weir. During backwash, the secondary loop valving is changed to reverse flow through the sand filter for approximately 3 minutes discharging filterable solids to the North Load-Out Pit.

The IXMs are large (218-cm [86-in.] x 178-cm [70-in.] x 202-cm [79.5-in.]) self-contained Duolite¹ mixed bed resin systems. The IXM also serves as the container for disposal. Each IXM uses 21 ft³ of resin. The IXMs are changed out when the removal efficiency for ¹³⁷Cs decreases from the initial 99 percent to the 20 percent range. The run times vary depending on basin water quality and changeouts can occur from two to six times per year.

At least one of the water treatment systems with ion exchange capabilities is kept in service. Outages occur occasionally for equipment failures, repairs, and component changeouts as necessary. Operation of the water treatment systems provides the control necessary to ensure facility air emissions remain within those forecasted in Section 4.2.

¹Registered trademark of Rohm and Haas, Inc.

3.2 RADIONUCLIDE CONTROL EFFICIENCIES

The radionuclide control efficiency for the water treatment system can be estimated based on recent operating history in the 105-KE Basin. The overall efficiency can be represented as follows:

$$\text{Efficiency} = 1 - \left(\frac{\text{radionuclide inventory with basin water treatment}}{\text{radionuclide inventory without basin water treatment}} \right)$$

To properly assess the efficiency, the above calculation is based upon data taken for time periods when past basin activities were similar to that expected during encapsulation. Accordingly, two time periods were selected.

- October 1992--The water treatment systems were shutdown during the entire month because of chiller installation and resin waste management issues.
- November 1992--One IXM was used for the time period of November 3, 1992 through November 30, 1992 (represents the minimum configuration necessary to provide water treatment).

For the two-month time period above, extensive sludge pumping and sludge debris raking were conducted in the discharge pickup chute area to prepare for encapsulation equipment installation.

Total radionuclides, as calculated from water samples taken from the center of the basin, can be used to calculate the efficiency of the water treatment system.

$$\text{Efficiency} = 1 - \frac{(45.9 \text{ Ci [November]})}{(76.5 \text{ Ci [October]})} = 40\%$$

The above represents an estimate of the water treatment system efficiency for removing radionuclides from the basin water, based upon basin water data.

3.3 WATER TREATMENT SYSTEMS COMPONENT RADIONUCLIDE REMOVAL EFFICIENCIES

The water treatment systems contain ion exchange components for removal of soluble radionuclides and particulate filters for removal of particulate radionuclides. These components can affect emission levels because their operation and basin activities determine the quantity of radionuclides that are present in the water at a given time. Experience has shown that the

operational controls presented in Section 3.4 can control radionuclide concentrations in the water so that emissions of radionuclides will not exceed those forecast in Section 4.2 for any 12-month period.

Because the primary consideration in effectiveness of control for radionuclide concentrations in the water is water treatment systems operation, the efficiency of the individual components is discussed in the following paragraphs.

The ion exchange columns and IXM radionuclide removal efficiencies vary with time and decrease with increasing run time. The IXMs are run to a plutonium removal efficiency approaching 20 percent. Initial plutonium removal efficiency is 99.6 percent and the average removal efficiency is about 81 percent (WHC 1993a). Radiological controls preclude loading the ion columns to as high a radionuclide content as the IXMs. The average removal efficiencies for plutonium isotopes by the ion columns and IXMs are 74 and 81 percent respectively.

The particulate cartridge filter of the primary water treatment system is designed to remove particulates greater than 10 microns. The particulate sand filter is designed to remove particles greater than 100 microns in diameter. The cartridge and sand filters remove approximately 12 percent of the total plutonium concentrations in the inlet stream, indicating some of the plutonium is particulate, but that most of it is either dissolved or very fine. Ion exchange column and IXM plutonium removal efficiencies indicate the majority of plutonium in the water is dissolved. The particulate filters show negligible removal efficiencies for the dissolved cesium and strontium as would be expected. The efficiencies of the individual components in the water treatment systems are presented in Table 3-1.

Table 3-1. Average Radionuclide Removal Efficiencies of Components of the Water Treatment Systems at 105-KE Basin.^a

Radionuclide	Ion exchange column (%)	Ion exchange module (%)	Sand filter (%)	Cartridge filter (%)
Strontium	91	93.5	Negligible	Negligible
Cesium	98	49	Negligible	Negligible
Plutonium	74.3	81	Less than 12	12

^a*Transport Mechanism of Radionuclides in 105-KE Fuel Storage Basin to Airborne Effluent Release*, Westinghouse Hanford Company, Richland, Washington, 1993.

3.4 OPERATIONAL CONTROL

The radionuclide control equipment for the 105-KE Basin is operated such that projected emissions will remain within those forecasted in Section 4.2.

At least one of the two water treatment systems is required to be in service¹ routinely with the following exceptions:

- An outage of up to 48 hours for changeout of the ion exchange cleanup components (IXMs or Ion Columns).
- An outage of up to 48 hours for changeout or backflushing, as appropriate, of the cartridge filters or sand filter, respectively.
- An outage of up to 24 hours for a power failure to the facility.
- An outage of up to 72 hours for a component failure or administrative issue.

If neither system can be returned to operation within the outage durations provided above, no fuel or sludge handling will be permitted until at least one of the two systems is returned to service.

¹"In service" is defined as follows:

- In the water treatment system consisting of a cartridge filter cell, a chiller, and three ion exchange columns, at least one pump will operate; at least one cartridge filter cell is on line; and three ion columns are receiving flow in parallel operation
- In the water treatment system consisting of skimmers, a sand filter, and two auxiliary IXMs, at least one pump will operate; the sand filter will be in service; and at least one IXM will be on line.

This page intentionally left blank.

4.0 PROVIDE EXPECTED ANNUAL EMISSIONS

Provide expected annual emissions with radionuclide control equipment in place, or using 40 CFR 61, Appendix D, methodology, for each radionuclide that could contribute 10 percent or more of the CEDE to the MEI.

The following provides projections of emissions based on (1) the methodology prescribed in 40 CFR 61, Appendix D, and (2) actual emissions data and good engineering judgment.

4.1 PROJECTED EMISSIONS BASED ON APPENDIX D

Table 4-1 presents the projected annual emissions from the encapsulation activity using 40 CFR 61, Appendix D, methodology. Radionuclides used in this projection are those that have been identified previously in Table 2-1 as representing the fuel element activity (9.3×10^6 Ci) and sludge activity (0.70×10^6 Ci). The inventory of each radionuclide was multiplied by a physical state adjustment factor to calculate a projected annual release.

As indicated in Table 4-1, the projected annual release for all radionuclides using the Appendix D Methodology is 2.55×10^5 Ci/yr.

4.2 PROJECTED EMISSIONS BASED ON GOOD ENGINEERING JUDGMENT

Table 4-2 was developed from actual data obtained from the fixed head sampler (RV11) shown in Figure 3-1. Chapter 5.0 presents a description of the fixed head sampler. This data was obtained during the sampling period from October through December 1992.

This sampling period was selected for two reasons. First, the extensive sludge pumping and sludge debris raking during this time period resulted in increased suspension of radionuclides in the basin water similar to that expected during encapsulation. Second, the radionuclides in the water were further elevated by the shutdown of the water treatment system during October and the first two weeks of December 1992. The shutdowns of the water treatment system were necessary to replace the existing water-cooled chiller with a new air-cooled chiller and to minimize the generation of transuranic waste associated with operation of the ion exchange columns. The combination of the increase in suspended sludge and the necessity of the water treatment system shutdowns during this period resulted in radionuclide concentrations in the water that afforded a bounding case projection of air emissions during encapsulation.

Table 4-1.^a Projected Annual Airborne Emissions
Using 40 CFR 61, Appendix D.

Radionuclide	Inventory (Ci)	Physical state adjustment factor	Projected (Ci/yr) annual emissions
⁸⁵ Kr	2.45 E+05	1.0	2.45 E+05
⁸⁹ Sr	2.59 E-05	10 ⁻³	2.59 E-08
⁹⁰ Sr	2.58 E+06	10 ⁻³	2.58 E+03
⁹⁵ Zr	2.66 E-02	10 ⁻³	2.66 E-05
⁹⁵ Nb	9.27 E-11	10 ⁻³	9.27 E-14
¹⁰³ Ru	5.02 E-09	10 ⁻³	5.02 E-12
¹⁰⁶ Ru	1.29 E+05	10 ⁻³	1.29 E+02
¹³⁴ Cs	1.73 E+04	10 ⁻³	1.73 E+01
¹³⁷ Cs	2.67 E+06	10 ⁻³	2.67 E+03
¹⁴⁴ Ce	4.45 E+05	10 ⁻³	4.45 E+02
²³⁴ U	5.22 E+02	10 ⁻³	5.22 E-01
²³⁵ U	9.45 E+01	10 ⁻³	9.45 E-02
²³⁸ U	4.24 E+02	10 ⁻³	4.24 E-01
²³⁸ Pu	2.86 E+04	10 ⁻³	2.86 E+01
²³⁹ Pu ^b	1.75 E+05	10 ⁻³	1.75 E+02
²⁴¹ Pu	3.71 E+06	10 ⁻³	3.71 E+03
Total	1.00 E+07		2.55 E+05

^aThe projected annual emissions presented in this table does not include 565 Ci of tritium (³H) noted in Table 2-1. If released, tritium would have a physical-state adjustment factor of 1.0, which would result in a projected release of 565 Ci/yr.

^bThis quantity represents all actinides.

Table 4-2.^a Projected Annual Airborne Emissions Using Fixed Head Sampler (RVII) Data.

Radionuclide	Average concentration (October 1992 to December 1992) ($\mu\text{Ci}/\text{ml}$)	Projected annual emissions (Ci/yr)
^{241}Am	1.8 E-14	1.4 E-05
^{60}Co	1.5 E-14	1.1 E-05
^{137}Cs	6.4 E-13	4.8 E-04
^{238}Pu	6.9 E-15	5.2 E-06
$^{239/240}\text{Pu}$	4.2 E-14	3.1 E-05
^{241}Pu	5.3 E-14	4.0 E-05
^{106}Ru	3.3 E-14	2.4 E-05
^{90}Sr	7.5 E-13	5.6 E-04
Total		1.2 E-03

^aThe projected annual emissions presented in this table do not include tritium (^3H). Multiplying the concentration of tritium in the basin water by the evaporation rate of the basin water yields the amount of tritium released.

Concentration of ^3H in basin water (WHC 1993a) = $3.0 \text{ E-03 } \mu\text{Ci}/\text{mL}$
 Evaporation rate of the basin water = $41.6 \text{ L}/\text{h}$

$$\left(3.0 \times 10^{-03} \frac{\mu\text{Ci}}{\text{mL}}\right) \left(4.16 \times 10^04 \frac{\text{mL}}{\text{h}}\right) \left(8,760 \frac{\text{h}}{\text{year}}\right) \left(\frac{10^{-06} \text{ Ci}}{1 \mu\text{Ci}}\right) = 1.2 \frac{\text{Ci}}{\text{year}}$$

Table 4-2 lists the individual radionuclide concentrations, averaged for the sampling period October through December 1992. The projected annual release for each radionuclide was calculated by multiplying the individual radionuclide concentrations by the anticipated maximum annual flow rate (WHC 1993b) with all four vent fans operating continuously. Formulas for the flow rate and projected annual emissions are as follows:

Annual Flow Rate:

$$\left(60 \frac{\text{min.}}{\text{h}}\right) \left(8,760 \frac{\text{h}}{\text{year}}\right) \left(50,273 \frac{\text{ft}^3}{\text{min.}}\right) \left(28.32 \frac{\text{L}}{\text{ft}^3}\right) \left(1000 \frac{\text{mL}}{\text{L}}\right) = 7.5 \text{ E}+14 \frac{\text{mL}}{\text{year}}$$

Expected Annual Emission:

$$\left(\text{concentration} \frac{\mu\text{Ci}}{\text{mL}}\right) \left(\frac{10^{-06} \text{Ci}}{1 \mu\text{Ci}}\right) \left(7.5 \text{ E}+14 \frac{\text{mL}}{\text{year}}\right) = \frac{\text{Ci}}{\text{year}} \text{ annual emission}$$

The results of this projection, $1.2 \times 10^{-3} \text{ Ci/yr}$, is the projected annual airborne emissions from the 105-KE Basin during encapsulation activities.

5.0 DESCRIBE THE MONITORING EQUIPMENT

Describe the monitoring equipment. Describe the minimum detectable concentration for each radionuclide that could contribute 10 percent or more of the CEDE to the MEI.

The exhaust system in the basin consists of two exhaust fans over the 105-KE Basin pool (low bay) and two in the high bay area of the facility, where the rail service and ion exchange treatment components are located. The exhaust flows have recently been measured and reported in WHC 1993b. The current airborne radionuclide emissions monitoring systems for the 105-KE Basin consist of an effluent sampling system and a fixed head sampling system. The sampling system of the two low bay exhausters consists of a single probe in each exhaust duct. The sample line from each probe is constructed of 3/8-in. O.D. copper tubing and extends from the exhaust duct to a tee fitting. From the tee a single sample line extends to the sample cartridge located at floor level for ease of filter exchange. The total sample line length from the probe to the sample cartridge is about 21 m (70 ft). There are seven right angle bends in the sample line.

The effluent sampling system uses a vacuum pump, which draws a sample of the air discharge through a 47-mm particulate filter. The sampler on the 105 KE Sampling System is a 1/4 horse power Gast Integral Vacuum Pump. The particulate sample is collected on 47-mm glass fiber filters. The sample cartridge containing the particulate filter contains a 0.03 m^3 (1.2 ft^3)/min flow limiting orifice. The sample cartridge also has the capability to contain a charcoal filter for sampling halogens, e.g., iodine. However, due to the short half-lives of the halogens, they are no longer present in the effluent and this sampling was discontinued after several years of negative results.

The effluent sampler on the 105-KE exhaust is equipped with vacuum gauges for determination of flow rate. System operability is checked daily and the vacuum gauges are checked weekly to verify a flow rate of at least 0.03 m^3 (1.2 ft^3)/min. The system is also checked weekly with a National Institute of Standards and Technology traceable vacuum gauge. The flow limiting orifice is designed to provide a constant 0.03 m^3 (1.2 ft^3)/min flow to the sampler. These flow measurements are logged and maintained as quality records.

The particulate filters are collected weekly. The filters are delivered to an offsite laboratory for weekly analysis of gross alpha/beta constituents and performance of a gamma scan. Weekly filters are composited monthly for ^{90}Sr and Pu isotopic analysis.

In addition to the effluent sampling system described above, additional fixed head samplers have been placed in the 105-KE Basin near the inlet to each of the low bay exhausts and at the mezzanine level in the high bay (Figure 3-1). These additional samplers employ similar particulate sampling equipment as described above. The purpose of the fixed head sampling system is to evaluate line loss effects associated with the effluent sampling system. The fixed head sample system results were used in projecting airborne emissions and doses presented in this Application. These fixed head samplers will remain operational during encapsulation activities.

The particulate radionuclides contributing 10 percent or more of the CEDE from 105-KE are $^{239/240}\text{Pu}$ and ^{241}Am (Table 5-1). The particulate filters are analyzed by International Technologies Laboratories of Richland. Contracted detection limits of $7.5 \text{ E}^{-16} \mu\text{Ci per ml}$ have been established for each radionuclide listed below.

Table 5-1. Particulate Radionuclides Contributing
10 Percent or More of the Committed
Effective Dose Equivalent.

Radionuclide	Committed effective dose equivalent as presented in Table 6-2 (%)
$^{239/240}\text{Pu}$	51
^{241}Am	34

6.0 PROJECTED DOSE TO THE MAXIMALLY EXPOSED INDIVIDUAL

Provide a projected dose to the MEI using an approved code or method.

The following provides projections of CEDE to the MEI using projected emissions based on (1) 40 CFR 61, Appendix D and (2) actual emissions data and good engineering judgment.

6.1 PROJECTED DOSE TO THE MAXIMALLY EXPOSED INDIVIDUAL USING 40 CFR 61, APPENDIX D, PROJECTED EMISSIONS AND CAP 88

The projected CEDE to the MEI using Appendix D projected emissions is shown in Table 6-1. The dose conversion factor used was derived from the EPA-approved code, CAP 88 (WHC 1991). The projected dose from each individual radionuclide was calculated by multiplying the projected annual emission (Table 4-1) by the dose conversion factor.

6.2 PROJECTED DOSE TO THE MAXIMALLY EXPOSED INDIVIDUAL USING GOOD ENGINEERING JUDGMENT PROJECTED EMISSIONS AND CAP 88

The projected CEDE to the MEI using good engineering judgement projected emissions is shown in Table 6-2. The dose conversion factor used was derived from the EPA-approved code, CAP 88 (WHC 1991). The projected dose from each individual radionuclide was calculated by multiplying the projected annual emission (Table 4-2) by the dose conversion factor.

6.3 PROJECTED DOSE TO THE MAXIMALLY EXPOSED INDIVIDUAL USING GOOD ENGINEERING JUDGMENT PROJECTED EMISSIONS AND GENII

The projected CEDE to the MEI, using good engineering judgment projected emissions is shown in Table 6-3. The dose conversion factor used was derived from the GENII code. The projected dose from each individual radionuclide was calculated by multiplying the projected annual emission (Table 4-2) by the dose conversion factor.

The application of either the GENII or CAP 88 codes to the projected emissions in Tables 6-2 or 6-3 result in effectively the same dose.

Table 6-1.^a Committed Effective Dose Equivalent to the Maximally Exposed Individual Using Projected Emissions Based on 40 CFR 61, Appendix D, Methodology (Table 4-1) and CAP 88 Dose Conversion Factors.^b

Radionuclide	Projected annual emissions (from Table 4-1) (Ci/yr)	CAP-88 dose conversion factor	Committed effective dose equivalent to maximally exposed individual (mrem/yr)	Dose, percent of total
⁸⁵ Kr	2.45 E+05	7.49 E-08	1.8 E-02	0.0
⁸⁹ Sr	2.59 E-08	3.11 E-03	8.1 E-11	0.0
⁹⁰ Sr	2.58 E+03	6.45 E-02	1.7 E+02	4.6
⁹⁵ Zr	2.66 E-05	3.90 E-03	1.0 E-07	0.0
⁹⁵ Nb	9.27 E-14	2.60 E-03	2.4 E-16	0.0
¹⁰³ Ru	5.02 E-12	2.10 E-03	1.1 E-14	0.0
¹⁰⁶ Ru	1.29 E+02	3.08 E-02	4.0 E+00	0.1
¹³⁴ Cs	1.73 E+01	4.62 E-02	8.0 E-01	0.0
¹³⁷ Cs	2.67 E+03	3.53 E-02	9.4 E+01	2.6
¹⁴⁴ Ce	4.45 E+02	2.02 E-02	9.0 E+00	0.2
²³⁴ U	5.22 E-01	4.70 E+00	2.5 E+00	0.1
²³⁵ U	9.45 E-02	4.37 E+00	4.1 E-01	0.0
²³⁸ U	4.24 E-01	4.19 E+00	1.8 E+00	0.0
²³⁸ Pu	2.86 E+01	1.18 E+01	3.4 E+02	9.3
²³⁹ Pu ^c	1.75 E+02	1.28 E+01	2.2 E+03	62.1
²⁴¹ Pu	3.71 E+03	2.03 E-01	7.5 E+02	20.8
Total	2.55 E+05		3.6 E+03	100.0

^aThe above table does not include tritium (³H). Using 565 Ci ³H, as noted in Table 4-1, and a dose conversion factor of 3.4 E-05, a dose of 1.9 E-02 mrem/yr would result.

^bThe maximally exposed individual is located 9.9 kilometers west of the 100 Area.

^cThis quantity represents all actinides.

Table 6-2.^a Committed Effective Dose Equivalent to the Maximally Exposed Individual Using Projected Emissions Based on the Fixed Head Sampler (Table 4-2) and CAP 88 Dose Conversion Factors.^b

Radionuclide	Projected annual emissions (from Table 4-2) (Ci/yr)	CAP-88 dose conversion factor	Committed effective dose equivalent to maximally exposed individual (mrem/yr)	Dose, percent of total
²⁴¹ Am	1.4 E-05	1.94 E+01	2.6 E-04	33.5
⁶⁰ Co	1.1 E-05	4.28 E-02	4.8 E-07	0.1
¹³⁷ Cs	4.8 E-04	3.53 E-02	1.7 E-05	2.2
²³⁸ Pu	5.2 E-06	1.18 E+01	6.1 E-05	7.8
^{239/240} Pu	3.1 E-05	1.28 E+01	4.0 E-04	50.8
²⁴¹ Pu	4.0 E-05	2.03 E-01	8.0 E-06	1.0
¹⁰⁶ Ru	2.4 E-05	3.08 E-02	7.5 E-07	0.1
⁹⁰ Sr	5.6 E-04	6.45 E-02	3.6 E-05	4.6
Total	1.2 E-03		7.9 E-04	100.0

^aThe above table does not include tritium (³H). Using an emission rate of 1.2 Ci per year, as calculated in the note for Table 4-2, and a dose conversion factor of 3.4 E-05, a dose of 4.0 E-05 mrem/yr would result.

^bThe maximally exposed individual is located 9.9 kilometers west of the 100 Area.

Table 6-3.^a Committed Effective Dose Equivalent to the Maximally Exposed Individual Using Projected Emissions Based on the Fixed Head Sampler (Table 4-2) and GENII Dose Conversion Factors.^b

Radionuclide	Projected annual emissions (from Table 4-2) (Ci/yr)	GENII dose conversion factor	Committed effective dose equivalent to maximally exposed individual (mrem/yr)	Dose, percent of total
²⁴¹ Am	1.4 E-05	1.70 E+01	2.3 E-04	33.5
⁶⁰ Co	1.1 E-05	1.60 E-02	1.8 E-07	0.0
¹³⁷ Cs	4.8 E-04	3.80 E-02	1.8 E-05	2.6
²³⁸ Pu	5.2 E-06	1.10 E+01	5.7 E-05	8.3
^{239/240} Pu	3.1 E-05	1.10 E+01	3.4 E-04	49.7
²⁴¹ Pu	4.0 E-05	1.80 E-01	7.1 E-06	1.0
¹⁰⁶ Ru	2.4 E-05	2.40 E-02	5.9 E-07	0.1
⁹⁰ Sr	5.6 E-04	5.90 E-02	3.3 E-05	4.8
Total	1.2 E-03		6.9 E-04	100.0

^aThe above table does not include tritium (³H). Using an emission rate of 1.2 Ci per year, as calculated in the note for Table 4-2, and a dose conversion factor of 3.5 E-05, a dose of 4.2 E-05 mrem/yr would result.

^bThe maximally exposed individual is located 9.9 kilometers west of the 100 Area.

7.0 BEST AVAILABLE RADIONUCLIDE CONTROL TECHNOLOGY ASSESSMENT

A best available radionuclide control technology (BARCT) assessment is currently being completed for the 105-KE Encapsulation Activity. The BARCT assessment will be submitted to the EPA for information purposes only as a separate document by the U.S. Department of Energy, Richland Field Office.

This page intentionally left blank.

8.0 REFERENCES

- 40 CFR 61, "National Emissions Standards for Hazardous Air Pollutants," *Code of Federal Regulations*, as amended.
- ANS, 1988, *Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type)*, ANS 57.7-1988, American Nuclear Society, LaGrange, Illinois.
- Rockwell, 1986, *ORIGEN II Prediction of N Reactor Fuel Fission Product Composition*, SD-CP-TI-094, Rockwell Hanford Company, Richland, Washington.
- WAC, 246-247, 1990, "Radiation Protection--Air Emissions," Washington Administrative Code, as amended.
- WHC, 1988a, *Hanford Site Solid Waste Acceptance Criteria*, WHC-EP-0063-3, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1988b, *Safety Analysis and Irradiated N Reactor Fuel*, WHC-SP-0300, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1991, *Unit Dose Calculation Methods and Summary of Facility Effluent Monitoring Plan Determinations*, WHC-EP-0498, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1993a, *Transport Mechanism of Radionuclides in 105-KE Fuel Storage Basin to Airborne Effluent Release*, WHC-SD-NR-ES-016, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1993b, *Test Report - 100K Basin Air Flow Effluent Volume Test*, WHC-SD-NR-TRP-015, Westinghouse Hanford Company, Richland, Washington.

This page intentionally left blank.

DATE

FILMED

6 / 22 / 94

END