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DOE/RL-94-14
Revision 0

UC-630,721

Radioactive Air Emissions Notice of Construction for the 105N Basin Stabilization

APR 1994
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Prepared for the U.S. Department of Energy
Office of Environmental Restoration and
Waste Management

United States
Department of Energy
P.O. Box 550
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Approved for Public Release

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Printed in the United States of America

DISCLM-5.CHP (8-91)

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ACRONYMS

ALARA	As Low As Reasonably Achievable
BARCT	Best Available Radionuclide Control Technology
basin	105N Basin
CAM	continuous air monitors
CAP-88	Clean Air Assessment Package 1988
CFR	*Code of Federal Regulations*
DAC	derived air concentration
DOE	U.S. Department of Energy
DOH	Washington State Department of Health
DOT	U.S. Department of Transportation
EDE	effective dose equivalent
EPA	U.S. Environmental Protection Agency
GEJ	good engineering judgement
HEPA	high-efficiency particulate air
IXM	Ion Exchange Modules
MEI	maximally exposed individual
PAPR	powered air purifying respirator
PNL	Pacific Northwest Laboratory
ROSEE	Remotely Operated Sediment Extraction Equipment
SNM	Special Nuclear Material
WAC	*Washington Administrative Code*

METRIC CONVERSION CHART

The following conversion chart is provided to the reader to aid in conversion.

Into metric units			Out of metric units		
If you know	Multiply by	To get	If you know	Multiply by	To get
Length			**Length**		
inches	25.40	millimeters	millimeters	0.0393	inches
inches	2.54	centimeters	centimeters	0.393	inches
feet	0.3048	meters	meters	3.2808	feet
yards	0.914	meters	meters	1.09	yards
miles	1.609	kilometers	kilometers	0.62	miles
Area			**Area**		
square inches	6.4516	square centimeters	square centimeters	0.155	square inches
square feet	0.092	square meters	square meters	10.7639	square feet
square yards	0.836	square meters	square meters	1.20	square yards
square miles	2.59	square kilometers	square kilometers	0.39	square miles
acres	0.404	hectares	hectares	2.471	acres
Mass (weight)			**Mass (weight)**		
ounces	28.35	grams	grams	0.0352	ounces
pounds	0.453	kilograms	kilograms	2.2046	pounds
short ton	0.907	metric ton	metric ton	1.10	short ton
Volume			**Volume**		
fluid ounces	29.57	milliliters	milliliters	0.03	fluid ounces
quarts	0.95	liters	liters	1.057	quarts
gallons	3.79	liters	liters	0.26	gallons
cubic feet	0.03	cubic meters	cubic meters	35.3147	cubic feet
cubic yards	0.76	cubic meters	cubic meters	1.308	cubic yards
Temperature			**Temperature**		
Fahrenheit	subtract 32 then multiply by 5/9ths	Celsius	Celsius	multiply by 9/5ths, then add 32	Fahrenheit

Source: *Engineering Unit Conversions*, M. R. Lindeburg, PE., Second Ed., 1990, Professional Publications, Inc., Belmont, California.

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RADIOACTIVE AIR EMISSIONS NOTICE OF CONSTRUCTION FOR THE 105N BASIN STABILIZATION

1.0 FACILITY INFORMATION

The 105N Basin (basin) Stabilization will place the basin in a radiologically and environmentally safe condition so that it can be decommissioned at a later date. The basin stabilization objectives are to inspect for Special Nuclear Material (SNM) (i.e., fuel assemblies and fuel pieces), remove the water from the basin and associated pits, and stabilize the basin surface. The stabilization will involve removal of basin hardware, removal of basin sediments, draining of basin water, and cleaning and stabilizing basin surfaces to prevent resuspension of radioactive emissions to the air. These activities will be conducted in accordance with all applicable regulations.

1.1 DESCRIPTION OF FACILITY

The basin is in the 105N Building, which is located in the 100N Area (Figure 1-1). The 100N Area is located in the Northern portion of the Hanford Site approximately 35 miles northwest of the city of Richland, Washington (Figure 1-2).

The basin is a reinforced unlined concrete structure 150 feet long, 50 feet wide, and 24 feet deep. The basin is segregated into seven areas sharing a common pool of water; the Discharge/Viewing ("D") Pit, the fuel segregation pit (including a water tunnel that connects the "D" pit and segregation pit), two storage basins designated as North Basin and South Basin, two cask load-out pits, and a fuel examination area. Figure 1-3 shows a map of the basin complex. The North Basin floor is entirely covered and the South Basin is partly covered by a modular array of cubicles formed by boron concrete posts and boron concrete panels. Normal water depth for the basin is 23 feet, 5 inches.

Two ancillary facilities, the charge "C" Elevator Pit, and a water Lift Station are associated with the basin but are separate from the fuel storage water pool (Figure 1-3). Currently, approximately 12 feet of water exists in the "C" Elevator Pit. The Lift Station is located underground and north of the North Basin. The Lift Station pump well serves as a collection and segregation point for waste water from the 105N Reactor building equipment and interior space drain system. During stabilization activities, water from piping, sumps, and the "C" Elevator Pit will be routed to the Lift Station. Water in the Lift Station and the basin will be removed for processing and disposition elsewhere on the Hanford Site. Hardware and sediment from the "C" Elevator Pit will not be removed as part of stabilization activities because of As Low As Reasonably Achievable (ALARA) concerns. The C Pit is contained within that part of the 105N Building that houses the reactor core and will be part of future decommissioning activities. Additionally, no other stabilization activities associated with the Lift Station will be performed because of ALARA concerns.

1.2 DESCRIPTION OF ACTIVITIES

The basin stabilization has been divided into seven tasks listed as follows:

- Task 1 - North Cast Pit
- Task 2 - South Cask Pit
- Task 3 - North Basin
- Task 4 - South Basin
- Task 5 - Examination Pit
- Task 6 - Segregation Pit
- Task 7 - Discharge Pit.

Each task will be completed by performing specific activities that are described in general below. All or portions of several activities may be conducted simultaneously. Sections 1.2.1 and 1.2.2 provide available details regarding the activities required to stabilize the basin.

The general approach under each task is to first hydroscrub the basin walls along the water line. The purpose of this activity is to reduce personnel exposure in the work area. When this activity is complete, personnel will start sediment removal with the Remotely Operated Sediment Extraction Equipment (ROSEE) system in conjunction with hardware and debris removal. ROSEE is a system that will vacuum the sediment through a cyclone separator for discharge into the North Cask Pit. Objects that cannot be removed with the ROSEE system (i.e., size limitations) will be sorted by dose rate. This sorting will be accomplished using a submerged radiation detection instrument.

Most of the waste generated during the basin cleanup activities is expected to be low-level radioactive waste. This waste includes cubicle lids, canisters, and fuel baskets. Two alternatives have been identified for disposal of low-level radioactive waste. The alternatives are to bury the low level waste in approved burial containers on the Hanford Site, or transport the waste to a U.S. Department of Energy (DOE)-approved smelter for volume reduction. Volume reduction for the burial option will include crushing the canisters and possible cutting of the fuel baskets.

The preferred alternative is to ship the large quantity of contaminated metal to a DOE-approved smelter for volume reduction. In this case, the canisters will not be crushed, nor will the fuel baskets be cut into smaller pieces. Instead, the canisters, baskets, and other metal items will be placed in vender-supplied containers and shipped to the smelting facility. At the smelting facility; the molten metal will be separated from the slag, which will contain the majority of the contamination. The metal will be retained at the smelting facility and the slag will be returned to the Hanford Site for burial. It is anticipated that for every 100 pounds of metal waste delivered to the smelting facility, 1 pound of slag waste will be returned to the Hanford Site Repository for burial as radioactive waste.

Depending upon the task, other activities may be performed concurrently with sediment removal and hardware movement and removal. These activities might be underwater cutting (plasma arc), mechanical shearing, canister cleaning, and canister crushing. When all material, hardware, and debris are

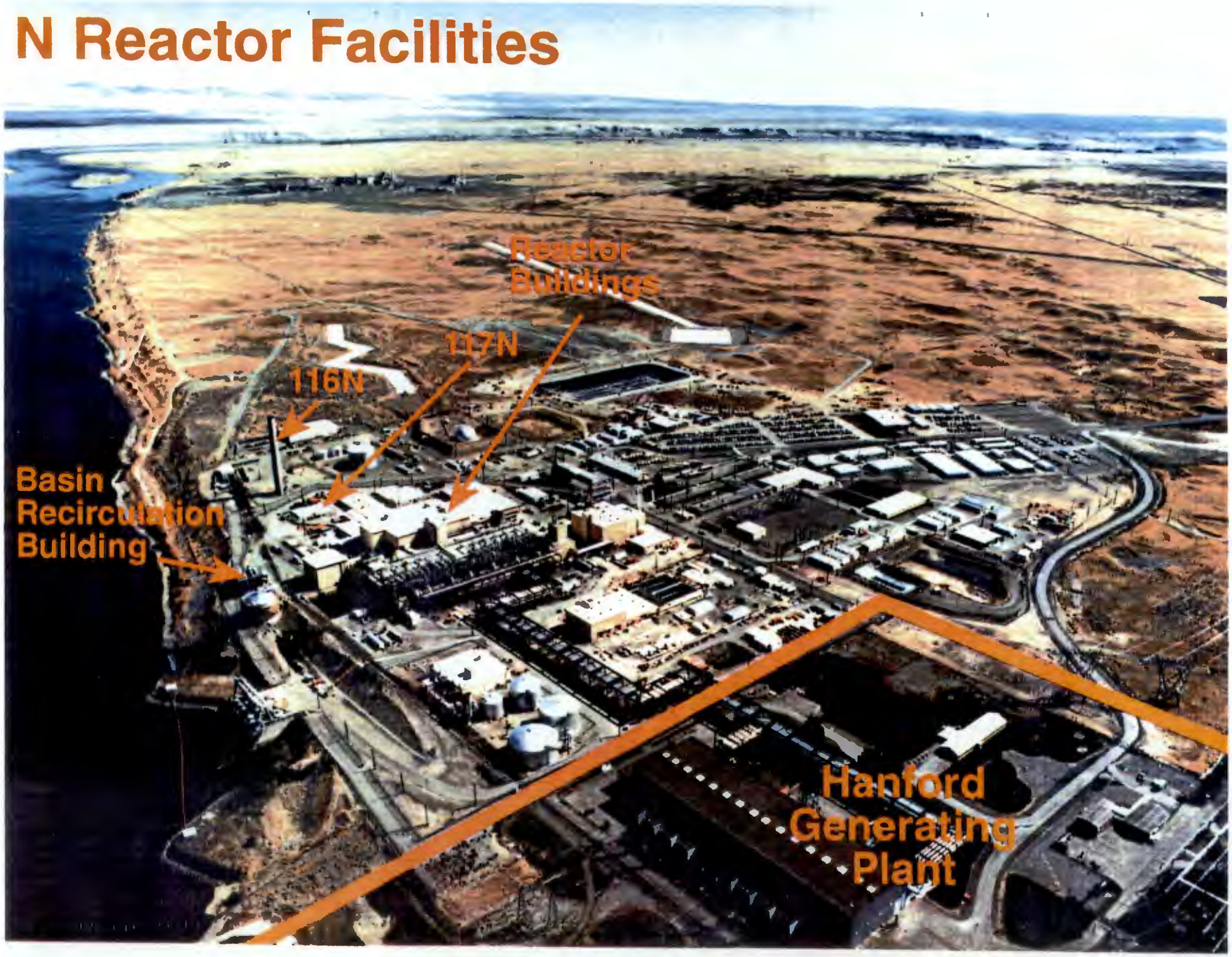

Figure 1-1. Location of 100N Area.

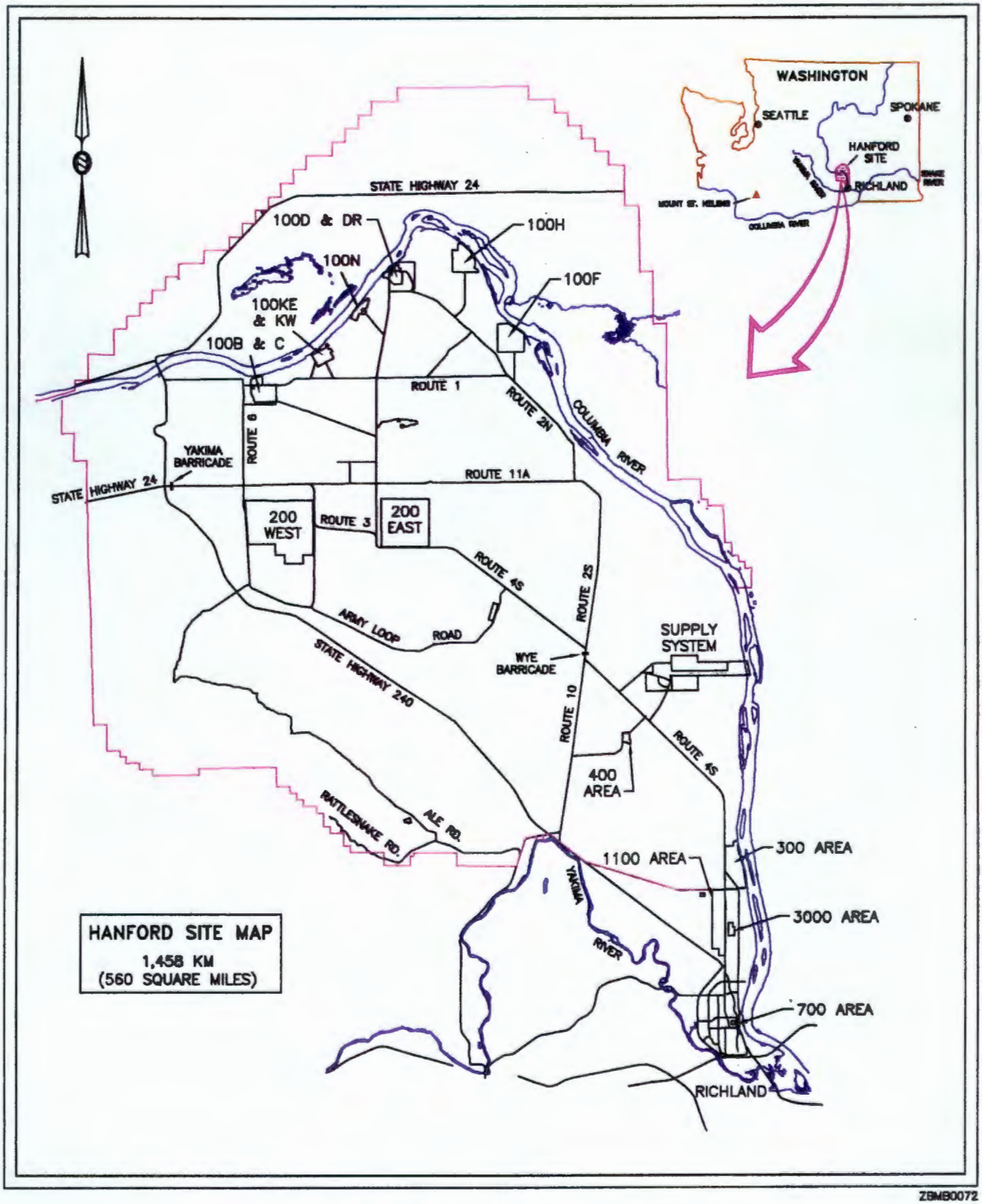

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Figure 1-2. Regional Location of the 100N Area.

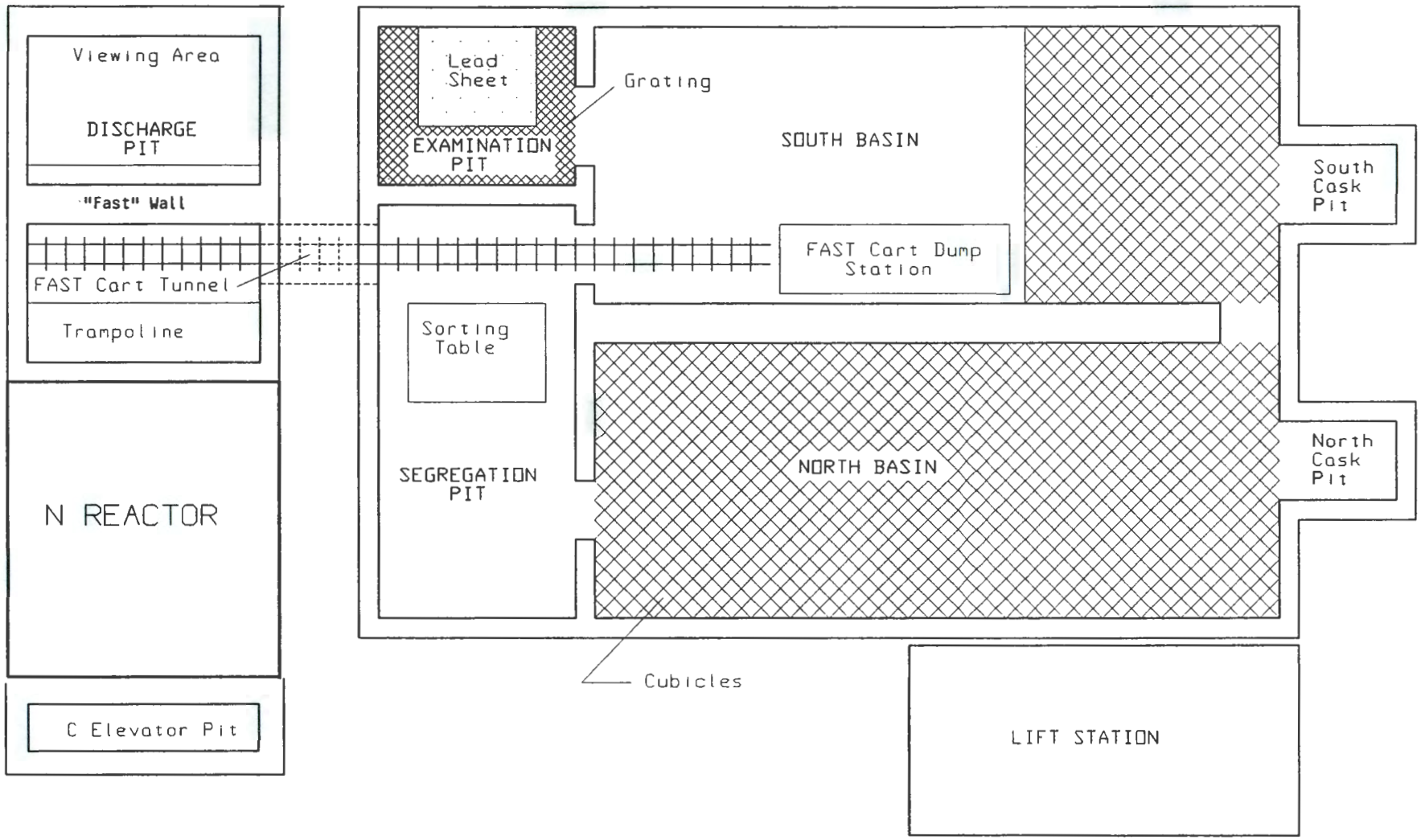


Figure 1-3. Map of the 105N Basin Complex.

removed from the basin, the basin areas will be vacuumed again using the ROSEE system. Concurrent with vacuum activities, each section of the basin will be inspected with remote radiation detection instrumentation and underwater video equipment to certify that all SNM has been removed from the basin. When SNM certification is complete, the basin will be ready for stabilization (i.e., water removal and surface sealing).

A sand filter will be installed in the basin with Ion Exchange Module(s) (IXM) on the outlet. The purpose of this system is to recirculate the basin water to maintain water clarity and reduce occupational radiation exposure. This will be a continuously running system with a flow of 700 gallons per minute. Capabilities for back-flushing the sand filter will be valved into the system. A line coming off the outlet of the IXMs will be used to supply the high-pressure wash station. Figure 1-4 presents a line diagram of the system components. In addition, the overflow from the ROSEE system cyclone separators will be valved into the inlet of the sand filter.

In general, the basin contamination level in the personnel work areas will be kept below 10,000 disintegrations per minute/100 square centimeters beta/gamma, and below 500 disintegrations per minute/100 square centimeters alpha. While these levels may be exceeded for specific localized activities, each specific job performed during the basin stabilization activities will be evaluated and contamination levels will be established based on the criteria set forth in the WHC-CM-1-6 Radiation Control Manual requirements. All activities are held to ALARA. Upon completion of each job or as conditions dictate, the basin area will be cleaned to below the contamination levels established prior to the job.

Full mask protection will be worn at 1/10 of a derived air concentration (DAC) based on Pu-239 and Sr-90, the most limiting isotope of concern. It is anticipated that some activities may require this type of protection. At 5 DAC, based upon the most limiting isotopes, a powered air purifying respirator (PAPR) is required for respiratory protection. It is not expected that any stabilization activities will require a PAPR. Fixed head continuous air samplers are currently installed at specific locations in the basin (Figure 1-5) and will be kept in service for the duration of the basin deactivation project. In addition, a portable fixed head sampler will be set up each the work location in accordance with WHC-CM-1-6 (WHC 1993c). Grab air samples will be taken when respiratory protection is worn to verify the adequacy of protection. Alpha and beta continuous air monitors (CAM) will be running in the basin at all times while work is being performed in the basin area. Figure 1-5 shows the locations of air sampling and monitoring devices.

The following section provides specific descriptions of the tasks in sequential order.

Task 1 - North Cask Pit

105N Basin Stabilization activities will begin in the North Cask Pit. In conjunction with the hardware removal from the North Cask Pit, hydroscrubbing of the exposed basin surface just above the water line (commonly referred to

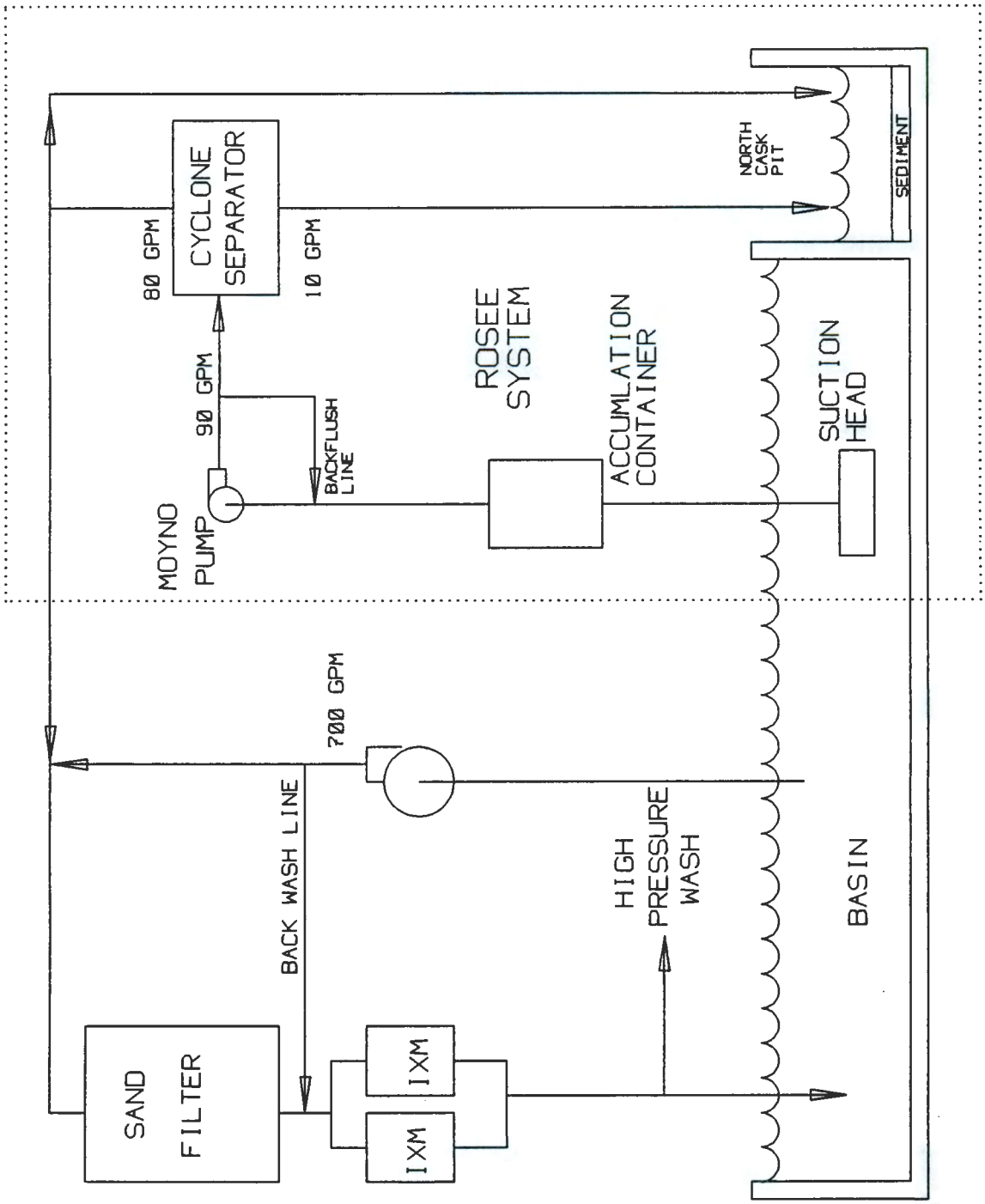
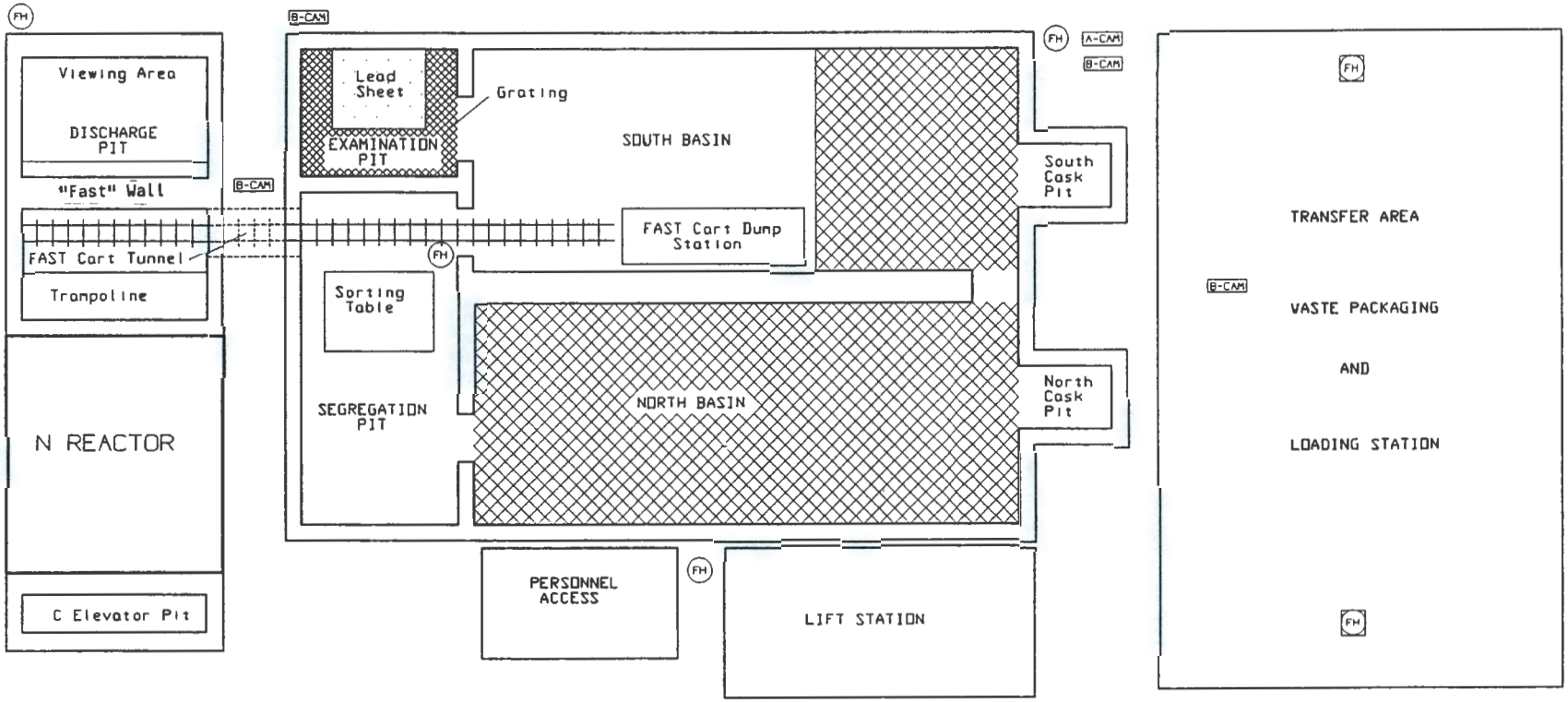


Figure 1-4. Sand Filter and Remotely Operated Sediment Extraction Equipment System Configuration.



A-CAM Alpha Continuous Air Monitor

B-CAM Beta Continuous Air Monitor

FH Fixed Head Area Air Sampler located at personnel working level

FH Fixed Head Area Air Sampler located 15 feet in overhead

Figure 1-5. Location of 105N Basin Air Sampling Instrumentation.

as the "bathtub ring") will be initiated in the Viewing Area of the Discharge Pit area. Hydroscrubbing will be accomplished by the use of either a hand-held wand, or a fixed apparatus attached to one of the overhead cranes.

Removal activities will consist of relocating the debris and hardware to the North Basin. This will be accomplished by personnel standing on the platform above the pit using remote handling tools, such as the existing tongs and hooks used during routine operations. After all items have been removed from the pit, sludge will be transferred to the North Basin using the ROSEE system so that inspection for SNM can be performed. The ROSEE system will be operated by personnel above the basin using remote controls and an overhead cranes. Inspection for SNM will be accomplished using remotely operated video equipment and radiation detection instrumentation operated from a position above the pit.

The opening from the North Cask Pit to the North Basin will be closed and sealed by the installation of a pre-fabricated coffer dam to contain sediment inside the pit. The North Cask Pit will be used thereafter as a repository for all sediments moved from other areas of the basin. Once the North Cask Pit has been thoroughly cleaned, a cover will be placed over the pit. The cover is intended to provide a platform for the ROSEE system cyclone separators that will discharge into the North Cask Pit, as well as to provide a radiation shielding barrier between the sediment accumulation in the North Cask Pit and the work area around the pit.

When sediment removal activities begin, the water level in the covered North Cask Pit will be lowered approximately 8 feet by pumping water from the pit to the North Basin. Lowering the water level in the pit will provide room for the water discharged from the cyclone separators and for the accumulation of sediment. The water level in the North Cask Pit will be lowered periodically to accommodate the increase in volume created during sediment relocation activities.

Task 2 - South Cask Pit

Concurrently with installation of the North Cask Pit coffer dam, stabilization activities will also occur in the South Cask Pit. The walls of the pit will be hydroscrubbed to reduce the dose rate contribution from the "bath tub" ring for personnel working around the pit. All debris and hardware found in the pit will be relocated to the South Basin for removal with South Basin activities. Once the coffer dam is installed in the North Cask Pit, sludge and sediment will be transferred to the North Cask Pit using the ROSEE system. The South Cask Pit will then be inspected to certify that no SNM is present. The South Cask Pit will henceforth be used exclusively for loading high-dose-rate items into submerged shipping casks.

Task 3 - North Basin

The walls of the North Basin will be hydroscrubbed to reduce the dose rate contribution from the "bath tub" ring for personnel working around the perimeter of the basin. Before removing hardware and debris, the canister cleaning and crushing equipment will be installed in the North Basin. On completing installation of the canister cleaning equipment, all loose hardware and debris located on top of the cubicle lids will be removed from the basin

to gain access to the cubicle lids. Fuel baskets will either be placed in the burial or shipping containers intact or cut into small pieces underwater for purposes of volume reduction. All materials will be sorted under water by radiation dose rate and only items showing acceptable radiation levels will be raised out of the basin water.

Canisters will be removed from the cubicles and cleaned as described in Section 1.2.1 to reduce the levels of radioactive contamination. The canisters then may be crushed to minimize burial volume, or left intact for shipment to an authorized DOE facility for smelting and volume reduction. The canisters will be raised out of the water, rinsed, and allowed to drain over the basin before loading into designated burial/shipping containers.

Sediment will be removed from the cubicles using the ROSEE system and relocated to the North Cask Pit. After the sediment activities are complete, the North Basin will be inspected to certify that no SNM is present.

Task 4 - Segregation Pit

Prior to the start of cleaning activities in the Segregation Pit, the wall surfaces in the pit along the water line will be hydroscrubbed to reduce the dose rate contribution in this area. After hydroscrubbing, equipment and debris will be removed. Major pieces of equipment located in the Segregation Pit include the fuel basket dumper, sections of process tubes, the sorting table, and the fuel canister loading equipment. These large items will be disassembled or cut into smaller pieces for volume reduction. All items, debris, equipment, and hardware (excluding high-dose-rate material) will be cleaned using high-pressure water, raised out of the water, rinsed, drained, and placed in designated burial or shipping containers. The "Fast" Cart track, which runs through the Segregation Pit, will be cleaned in place and left to be stabilized with the rest of the basin.

After all loose items, equipment, and hardware are removed from the Segregation Pit, sediment will be relocated using the ROSEE system, *then* the Segregation Pit will be inspected to certify that no SNM is present.

Task 5 - South Basin

The walls of the South Basin will be hydroscrubbed to reduce the dose rate contribution from the "bath tub" ring for personnel working around the perimeter of the basin. Upon completion of hydroscrubbing, the canister cleaning and crushing equipment will be relocated to the South Basin. In addition, the basin encapsulation equipment, located in the southwest corner of the basin, will be disassembled to the extent possible using existing basin tools. Components removed from the encapsulation equipment will be disposed of as radioactive waste, however the remainder of the equipment will be left in place and stabilized with the rest of the basin. Canisters will be removed from the cubicles and cleaned to reduce the levels of radioactive contamination.

Sections of process tubes found at other locations in the basin will be relocated to the South Basin where they will be cut with a mechanical tube cutter, which is currently located in the basin. If the tube cutter is not

operational, the process tubes will be cut into smaller sections using a guillotine device (slow speed hydraulic shears). The cut tube sections will be loaded underwater into shielded burial casks as radioactive waste.

Sediment relocation will be initiated in conjunction with hardware and debris removal. Canister cleaning and crushing will be conducted along with disassembly of the "Fast" Cart Dump Station. All materials will be sorted under water by dose rate and only items showing acceptable radiation levels will be raised out of the basin water. When all the canisters are removed from the basin, the canister cleaning and crushing equipment will be lifted out of the basin, rinsed, and disposed of or stored.

Sediment will be relocated to the North Cask Pit from the cubicles using the ROSEE system. The South Basin will then be inspected to certify that no SNM is present.

Task 6 - Examination Pit

Prior to the start of cleaning activities in the Examination Pit, the wall surfaces in the pit along the water line will be hydroscrubbed to reduce the dose rate contribution in this area. After hydroscrubbing, equipment and debris will be removed. The task of cleaning out the Examination Pit, located east of the South Basin, will begin with dismantling and removing the lead shielding on the grating above the fuel examination station. The lead will be packaged in DOE/U.S. Department of Transportation (DOT)-approved containers for final disposition.

After all loose items, equipment, and hardware are removed from the Examination Pit, and the sediment is relocated using the ROSEE system, the Examination Pit will be inspected to certify that no SNM is present.

Task 7 - Discharge Pit

Prior to the start of cleaning activities in the Discharge Pit, the wall surfaces in the Viewing Area of the Discharge Pit will be hydroscrubbed along the water line if not already completed during Task 1. After hydroscrubbing, equipment and debris will be removed. All items, debris, equipment, and hardware (excluding high-dose-rate material) will be cleaned using high-pressure water, raised out of the water, rinsed, and placed in designated burial or shipping containers. High-dose-rate items will be moved under water and placed in the burial cask in the South Cask Pit. The "Fast" Cart track will be cleaned in place and left to be stabilized with the rest of the basin. The "Fast" Carts, located in the tunnel running between the Discharge Pit and the Segregation Pit, will be removed, cleaned with high-pressure water, and cut up for waste volume reduction prior to being placed in burial containers or loaded into shipping containers for smelting.

The trampoline, which is located in the Reactor Area, will not be removed as part of stabilization activities because of ALARA concerns. Inspection for fuel pieces under the trampoline will be performed with underwater radiation detection instrumentation and underwater video equipment.

A sheet metal cover will be placed over the Viewing Area of the Discharge Pit to seal off the Reactor Area from the fuel storage area. The "Fast" Cart tunnel will also be sealed after the water is removed to accomplish this separation.

After all loose items, equipment, and hardware are removed from the Discharge Pit, and the sediment is relocated to the North Cask Pit, the Discharge Pit will be inspected to certify that no SNM is present.

1.2.1 Underwater Activities

Stabilization of the basin will consist of the following underwater activities.

Cleaning the surface of hardware and debris underwater using manual scrubbing, high-pressure water, or a mechanical cleaning station.

Underwater cleaning of equipment, hardware, and debris found in the basin will involve personnel standing on the working platforms above the basin using handheld, remotely operated tools and controls for the overhead cranes. These tools include fuel tongs and hooks. Personnel will grasp small items with the tongs, or attach the item to a hook, so that it can be positioned near the high-pressure water nozzle. Large items will be repositioned with the remotely operated overhead cranes. Personnel will then direct the high-pressure water on the item to be cleaned. In some cases, a wire brush on the end of a pole may be used to scrub items that may be encrusted with loose scale. Fuel canisters will be cleaned at a mechanical cleaning station positioned underwater. The item will then be placed near an underwater radiation detector to determine its associated radiation reading to assess handling requirements.

The canister cleaning station is a specially designed hydraulically actuated automated cleaning device. When canisters are removed from the cubicles, they will be placed on the cleaning station. The canisters are inverted and placed over the rotating brushes. The external surface of the canisters will be washed with high-pressure water. The outside surface of the aluminum canisters will be scraped to remove loose scale at the canister cleaning station. The canisters then will be removed from the canister cleaning station and either moved to the canister crushing station, or removed from the basin for shipment to an offsite DOE facility.

The following table lists the material in the basin that is scheduled to be cleaned and the cleaning method.

The actual quantity of debris in the basin is not known, however the type of material expected to be encountered includes hoses, wire, "buggy springs" (items that were part of the fuel element assembly), and items that accidently fell into the basin, such as, gloves, plastic, and hand tools.

105N Basin Material Inventory.

Material	Quantity	Cleaning method
Cubicle Lids	536	High-pressure water.
Fuel Shipping Capsules	45	High-pressure water.
Fuel Storage Canisters	1500	Mechanical cleaning station.
Canister Covers (Caps)	140	High-pressure water.
Fuel Baskets	146	High-pressure water and manual scrubbing.
Fuel Spacers	40	High-pressure water.
"Fast" Cart Dumping Station	3	High-pressure water and manual scrubbing.
Fuel Sorting Table and Packaging Sta.	1	High-pressure water and manual scrubbing.
Fuel Dump Baskets	1	High-pressure water and manual scrubbing.
Cubicles	536	Vacuum using ROSEE system.
FAST Carts	3	High-pressure water.
Debris	unknown	High-pressure water and manual scrubbing.

Using ROSEE to vacuum sediment and small debris, separate out the water and transfer the sediment and debris to the North Cask Pit.

The components of the ROSEE system include a submersible tractor equipped with lights, video camera, robotic arm, radiation detector, and vacuum head; a cubicle-cleaning vacuum head; a trash separator with radiation detector to collect the large debris from the vacuum stream; a positive displacement pump that operates at 90 gallons per minute; and two parallel cyclone separators. Figure 1-6 shows the system configuration and layout.

The cyclone separators will direct a flow, approximately 10 gallons per minute, consisting of sediment particles 10 microns and larger into the North Cask Pit. The remaining flow from the cyclone separators will pass through a backwashable filter unit to remove particles 10 microns and smaller. The backwash from the filter unit will be discharged into the North Cask Pit.

The tractor component will be positioned in the basin with one of the overhead cranes. Operation of the tractor will be performed by personnel using a remote control station. The vacuum attachment on the tractor can be operated through the vacuum head built into the bottom of the tractor, or the robotic arm can remove the vacuum hose and manipulate it in front of the tractor unit. The radiation detector on the robotic arm will be used to identify potential fuel pieces or fragments found in its path. The tractor unit will traverse the tops of the cubicles or basin floor until all visible sediment is removed.

Canister removal and cubicle cleaning will be handled in the following sequence. First, the cubicle lids will be vacuumed with the ROSEE system to remove sediment from the tops of the cubicles. Each cubicle will then be opened to gain access to the canisters. When the canisters are removed from a cubicle, the cubicle lid will be placed back in the closed position. When all the canisters have been removed, the cubicles will be cleaned out with the ROSEE system. This will be accomplished by cleaning and inspecting one row of cubicles at a time. For example, cleaning will start at row 1, cubicle 1. The lid will be removed and placed in a fuel basket. The ROSEE suction head will be lowered into the cubicle, then withdrawn. Next, the video camera will be lowered into the cubicle to inspect for SNM. This sequence will be repeated until all cubicles in the row, and all cubicles in the basin, have been cleaned and inspected. Once a fuel basket is full of lids, it will be moved under water to the North Basin where the lids will be hydroscrubbed underwater before they are brought out of the basin and placed into designated burial/shipping containers.

Crushing fuel canisters under water at a hydraulic-powered crushing station.

The approximate number of canisters in both the North and South Basins are estimated to be 1,500; 1,200 of which are of the stainless steel type and 300 of the aluminum type. The preferred method of canister disposal is to ship them offsite for volume reduction at a DOE-approved smelting facility. If the preferred method is not selected, the canisters will be crushed in the basin and packaged for disposal on the Hanford Site.

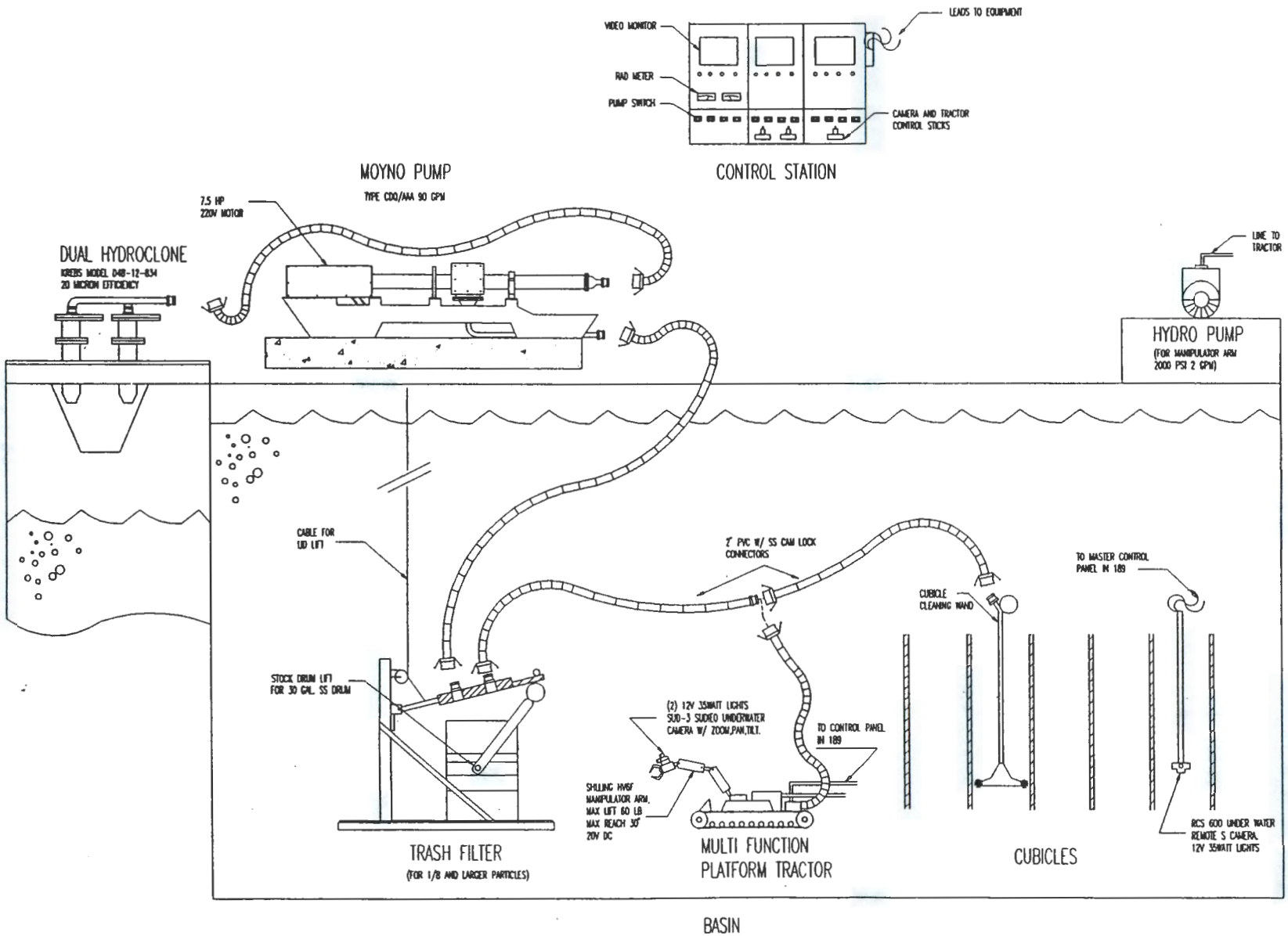


Figure 1-6. Remotely Operated Sediment Extraction Equipment System Configuration.

The canister crushing station is a specially designed hydraulic ram system that can crush both stainless steel and aluminum canisters. The crushing station will be lowered into the basin and set on top of the cubicles where the canisters are to be removed. One canister at a time will placed in the crushing device. The hydraulic ram will compress the canister to approximately 17 percent of its original volume. The crushed canister will be removed from the device and removed from the basin for disposal.

Cutting large items under water, such as process tubes, sorting table, fuel baskets, into smaller sections for packaging.

All cutting activities will be performed under water. The following table identifies the material to be cut and the method of cutting.

Volume Reduction Methods.

Item	Method	Description
Process Tubes	Tube Cutter/ Guillotine	Tube cutter currently in basin. It is air operated with the exhaust plumbed to atmosphere. The Guillotine is a hydraulic shear and will be used under water.
Fuel Baskets	Plasma Arc	Will be cut into smaller sections under water.
Sorting Table and "Fast" Carts	Plasma Arc	Will be cut into smaller sections underwater.

Process tubes located in the basin are considered high-dose-rate items, and therefore, will not be raised out of the basin water. Instead, they will be placed into the underwater cask and buried as radioactive waste on the Hanford Site. Because the process tubes are of various lengths, they will have to be cut to fit in the burial cask. The process tubes will be cut with the existing tube cutter currently located in the basin. The tube cutter is an air-actuated device that operates in the same manner as standard pipe cutters. If the tube cutter in the basin is found to be inoperable, a "guillotine" device will be placed in the basin to cut the process tubes. This device has a hydraulic-actuated blade that shears off sections of tubing.

Moving high-dose-rate items underwater and packaging in designated under water burial/shipping containers.

All high-dose-rate material and items will be handled, moved, and packaged under water. Current survey data have identified most of the high dose rate items. However, many items or sources of high radiation levels could not be identified because of the conjunction of material in some of the pits. Movement of small items will generally be performed by personnel using

remote handling tools and keeping the object under water. Larger items will be attached to a crane hook and moved under water. Figure 1-7 depicts the flow path for movement of high-dose-rate items under water.

1.2.2 Above Water Activities

Stabilization of the basin will consist of the following above water activities.

Lifting hardware and large debris from the basin water and packaging in designated burial/shipping containers.

Low-dose-rate items will first be hydroscrubbed under water to remove loosely attached surface contamination, then raised out of the water and rinsed with filtered basin water. Rinsing activities will not add additional water to the basin. Rinse water for the decontamination activities will use the discharge from the basin filtration system installed for maintaining water clarity.

Smaller items, such as, tools, hoses, cables, small metal components, and debris, will be placed in an underwater accumulation container. When the accumulation container is full, the contents will be used to fill the voids in the underwater shipping cask located in the South Cask Pit. These shipping casks were specially designed to transport irradiated fuel to various locations within the Hanford Site.

High-dose-rate items will not be raised out of the water. These items will be packaged under water in approved shipping/burial casks for disposal. Smaller objects with high-dose-rates will also be used to fill voids in the underwater casks. When the cask is full, it will be raised out of the basin and washed off with filtered basin water as it is withdrawn. The cask will be drained using built-in drain valves and set down at a designated location just west of the cask pit. After the cask has dried, it will be painted to fix any residual contamination on its surface. The cask will then be relocated to a designated staging area in the basin and covered with plastic until it is loaded onto a flat-bed truck for transportation to an authorized disposal facility.

It is important to note that all material and debris not easily recognizable will be inspected with underwater video equipment to determine its identity. Therefore, all fuel assemblies, if discovered, and fuel chips or fragments will be placed in a designated canister located in the basin. Upon final inspection of the basin, all canisters holding fuel will be loaded into an underwater shipping cask, which will be transported to the 100-K fuel storage basin for final disposition.

Wash basin walls with high-pressure water. This activity will be conducted out of and above the water.

The purpose of this activity is to reduce the dose rates created by radioactive material plating out, or leaching into the basin wall surface along the water line, often referred to as the "bathtub ring" effect. This activity will be performed by personnel using a high-pressure lance.

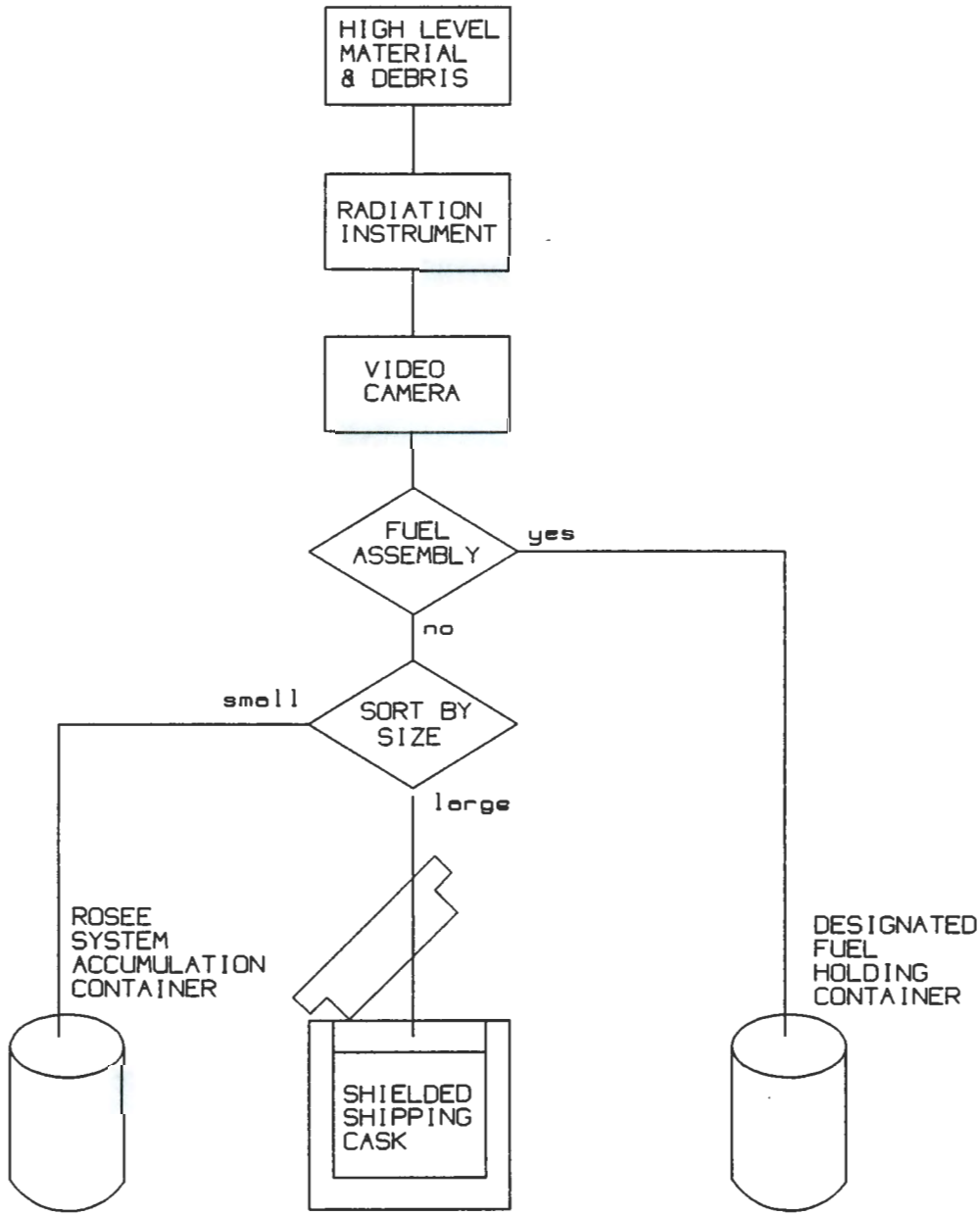


Figure 1-7. High Level Radioactive Material Flow Diagram.

Personnel will stand on the working platform above the edge of the basin and direct the blast of the water lance along the basin wall surface. The nozzle of the lance will be held approximately 6 inches from the wall surface and directed down toward the basin water at an angle of approximately 33 degrees. The cleaning rate is expected to be approximately 3 feet per minute.

1.3 FACILITY IDENTIFICATION

The basin will utilize the existing 116N Stack. It is registered with the Washington State Department of Health (DOH) in the "Source Registration for Radioactive Air Emissions". The stack identification number is R DNR 001 002 A.

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2.0 SOURCE INFORMATION

This section provides detailed information regarding the source and quantity of airborne radionuclide emissions resulting from the proposed basin stabilization activities.

2.1 HISTORICAL BACKGROUND

This section describes the past activities that produced the current radionuclide inventory in the basin, and contrasts past operations and planned operations and their relative impacts on air emissions.

2.1.1 Radionuclide Sources

105N Basin was designed as a place to package and temporarily store spent fuel elements and irradiated fuel spacers discharged from the N Reactor. Fuel handling and storage operations were conducted in seven interconnected basins with a common pool of water, one other remote basin ("C" Pit), and a water lift station. During N Reactor operation, underwater transfer carts moved the spent fuel elements from their discharge point at the rear face of the reactor to the storage basin, where they were sorted by enrichment and placed in storage canisters. The filled canisters were moved by bridge cranes to storage cubicles, formed by a lattice of borated concrete walls. The basin began operation in 1963 and ceased storing spent fuel in 1989, when all fuel was transferred to the 100-K fuel storage basins.

A significant amount of radioactive material was deposited in the basin as a result of the fuel handling and storage operations. Sources of radionuclides included the following:

- Irradiated uranium fuel elements with damaged cladding that introduced spent fuel particles (including transuranic isotopes and fission products) into the basin water and sediment
- Irradiated lithium targets with damaged cladding that introduced tritium into the basin water
- Corrosion of irradiated reactor hardware (e.g., spacers, buggy springs)
- Hardware and debris with surficial deposits of contamination.

Fuel elements and tritium targets are no longer stored in the basin. However, small fragments from damaged fuel elements are potentially present. In addition, an observed increase in tritium activity in the basin water indicates that a lithium target or fragments of targets remain in the basin. The current inventory of radionuclides in the basin is presented in Section 2.2.1.

2.1.2 Past Air Emissions

The proposed stabilization activities are expected to result in fewer emissions of radionuclides to the atmosphere than occurred prior to 1988, when the N Reactor was operating and the basin was in use. A quantitative comparison of historical emissions and emissions expected during stabilization from the basin would not be meaningful, because past monitoring results were probably biased low due to the absence of isokinetic sampling. However, a qualitative comparison of past and planned basin operating conditions (Table 2-1) indicates that expected emissions during stabilization will be substantially lower than past emissions.

2.2 SOURCE TERM DESCRIPTION

This section provides detailed information regarding the source and quantity of airborne radionuclide emissions resulting from the proposed basin stabilization activities. Represented are the estimates of the total radionuclide inventory in the basin, the mechanisms whereby radionuclides are transported to the basin air space, the source term in the air space, and the air emissions. The expected annual average release rates of the radionuclides emitted by the basin were calculated using good engineering judgement (GEJ), which provide a more realistic description of the expected potential emissions.

2.2.1 Radionuclide Inventory of 105N Basin

Radionuclides are assumed to be present in the basin in four physical forms as follows:

- Solute and suspended solids in basin water
- Basin sediment and wall deposits
- Surface deposits on hardware
- Fragments of irradiated fuel elements and lithium targets.

The radionuclide inventory associated with each physical form, decay-corrected to February 1994, is presented in Table 2-2. The basis for the inventory is discussed below.

Water. The basin Complex contains approximately 4.1 million liters (1.08 million gallons) of water (Appendix A-1). A grab sample of the water is collected monthly and analyzed. Suspended solids measured in the water samples typically constitute about 5 parts per million by weight, and the activity in these solids is included in the reported analyses. The water activity presented in Table 2-2 is based on the mean activity of samples collected from August 1992 to August 1993 at the upper 95 percent confidence limit[1].

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[1] The confidence interval is a range on either side of a sample mean. A statistical evaluation was done based on the mean, the standard deviation of the data, and the number of samples. At the upper 95 percent confidence limit in a one-sided test, there is a 95 percent probability that the true mean is equal to or less than the limit.

Table 2-1. Comparison of Past and Planned Storage 105N Basin Operating Conditions and Impact on Air Emissions.

Past operating condition	Future operating condition	Impact of changed condition
Air from the basin pool and transfer area was exhausted directly to the atmosphere without emissions controls.	Air from the basin is exhausted through the N Reactor Zone I ventilation system, which includes a bank of HEPA filters, then discharged via the 116N Stack.	Most of the radionuclide source term occurs as solid particulate. HEPA filters remove particulate with an installed efficiency of 99.95 percent.
N Reactor was in operation.	N Reactor has not operated since January 1987.	Radionuclides with half-lives of less than 1 year have decayed to less than 1 percent of their original activity.
Fuel was being actively discharged to and removed from the basin. Up to 1,250 metric tons of irradiated uranium fuel were stored in the basin at one time.	Fuel elements were removed from the basin in 1989.	The radionuclide inventory of the basin is several orders of magnitude lower now than during active operation.
Major maintenance activities were ongoing in Zone I of the reactor during 1987 and 1988.	Activity in Zone I of the reactor will be minimal during the basin stabilization project.	Maintenance activities added to the air stream treatment requirements of the ventilation system. Those requirements will not be present.
Clean water was added to the basin continuously during active operations; the overflow was discharged to a crib.	No water is added to the basin, and there is no overflow to outside areas.	The water additions diluted basin water activity. Without the additions, there has been an observed increase in tritium activity in the water, which increased tritium emissions associated with evaporation.

HEPA = high-efficiency particulate air.

Table 2-2. 105N Basin Radionuclide Inventory (as of February 1994).

Nuclide	Half-life (years)	Water		Sediment		Hardware		Fuel elements total Ci****	Total Ci	Notes
		µCi/L	Total Ci*	µCi/L	Total Ci**	µCi/L	Total Ci***			
Am-241	4.32 E+02	8.99 E-05	3.69 E-04	2.34 E-01	4.68 E-01	5.98 E-02	6.10 E+00	3.44 E+00	1.00 E+01	a,b,c
Ce-144	7.80 E-01			1.79 E-02	3.58 E-02	2.13 E+00	2.17 E+02	1.22 E+02	3.39 E+02	b,d,e
Co-60	5.26 E+00	7.57 E-04	3.10 E-03	5.82 E+00	1.16 E+01	1.06 E-01	1.08 E+01		2.24 E+01	a,b,c
Cs-134	2.05 E+00	2.26 E-03	9.26 E-03	3.87 E-03	7.74 E-03	9.78 E-03	9.98 E-01	4.73 E+00	5.75 E+00	a,f
Cs-137	3.00 E+01	1.43 E+00	5.85 E+00	5.95 E-01	1.19 E+00	1.50 E+00	1.53 E+02	7.28 E+02	8.88 E+02	a,b,c
H-3	1.22 E+01	3.90 E+01	1.60 E+02					1.40 E+02	3.00 E+02	a,g,h
K-40	1.40 E+09	2.04 E-03	8.37 E-03						8.37 E-03	a,g
Mn-54	8.30 E-01	2.40 E-04	9.84 E-04	1.56 E-02	3.12 E-02	2.90 E-04	2.95 E-02		6.18 E-02	a,b,c
Pu-238	8.64 E+01	3.62 E-05	1.48 E-04	5.32 E-01	1.06 E+00	1.33 E-01	1.36 E+01	7.83 E+00	2.25 E+01	a,c,i
Pu-239/240	2.40 E+04	2.47 E-04	1.01 E-03	3.25 E+00	6.51 E+00	8.36 E-01	8.52 E+01	4.79 E+01	1.40 E+02	a,b,c
Pu-241	1.44 E+01	5.24 E-03	2.15 E-02	6.90 E+01	1.38 E+02	1.77 E+01	1.81 E+03	1.02 E+03	2.96 E+03	j
Ru-106	1.01 E+00			2.63 E-02	5.25 E-02	6.14 E-01	6.26 E+01	3.51 E+01	9.78 E+01	b,d,k
Sb-125	2.71 E+00	3.48 E-03	1.43 E-02	6.96 E-01	1.39 E+00	9.48 E-02	9.67 E+00		1.11 E+01	a,c,l
Sr-90	2.77 E+01	3.28 E+00	1.34 E+01	6.56 E+02	1.31 E+03	1.46 E+00	1.49 E+02	7.06 E+02	2.18 E+03	a,l,m
U-234	2.47 E+05			9.69 E-03	1.94 E-02	2.49 E-03	2.54 E-01	1.42 E-01	4.16 E-01	n,o
U-235	7.10 E+08			1.76 E-03	3.51 E-03	4.51 E-04	4.60 E-02	2.58 E-02	7.54 E-02	n,o
U-238	4.50 E+09			7.88 E-03	1.58 E-02	2.02 E-03	2.06 E-01	1.16 E-01	3.38 E-01	n,o

*Assumed water volume is 4.1 million liters (1.1 million gallons).
**Assumed mass of centrifuged sediment is 2,000 kilograms.
***Assumed mass of hardware is 102,000 kilograms.
****Assume 1 canister (700 pounds) of irradiated uranium fuel and 1 irrediated lithium target.
a-Water activity based on samples collected from 8/92 to 8/93; decay-corrected to 2/94.
b-Sediment activity based on samples collected in 1987 and centrifuged; decay-corrected to 2/94.
c-Hardware activity based on essays of fuel canisters; decay-corrected to 2/94. Transuranic assumed to be Am-241, Pu-238, and Pu-239,240.
d-Below detection limits in water analyses.
e-Hardware activity based on ratio of Ce-144 to Pu-239 in 10-year-old fuel.
f-Sediment and hardware activity based on ratio of Cs-134 to Cs-137 in 10-year-old fuel.
g-Occurs as soluble species; sediment and hardware activity assumed negligible.
h-Equivalent of one tritium-bearing lithium target assumed to contain a current inventory of 140 curies tritium.
i-Sediment activity based on ratio of Pu-238 to Pu-239 in 10-year-old fuel.
j-Water, sediment, and hardware activity based on ratio of Pu-241 to Pu-239 in 10-year-old fuel.
k-Hardware activity based on ratio of Ru-106 to Pu-239 in 10-year-old fuel.
l-Sediment activity based on water activity, assuming 5 parts per million by weight solids suspended in water and all water activity attributable to suspended solids.
m-Hardware activity based on ratio of Sr-90 to Cs-137 in 10-year-old fuel.
n-Not analyzed in water samples; assumed negligible due to low solubility.
o-Sediment and hardware activity based on ratios to Pu-239 in 10-year-old fuel.

Sediment. A layer of sediment, ranging from 0 to 3 inches thick (Subrahmanyam 1988), covers the bottom of the basin. It is composed of fine silt and dust, insect matter, algae, corrosion products, and small debris. Given the thickness of the sediment layer, the total volume of settled solids in the basin has been estimated at 1.16×10^4 liters (410 cubic feet) (Subrahmanyam 1988). Based on laboratory measurements of the density of the settled solids versus centrifuged solids, the total volume of centrifuged solids in the basin has been estimated at 1.74×10^3 liters (61 cubic feet) and the mass of centrifuged solids has been estimated at 2,000 kilograms (Subrahmanyam 1988). For conservatism in calculating the source term, the volume of solids has been assumed to be 70 cubic feet.

In 1987, nineteen samples of the sediment were collected and analyzed from different locations throughout the basin (Subrahmanyam 1988). Sediment activity was reported based on centrifuged solids. Because reactor operations ceased in January of 1987, and all fuel was removed from the basin by 1989, the 1987 data (with appropriate decay corrections for the intervening years) are expected to be representative of the current source term in the sediment. The sediment activity presented in Table 2-2 is the mean activity of the centrifuged sediment samples at the upper 95 percent confidence limit[2].

Hardware. A variety of hardware and debris is present in the basin. The types of hardware and estimated volumes, masses, and surface areas are presented in Table 2-3.

Only a small fraction of the total hardware, primarily process tubes, fuel spacers, and buggy springs, was irradiated in the reactor and would be expected to contain activation products. The activation products are an integral part of the metal of the hardware and are not available to the air.

The majority of contamination associated with the hardware is present in the corrosion and other deposits that have accumulated on the surface of the hardware. In 1990, 12 aluminum fuel canisters were removed from the basin. All 12 were assayed for transuranic isotope activity and two were assayed for gamma radionuclides as part of an evaluation of cleaning techniques. Activity was reported as curies per unit mass of the canisters (Appendix A-2).

The level of contamination associated with the aluminum canisters is expected to be higher than for the stainless steel canisters or other hardware in the basin. The aluminum canisters comprise approximately 300 of the 1,500 canisters in the basin (WHC 1993b). The other 1,200 canisters, which are constructed of stainless steel, and most of the other hardware in the basin is constructed of either stainless or carbon steel. Aluminum develops a relatively thick, porous corrosion surface that adheres tightly and traps contaminants. The aluminum itself is a relatively porous material. In contrast, stainless steel does not develop a corrosion surface, and film deposits wash off readily. In addition, the aluminum canisters are an older design that includes drainage holes and a mesh screen in the bottom on which

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[2] The confidence interval is a range on either side of a sample mean. A statistical evaluation was done based on the mean, the standard deviation of the data, and the number of samples. At the upper 95 percent confidence limit in a one-sided test, there is a 95 percent probability that the true mean is equal to or less than the limit.

Table 2-3. Hardware Inventory to be Removed from 105N Basin.

Material	Quantity	Dimensions (in.)	Volume/item (ft^3)	Total volume (ft^3)	Surface area/item (ft^2)	Total surface area (ft^2)	Total mass (kg)	Notes
Cubicle lids	536	0.37 x 23.5 x 27.5	1.38 E-01	7.42 E+01	9.24 E+00	4.95 E+03	6.69 E+03	a,b,c
Fuel storage canisters	1,500	9.5 x 18.4 x 29	2.93 E+00	4.40 E+03	1.37 E+01	2.05 E+04	3.12 E+04	a,b,c
Canister covers	140	.79 x 7.98 x 5.13	1.87 E-02	2.62 E+00	7.12 E-01	9.97 E+01		a,b,c
Fuel baskets	146	40 x 40 x 40	3.70 E+01	5.41 E+03	5.56 E+01	8.11 E+03	3.59 E+04	a,b,c,d
Fuel spacers	40	2.5 OD x 20	5.68 E-02	2.27 E+00	1.09 E+00	4.36 E+01	3.60 E+01	a,b
Dump carts	3	34 x 37 x 84	6.12 E+01	1.83 E+02	1.00 E+02	3.01 E+02	5.44 E+03	a,b,c,e
Sorting tray	1	89 x 123 x 166	1.05 E+03	1.05 E+03	6.41 E+02	6.41 E+02	e	a,b,c,e
Dump basket	1	47 x 49 x 97	1.29 E+02	1.29 E+02	1.61 E+02	1.61 E+02	e	a,b,c,e
Miscellaneous debris	100		2.85 E+00	2.85 E+02	1.21 E+01	1.21 E+03	1.81 E+04	b,f
Process tubes	20	4 OD x 120	8.73 E-01	1.75 E+01	1.05 E+01	2.09 E+02	3.93 E+03	a,c,g
Fuel shipping capsules	45	3.5 OD x 28	1.56 E-01	7.02 E+00	2.14 E+00	9.62 E+01	7.62 E+02	a,c,h
Total				1.16 E+04		3.63 E+04	1.02 E+05	

a-Quantity and dimensions from WHC-SP-0615, Rev. 4 (WHC 1993b).
b-Total mass from WHC-SP-0615, Rev. 4 (WHC 1993b).
c-Volume and surface area calculated from dimensions.
d-Fuel baskets assumed to have five sides, open at top.
e-Masses of dump cart assemblies, sorting tray, and dump basket are combined.
f-Number of items and dimensions not available. Number of items assumed to be 100. Average shape assumed to be cubic. Surface area of a cube = 6 x (volume to the 2/3 power).
g-Density of zircalloy in process tubes is 0.237 pounds/cubic inch. Zircalloy assumed 1/4-inch thick.
h-Density of aluminum in shipping capsules is 0.1 pound/cubic inch. Aluminum assumed 1/4-inch thick.
OD = outside diameter.

material, including fuel fragments and sediment deposits, has collected over the years. The stainless steel canisters have smooth bottoms without holes or screens, so material has not accumulated.

Because other hardware in the basin has not been assayed, the activity-per-unit-mass data for the aluminum canisters have been assumed to apply to the rest of the hardware as well. Table 2-2 presents the mean activity of the aluminum canisters at the upper 95 percent confidence interval[3]. The total hardware activity presented is based on the total mass of hardware in the basin. However, for the reasons stated previously, it should be understood that use of the aluminum canister assay data is a conservative case.

Irradiated Uranium Fuel Fragments. Based on visual inspection and past SNM records, the basin has been declared free of any known quantities of irradiated uranium fuel. However, for planning purposes, it has been assumed that some fuel element fragments will be found during basin stabilization activities. The volume of fuel found is assumed to fit into one fuel canister. The assumption is a conservative estimate. It is anticipated that if any fuel fragments are found, the quantity will be less than one canister. However, one canister will be needed to handle any fuel fragments. Therefore, for ease in calculating potential emissions, one canister volume was used. A canister is designed to hold 14 fuel element assemblies, with one fuel element assembly weighing approximately 50 pounds, giving a total approximate fuel weight of 700 pounds.

The fuel fragments result from fuel that was discharged from N Reactor between 7 and 31 years ago. To calculate the source term associated with the fragments, the following assumptions were made:

- The majority of the N Reactor fuel was 0.95 weight percent enriched with ^{235}U isotope. A small amount of natural uranium fuel was also irradiated. Because more highly enriched fuel results in higher radionuclide levels, the fragments are assumed to result from enriched fuel.

- Fuel can be irradiated to either weapons-grade assay levels (^{240}Pu levels of 6 percent) or fuels-grade assay levels (^{240}Pu levels of 12 percent). Fuels-grade levels are reached through longer irradiation times and result in higher radionuclide levels. Because the source of the fragments is unknown, it is assumed that they are fuels-grade.

- A 10-year average decay time was assumed for the fuel fragment inventory.

The radionuclide inventory associated with the fuel fragments is provided in Table 2-2.

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[3] The confidence interval is a range on either side of a sample mean. A statistical evaluation was done based on the mean, the standard deviation of the data, and the number of samples. At the upper 95 percent confidence limit in a one-sided test, there is a 95 percent probability that the true mean is equal to or less than the limit.

Irradiated Lithium Targets. Visual inspection has not revealed the presence of tritium-bearing lithium targets. However, basin water samples show a continual increase in tritium activity. The increase has been attributed to the presence of a lithium target or target fragments in quantities approximating one target. Target failures during past operations resulted in the release of approximately 140 curies of tritium per target. For inventory purposes, it is assumed that a target or fragments of targets contain 140 curies of tritium.

2.2.2 Chemical and Physical Forms of Releases

Airborne releases from the basin stabilization project will be primarily particulate forms. The dominant chemical species are likely to be oxides and hydroxides of the radionuclides. Salts (e.g., chlorides and nitrates) might also occur. The only significant volatilized radionuclide will be tritium.

2.2.3 Emissions Release Rates

The following sections provide expected annual emissions based on (1) GEJ and (2) the methodology prescribed in 40 *Code of Federal Regulations* (CFR) 61, Appendix D.

2.2.3.1 Good Engineering Judgement. This section provides the engineering evaluations and assumptions used in determining the GEJ emissions estimates.

Stabilization of the basin will consist of several activities. Activities performed under water include the following:

- Cleaning the surface of hardware and debris using manual scrubbing, high-pressure water, or a mechanical cleaning station
- Crushing fuel storage canisters at a hydraulic-powered crushing station
- Placing smaller items inside larger items to reduce waste volumes
- Cutting large items (e.g., process tubes, a sorting table, fuel baskets) into smaller sections for packaging. Potential cutting methods include plasma torch, hydraulic shears or nibblers, and tube cutters
- Using ROSEE to vacuum sediment and small debris from the basin, separate out water, and transfer the sediment and debris to the North Cask Pit.

Above water activities include the following:

- Lifting hardware and large debris from the basin and packaging into appropriate shipping containers
- Washing basin walls with high-pressure water.

Based on these activities, the following mechanisms for transporting radionuclides to the basin air space are evaluated as follows:

- Evaporation
- Debris washing, crushing, packaging, and removal
- Cutting
- Hydrowashing basin walls.

Evaporation. For planning purposes, it has been assumed that approximately 302,800 liters (80,000 gallons) of water will evaporate annually from the basin during the stabilization project (Appendix A-1). The current rate of evaporation is approximately 227,100 liters/year (60,000 gallons/year). The assumed quantity of 302,800 liters addresses the potential for enhanced evaporation due to increased surface area as stabilization is conducted.

With the exception of tritium, the amount of radioactivity transported to the air via this mechanism is assumed to be negligible. Both the vapor pressure and the mole fraction for each of the radionuclides in the water are very small, so the partial pressure (the product of vapor pressure and mole fraction) would be extremely small.

Tritium occurs in the water as tritiated water, and is assumed for this evaluation to have the same vapor pressure as nontritiated water. Assuming a net evaporation rate of 302,800 liters/year and a tritium activity of 39 microcuries per liter, the source term from evaporation will be 11.8 curies/year of tritium.

Debris Scrubbing, Packaging, and Removal. Radiological control procedures will be in place to minimize dispersion of contaminants during debris handling and removal activities. However, it is assumed that underwater activities will resuspend at least some of the settled sediment into the overlying water, increasing the activity of the water. Splashing might occur during these activities that could generate airborne droplets or contaminate surfaces outside of the storage pools. The water film adhering to the hardware and debris, as it is removed, could also generate airborne droplets and/or contaminate surfaces outside of the pools.

While large water droplets are expected to fall back into the pools, small water droplets could evaporate quickly in the air (because of the larger surface area-to-volume ratio) resulting in both soluble and insoluble radionuclides becoming airborne. Very small droplets (less than 10 microns) could be carried away by the air flow. Water that splashes outside the storage pools could dry on exposed surfaces of the basin; radionuclides from the water residue could then be resuspended as particulate.

To determine the potential source term, the following was assumed:

- The quantity of water droplets formed above the storage pools and the amount of splashing to outside surfaces is related to the amount of hardware being handled and removed from the pool

- A 1-millimeter film of water adheres to each piece of hardware removed from the basin

- 10 percent of the volume of this film is dispersed to the basin air space through evaporation, carryover of small droplets, or splashing and resuspension outside the storage pools. The 10 percent value was determined using GEJ and is conservative.

The total surface area of hardware was determined to be approximately 3,300 square meters (36,300 square feet) (Table 2-3). Using the above assumptions, the volume of water dispersed to the air space is:

$$3300\ m^2 \times 1\ mm \times \frac{1\ m}{1000\ mm} \times \frac{1000\ l}{1\ m^3} \times 10\% = 330\ liters$$

The net result is that 330 liters of water with suspended solids is added to the air space. The activity of the water assumes the following:

- 100 percent of the sediment in the vicinity of a piece of hardware being handled is resuspended into the column of water overlying sediment. (The resulting water-solids mixture contains about 500 parts per million by weight solids.)

- The activity per unit mass in the sediment is relatively uniform throughout the basin. This was supported by statistical evaluations of the 19 sediment samples collected from different locations of the basin in 1987 (Subrahmanyam 1988).

- The mass of sediment is uniformly distributed throughout the basin.

- All of the activity associated with the hardware is assumed to be removed during cleaning and also disperses into the water throughout all of the basin.

The activity of the water, including the suspended solids, is provided in Table 2-4. The emissions from debris handling and removal is the activity multiplied by 330 liters of water.

Cutting Activities. Hardware that might have to be cut prior to packaging consists of process tubes, a sorting table, and fuel baskets, all of which are constructed of steel. Cutting will be performed under water at depths of approximately 20 feet or greater. Potential cutting methods include plasma torch, hydraulic shears or nibblers, and tube cutters.

Mechanical cutting methods (e.g., shears, nibblers, and cutters) are expected to produce particles. The particles will include particles of the contaminated surfaces of the hardware. However, there is little motive force (e.g., bubbling) during mechanical cutting to carry these particles to the pool surface, so no significant dispersion of the particles directly to the air space is anticipated. The particles will contribute to the sediment radionuclide inventory, but the contribution is assumed to be small.

Plasma torch cutting is more likely to provide a direct contribution to the air source term via both enhanced evaporation and the dispersion of fume-like particles. Plasma torches produce a high-temperature (greater than

Table 2-4. Emissions from Debris Handling and Removal.

Nuclide	Activity in water film (μCi/L)*	Total emissions (Ci)**
Am-241	1.60 E+00	5.29 E-04
Ce-144	5.29 E+01	1.75 E-02
Co-60	5.47 E+00	1.81 E-03
Cs-134	2.48 E-01	8.17 E-05
Cs-137	3.91 E+01	1.29 E-02
H-3	3.90 E+01	1.29 E-02
K-40	2.04 E-03	6.73 E-07
Mn-54	1.51 E-02	4.97 E-06
Pu-238	3.57 E+00	1.18 E-03
Pu-239/240	2.24 E+01	7.39 E-03
Pu-241	4.75 E+02	1.57 E-01
Ru-106	1.53 E+01	5.04 E-03
Sb-125	2.70 E+00	8.92 E-04
Sr-90	3.59 E+02	1.19 E-01
U-234	6.66 E-02	2.20 E-05
U-235	1.21 E-02	3.99 E-06
U-238	5.42 E-02	1.79 E-05

*Assumes resuspension of all sediment and dispersion into water of all deposits on hardware.

**Assumes a total volume dispersed to air of 330 liters.

11,000 degrees celsius) ionized gas that severs virtually any conductive material (McGough and Knetl 1990). At these temperatures, water both vaporizes and dissociates into hydrogen and oxygen gas, generating bubbles that can carry particulate. Experience with underwater plasma-cutting at Three Mile Island and Yankee Rowe Nuclear indicates that a very fine particulate is generated, and that some of the particulate is carried to the air-water interface (Appendix A-3).

The vendor was not aware of any concerns or evaluations related to gaseous radionuclide emissions. However, the temperatures reported at the cutting surface exceed the boiling point of the radionuclides in the basin, so vaporization of the radionuclides was evaluated (Appendix B). Even with a worst-case evaluation of the heat energy generated by plasma torch cutting and the heat capacity of the surrounding water, the temperature of the water would decrease rapidly with distance from the torch. Assuming that all of the energy supplied to the torch is transferred into heating the water and that the torch is used continuously for 8 hours, the maximum temperature observed 5 feet from the cutting surface would be less than 100 degrees celsius. Cutting will be performed at depths of 20 feet or greater. Therefore, any radionuclides volatilized at the cutting surface will recondense before contacting the air.

Additional results of the thermal evaluation in Appendix B include the following.

- The maximum amount of water that could be heated to 100 degrees celsius during cutting would be approximately 14,000 liters.
- The maximum amount of additional water that could be evaporated would be approximately 1,700 liters. Additional evaporation due to plasma torch cutting is included in the assumed evaporation rate as discussed earlier.

The quantity of particulate generated by plasma torch cutting assumes the following.

- The kerf (the groove made by cutting) is 0.5 inch wide.
- The hardware that might be cut consists of process tubes, all structural steel (including the sorting table), and the fuel baskets.
- Equipment is cut into 3-foot lengths. Assuming that mass is proportional to length, the mass of the kerf is then about 1.4 percent of the total mass of the hardware.
- All of the mass associated with the kerf converts to particulate.

In determining the emissions to the air space based on GEJ, it was additionally assumed that 10 percent of the resulting particulate reaches the surface of the pool and is dispersed into the air space by bubbles. The activity of the particulate is assumed to be the activity of the hardware per unit mass presented in Table 2-2. Table 2-5 presents the resulting emissions.

Table 2-5. Emissions from Plasma Torch Cutting.

Nuclide	Total activity in cut hardware (Ci)*	Total emissions (Ci)**
Am-241	2.69 E+00	3.77 E-03
Ce-144	9.57 E+01	1.34 E-01
Co-60	4.77 E+00	6.67 E-03
Cs-134	4.40 E-01	6.16 E-04
Cs-137	6.77 E+01	9.48 E-02
H-3***		6.62 E-02
K-40		
Mn-54	1.30 E-02	1.82 E-05
Pu-238	5.98 E+00	8.38 E-03
Pu-239/240	3.76 E+01	5.27 E-02
Pu-241	7.98 E+02	1.12 E+00
Ru-106	2.76 E+01	3.87 E-02
Sb-125	4.27 E+00	5.97 E-03
Sr-90	6.56 E+01	9.19 E-02
U-234	1.12 E-01	1.57 E-04
U-235	2.03 E-02	2.84 E-05
U-238	9.11 E-02	1.28 E-04

*Cut hardware assumed to include fuel baskets, dump cart assemblies, sorting tray, dump basket, and process tubes. Total mass approximately 45,000 kilograms. Activity per unit mass Table 2-2.

**Assumes 1.4 percent of mass becomes particulate, and 10 percent of particulate released to air at air-water interface.

***Tritium from additional 1,700 liters evaporation during cutting.

Hydroscrubbing. A ring consisting primarily of mineral deposits and algae is present on the walls of the storage pools at the air-water interface. During stabilization, a stream of high-pressure water discharged at about 20 gallons per minute will be directed at the ring to wash the material down into the basin water. The water source is likely to be filtered basin water. Aerosol generated from hydroscrubbing could contribute to the air emissions.

Hydroscrubbing procedures are designed to minimize splashback and mist generation. In the case of the basin, the water stream will be aimed at a point on the wall approximately 0.5 foot above the air-water interface. The spray nozzle will be located about 6 to 8 inches from the wall and angled downwards about 33 degrees. In the past, using similar techniques at 100 Area facilities, the aerosol produced was so slight and localized that operators positioned 10 to 20 feet away from the impact point did not require respiratory protection, and visibility was only slightly impaired (Appendix A-4).

To determine the emissions, the following parameters and assumptions were used.

- The perimeter of the storage pools in the basin measures 280 meters (914 feet).
- The air space for a distance of 1 meter (3 feet) above the water surface and 2 meters (6 feet) out from the wall is assumed to become laden with aerosol, and all of the aerosol from this volume contributes to the air space source term.
- The aerosol loading in the affected air volume is assumed to be 1,000 milligrams per square meter (the equivalent of a heavy fog). Based on the past experience cited above, the assumption of aerosol loading approximating a heavy fog is conservative. However, it provides a bounding case for this analysis.
- The solid deposits on the basin walls are assumed to be similar to the basin sediment in terms of isotope type and activity. The aerosol droplets are assumed to contain 500 parts per million by weight of these solids, a loading similar to that assumed for the film on hardware removed from the basin. The activity of the water in the aerosol is assumed to be that of basin water.

If all of the aerosol were to be condensed it would be equivalent to 0.6 liter of water containing 500 parts per million by weight of suspended solids. The emissions are presented in Table 2-6.

Projected annual abated emissions based on the engineering evaluations identified in the above information are presented in Table 2-7. In addition to those assumptions identified in the above information, the following assumptions were made.

- The debris washing, packaging, and removal activities are assumed to occur over a 2-year period. The source term to the basin air space from these activities was therefore divided by two for the annual release.

Table 2-6. Emissions from Hydroscrubbing.

Nuclide	Aerosol with solids (μCi/L)*	Total emissions (Ci)**
Am-241	1.60 E+00	9.62 E-07
Ce-144	5.29 E+01	3.17 E-05
Co-60	5.47 E+00	3.28 E-06
Cs-134	2.48 E-01	1.49 E-07
Cs-137	3.91 E+01	2.35 E-05
H-3	3.90 E+01	2.34 E-05
K-40	2.04 E-03	1.22 E-09
Mn-54	1.54 E-02	9.04 E-09
Pu-238	3.57 E+00	2.14 E-06
Pu-239/240	2.24 E+01	1.34 E-05
Pu-241	4.75 E+02	2.85 E-04
Ru-106	1.53 E+01	9.17 E-06
Sb-125	2.70 E+00	1.62 E-06
Sr-90	3.59 E+02	2.16 E-04
U-234	6.66 E-02	4.00 E-08
U-235	1.21 E-02	7.25 E-09
U-238	5.42 E-02	3.25 E-08

*Activity in condensed aerosol, assuming condensed aerosol is equivalent to basin water with approximately 500 parts per million by weight suspended sediment.

**Assumes a total condensed aerosol volume of 0.6 liter.

Table 2-7. Emissions Results Based on Good Engineering Judgement.

Nuclide	Evaporation (Ci/yr)	Debris removal (Ci/yr)* from Table 2-4	Cutting (Ci/yr) from Table 2-5	Hydroscrubbing (Ci/yr) from Table 2-6	Unabated emissions (Ci/yr)	HEPA filter DF**	Abated emissions (Ci/yr)
Am-241		2.64 E-04	3.77 E-03	9.62 E-07	4.04 E-03	2,000	2.02 E-06
Ce-144		8.73 E-03	1.34 E-01	3.17 E-05	1.43 E-01	2,000	7.13 E-05
Co-60		9.03 E-04	6.67 E-03	3.28 E-06	7.58 E-03	2,000	3.79 E-06
Cs-134		4.08 E-05	6.16 E-04	1.49 E-07	6.57 E-04	2,000	3.29 E-07
Cs-137		6.46 E-03	9.48 E-02	2.35 E-05	1.01 E-01	2,000	5.06 E-05
H-3	1.18 E+01	6.43 E-03	6.62 E-02	2.34 E-05	1.19 E+01	1	1.19 E+01
K-40		3.37 E-07		1.22 E-09	3.38 E-07	2,000	1.69 E-10
Mn-54		2.49 E-06	1.82 E-05	9.04 E-09	2.07 E-05	2,000	1.04 E-08
Pu-238		5.89 E-04	8.38 E-03	2.14 E-06	8.97 E-03	2,000	4.48 E-06
Pu-239/240		3.69 E-03	5.27 E-02	1.34 E-05	5.64 E-02	2,000	2.82 E-05
Pu-241		7.83 E-02	1.12 E+00	2.85 E-04	1.20 E+00	2,000	5.98 E-04
Ru-106		2.52 E-03	3.87 E-02	9.17 E-06	4.12 E-02	2,000	2.06 E-05
Sb-125		4.46 E-04	5.97 E-03	1.62 E-06	6.42 E-03	2,000	3.21 E-06
Sr-90		5.93 E-02	9.19 E-02	2.16 E-04	1.51 E-01	2,000	7.57 E-05
U-234		1.10 E-05	1.57 E-04	4.00 E-08	1.68 E-04	2,000	8.39 E-08
U-235		1.99 E-06	2.84 E-05	7.25 E-09	3.04 E-05	2,000	1.52 E-08
U-238		8.94 E-06	1.28 E-04	3.25 E-08	1.36 E-04	2,000	6.82 E-08

*Debris removal occurs over a 2-year period.

**Assumes an installed HEPA filter removal efficiency of 99.95 percent (decontamination factor of 2,000).

- The source terms from plasma torch cutting and hydroscrubbing were assumed to occur during a 1-year period.

- The high-efficiency particulate air (HEPA) filter is assumed to have an installed removal efficiency for particulate of 99.95 percent.

2.2.5.2 40 CFR 61, Appendix D Methodology. The following section is presented as information only. Table 2-8 presents the projected annual emissions from the basin stabilization project using 40 CFR 61, Appendix D methodology.

The radionuclides used in this projection are those identified previously as representing the activity associated with the water, sediment, hardware, and fuel element fragments (Table 2-2). The following assumptions were used.

- A physical adjustment factor of 1 was applied to all tritium.

- A physical adjustment factor of 1 was applied to the activity in 25 percent (by weight) of the hardware that will be cut (approximately 45,000 kilograms of hardware). This is based on the assumption that the plasma torch will heat the metal for approximately 4 inches on either side of the cutting point to 100 degrees celsius or greater.

- A physical adjustment factor of 1 was applied to the activity in 14,000 liters of water. This is the maximum amount of water that could be heated to 100 degrees celsius or greater if a plasma torch is used (Appendix B).

- A physical adjustment factor of 0.001 was applied to all sediment, because the sediment exists as solid particulate.

- A physical adjustment factor of 0.001 was applied to all activity associated with the hardware that will not be heated to greater than 100 degrees celsius, including hardware that will not be cut (approximately 57,000 kilograms of the total) 75 percent (by weight) of the hardware that will be cut.

- A physical adjustment factor of 0.001 was applied to the radionuclide inventory associated with the fuel element and target fragments. Although this inventory is in a solid form, the fragments are subject to corrosion and dispersion as solid particulate.

- An emissions control adjustment factor of 0.01 was applied to all radionuclides except tritium to account for the use of HEPA filters.

Table 2-8. Projected Annual Emissions Results Based on 40 CFR 61 Methodology.

Nuclide	Activity** (Ci) above 100 °C	Release rate multiplier	Resultant activity** (Ci)	Remaining activity* (Ci)	Release rate multiplier	Resultant activity* (Ci)	Total emission rate (Ci/yr)***	HEPA adjustment factor	Total emissions (Ci/yr)
Am-241	6.73 E-01	1	6.73 E-01	9.34 E+00	0.001	9.34 E-03	6.82 E-01	0.01	6.82 E-03
Ce-144	2.39 E+01	1	2.39 E+01	3.15 E+02	0.001	3.15 E-01	2.42 E+01	0.01	2.42 E-01
Co-60	1.19 E+00	1	1.19 E+00	2.13 E+01	0.001	2.13 E-02	1.21 E+00	0.01	1.21 E-02
Cs-134	1.10 E-01	1	1.10 E-01	5.64 E+00	0.001	5.64 E-03	1.16 E-01	0.01	1.16 E-03
Cs-137	1.69 E+01	1	1.69 E+01	8.71 E+02	0.001	8.71 E-01	1.78 E+01	0.01	1.78 E-01
H-3	5.45 E-01	1	5.45 E-01	2.99 E+02	1	2.99 E-01	3.00 E+02	1	3.00 E+02
K-40	2.86 E-05	1	2.86 E-05	8.34 E-03	0.001	8.34 E-06	3.69 E-05	0.01	3.69 E-07
Mn-54	3.26 E-03	1	3.26 E-03	5.85 E-02	0.001	5.85 E-05	3.32 E-03	0.01	3.32 E-05
Pu-238	1.50 E+00	1	1.50 E+00	2.10 E+01	0.001	2.10 E-02	1.52 E+00	0.01	1.52 E-02
Pu-239/240	9.40 E+00	1	9.40 E+00	1.30 E+02	0.001	1.30 E-01	9.53 E+00	0.01	9.53 E-02
Pu-241	1.99 E+02	1	1.99 E+02	2.76 E+03	0.001	2.76 E+00	2.02 E+02	0.01	2.02 E+00
Ru-106	6.90 E+00	1	6.90 E+00	9.09 E+01	0.001	9.09 E-02	6.99 E+00	0.01	6.99 E-02
Sb-125	1.07 E+00	1	1.07 E+00	1.00 E+01	0.001	1.00 E-02	1.08 E+00	0.01	1.08 E-02
Sr-90	1.65 E+01	1	1.65 E+01	2.16 E+03	0.001	2.16 E+00	1.86 E+01	0.01	1.86 E-01
U-234	2.80 E-02	1	2.80 E-02	3.88 E-01	0.001	3.88 E-04	2.84 E-02	0.01	2.84 E-04
U-235	5.08 E-03	1	5.08 E-03	7.03 E-02	0.001	7.03 E-05	5.15 E-03	0.01	5.15 E-05
U-238	2.28 E-02	1	2.28 E-02	3.15 E-01	0.001	3.15 E-04	2.31 E-02	0.01	2.31 E-04

*Less than 100 °C. Includes all of the activity in the sediment, fuel elements, and hardware that will not be cut; activity in all water minus 1,400 liters; and 75 percent of the activity in hardware that will be cut.

**Heated to 100 °C or greater. Includes activity in 14,000 liters of water and 25 percent of hardware above 100 °C and the remaining activity.

***Sum of the activity.

HEPA = high-efficiency particulate air (filter).

2.2.4 Operating Mode

For purposes of the release evaluations, the debris handling and removal activities are assumed to occur over a period of 2 years. It is assumed that hardware cutting and hydroscrubbing will also occur during one of those 2 years, so that all activities contribute concurrently to the worst-case annual emission. The actual time spent cutting and hydroscrubbing is expected to be on the order of hours.

2.2.5 Fraction of Facility's Inventory Available for Potential Release to the Air

The phrase "fraction of inventory available for potential release to the air" is interpreted to mean "potential to emit". Neither phrase is defined in existing DOH regulations; however, a definition for "potential to emit" currently is being developed by the DOH for inclusion in revised regulations. Until the revised regulations have been issued, the DOH has provided guidance that the definition of "potential to emit" found in 40 CFR 61 is to be followed. Therefore, Table 2-7, the Unbated emissions column, presents the potential annual radionuclide release rates to air assuming unabated (no pollution control equipment in place) emissions (Ci/yr).

2.3 EFFLUENT SYSTEM LAYOUT

This section describes the ventilation system for the basin and the removal efficiencies for control devices in the system.

2.3.1 Airflow Configuration

The N Reactor is separated into five confinement zones for radiation control purposes. Confinement Zone I is the innermost zone surrounding the reactor core and its directly associated systems and areas. Because of high radiation levels around the reactor, Zone I is vented through a HEPA filtration system located in the 117N Filter Building prior to discharge through the 116N Stack. Zone I is maintained at a negative pressure relative to other zones and the external atmosphere.

Prior to 1993, part of the basin air space (considered part of Zone III) was exhausted directly to the atmosphere via roof exhausters. In October 1993, doorways between Zone I and the basin were opened to allow air to flow from the basin to Zone I. Tests were conducted that demonstrated that the new configuration maintained a negative pressure differential of -0.02 inch to -0.1 inch water gauge between the basin, adjacent air spaces, and the atmosphere (Myott 1993). The new configuration was approved for permanent use, and the N Reactor Zone I ventilation system with its HEPA filtration system now serves as the ventilation system for the basin.

The basin/transfer area airflow configuration is shown in Figure 2-1. Figure 2-2 shows the airflow configuration through the 117N HEPA filter building.

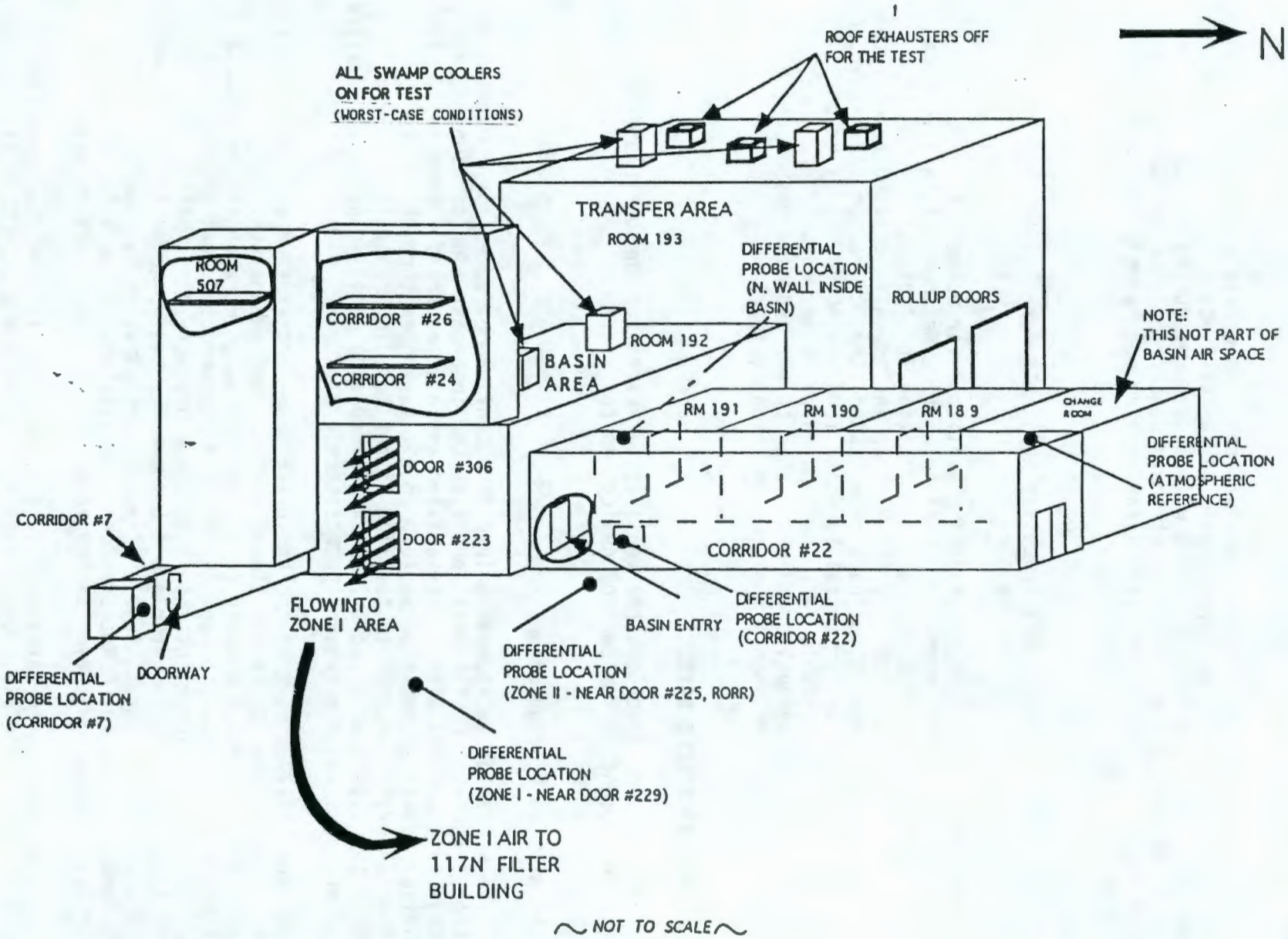

Figure 2-1. Simplified Sketch 105N Basin/Transfer Area Air Space.

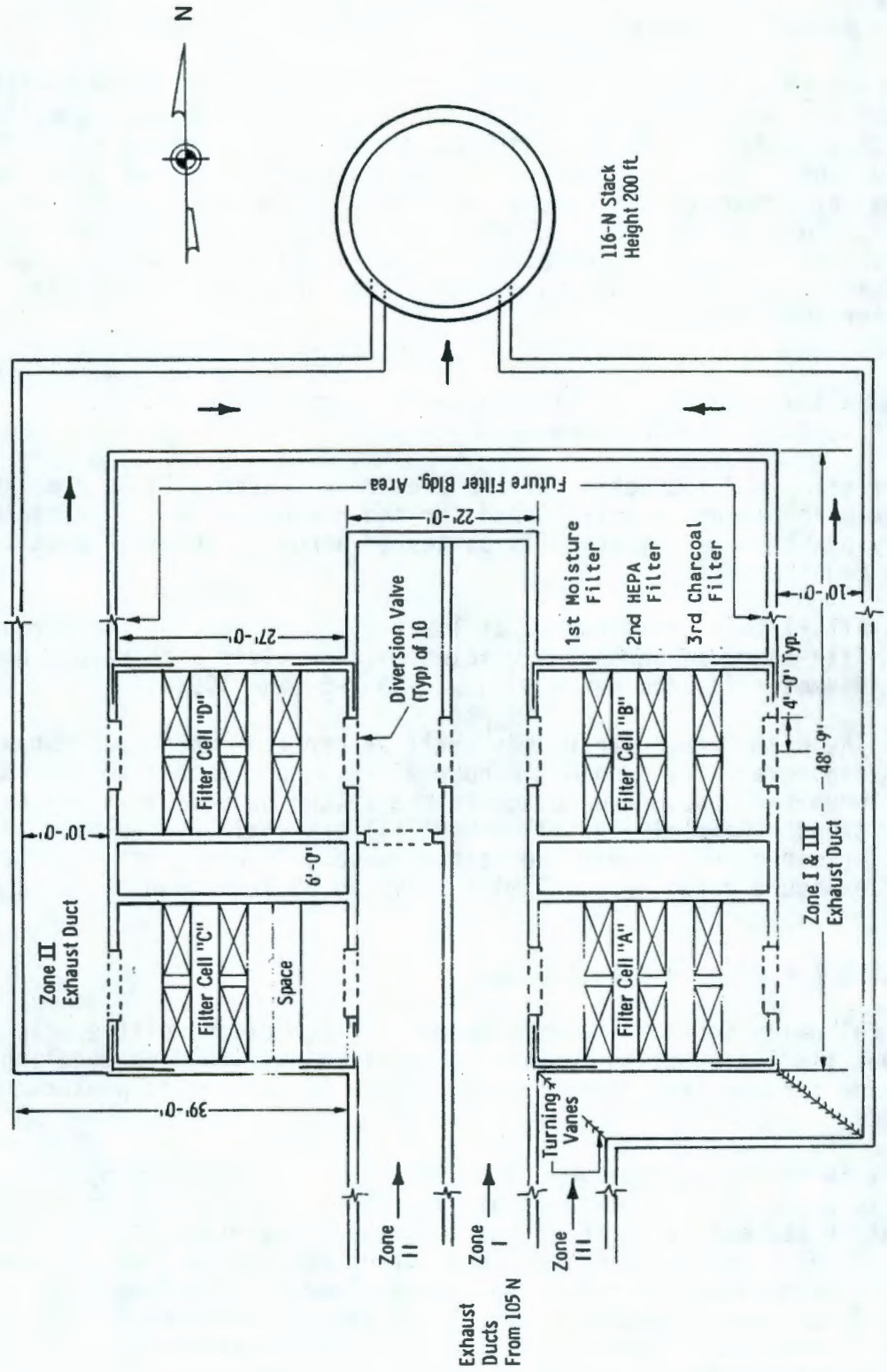

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Figure 2-2. Exhaust Filter and Diversion System - 117N.

2.3.2 Efficiency Values of Each Control Device for Removal of Radioactivity

The bank of HEPA filters in the 117N Filter Building is the control device used for removal of radioactive particles from the basin ventilation system. In accordance with Hanford Site procedures, an installed HEPA or HEPA-equivalent filter shall have a leakage rate efficiency of 99.95 percent for removal of airborne particulate (WHC 1993a). The basis is *The Nuclear Air Cleaning Handbook*, Section 8.2, which states that by definition a HEPA system must exhibit an installed decontamination factor of 2,000; an efficiency of 99.95 percent for aerosols having a nominal median diameter less than 1 micrometer (WHC 1993a).

2.3.3 Means and Frequency of Testing and Inspecting Effluent Treatment System

The testing and inspection of the treatment system will be done in accordance with protocols established for the Hanford Site. At a minimum, the efficiency of the HEPA filters will be tested annually and must meet the following requirements.

- All filters shall remove at least 99.95 percent of the Hanford Site-approved aerosol particles with particle size ranges and median diameter in accordance with WHC-CM-7-5 (WHC 1993a).

- The HEPA filter cartridges shall be replaced when continuous exposure rates exceed 1 R/hour at 15.3 centimeters (6 inches), or when the pressure drop across the filter exceeds 5 inches water gauge. Remotely installed HEPA filters shall be replaced when the pressure drop across the filter exceeds 5 inches water gauge, or exposure rates exceed limits provided by Radiation Protection.

2.3.4 Stack (or Release Point) Data

The following sections provide detailed descriptions of the release point. Detailed information required to perform offsite dose modeling for radionuclide airborne emissions from the basin activities is presented in Appendix C.

2.3.4.1 116N Stack Description. The 116N Stack is a converging stack that is the release point for the basin stabilization activities. It is located approximately 152 meters north of the 105N Building, which houses the basin. The stack is 61.3 meters in height above grade (68 meters from the base). The top of the stack is circular with an inside diameter of 4.3 meters, an outside diameter of 4.6 meters, and has a cast iron cap. The bottom of the stack has an inside diameter of 6.6 meters and an outside diameter of 7.2 meters. The base is a steel reinforced concrete octagon 10.2 meters in width.

The system was exhausting about 60,000 cubic feet per minute until October 1993 when the exhaust system was reconfigured to direct basin air through the 116N Stack. The current exhaust fan flow rate is about 100,000 cubic feet per minute. The calendar year 1992 volumetric flow was

approximately 3.6 E+10 cubic feet. The stack exhausts room air so the stack temperature remains relatively constant. During the stack flow tests conducted in October 1993, the stack gas temperature was measured at about 70 degrees fahrenheit.

2.3.4.2 117N Filter Building. The air emissions from the basin activities pass through the 117N Filter Building prior to discharge through the stack. The 117N Filter Building is located about 53 meters north of the 105N Building, which houses the basin and is built with most of the building below grade. The roof of the building is above grade and has sections that can be removed to facilitate filter changeout. As explained in Section 2.3.1, the 117N Filter Building is connected to the basin area by the exhaust ducts. Zone I and II air is routed through the 117N Filter Building where the air is subjected to one bank of HEPA filtration. The bank of HEPA filtration consists of three stages; first stage consists of a moisture filter, the second stage is the HEPA filter, and the third stage is a charcoal filter. After the air is routed through the HEPA filtration bank, it is discharged through the 116N Stack.

2.3.5 Description of the Effluent Sampling/Monitoring System(s)

The 116N Stack is continuously sampled for airborne particulates. The 47-millimeter sample filter is removed weekly and analyzed for total α, total ß, and gamma emitting radionuclides. The weekly filters are composited monthly for Sr-90, iso-Pu, and Am-241. The charcoal sample collection filters are no longer installed after obtaining about 2 years of negative results and based on the short half-lives of the halogens. The sample pump is checked quarterly to verify sample flow. The sample pump is also inspected quarterly as part of a preventive maintenance schedule.

The sampling system is characterized by a single sample probe in the stack at the approximately 60-foot level. The system was designed for about 210,000 cubic feet per minute. The sample line is constructed of 60 feet of 5/8-inch stainless steel tubing. The line is heat traced and insulated. There are seven bends in the sample line. Sample flow regulation is provided by a flow-limiting orifice in the sample cartridge containing the filter media. The orifice limits the sample flow to 1.2 cubic feet per minute.

2.3.6 Environmental Sampling Monitoring System

The Hanford Site maintains a comprehensive environmental sampling and monitoring program. Information describing the program has been previously provided to the DOH.

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3.0 GENERAL INFORMATION ON SAMPLE ANALYSIS

The stack sample media will be analyzed in accordance with existing procedures previously supplied to the DOH during audits.

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4.0 DEMONSTRATION OF COMPLIANCE

This section contains the information about the dose modelling effort required to demonstrate compliance with the national and state offsite dose impact standard of 10 millirem/year effective dose equivalent (EDE) to the maximally exposed individual (MEI).

4.1 METHODOLOGY USED TO DEMONSTRATE COMPLIANCE

The Clean Air Assessment Package 1988 (CAP-88) computer code was used to calculate the EDE from the proposed basin stabilization project to the MEI, according to requirements of the *Washington Administrative Code* (WAC) 246-247.

4.2 MODELLING METHODOLOGY

CAP-88, a U.S. Environmental Protection Agency (EPA)-approved code package, has the capability to model both ground-level and elevated airborne releases from locations in the 100, 200 East, 200 West, and 300 Areas. The model calculates EDE to an individual member of the public using Hanford Site meteorological data. Doses from radioactive daughter products of primary radionuclides are included.

A CAP-88 evaluation was run specifically for the projected releases of basin emissions from the 116N Stack. Input parameters and detailed analyses are provided in Appendix C. The resulting dose factors for a 1-Ci release are provided in Table 4-1.

4.3 RESULTS OF METHOD (EFFECTIVE DOSE EQUIVALENT FOR WHOLE BODY AND RELEVANT ORGANS)

Table 4-1 provides the total projected EDE from both abated and unabated airborne emissions resulting from basin stabilization activities based on GEJ. The total projected EDE to the offsite MEI from unabated emissions is 0.27 millirem/year. The total projected EDE to the offsite MEI from abated emissions is 0.0002 millirem/year. The dose attributable to radiological emissions from the basin stabilization project will contribute 0.002 percent of the WAC 246-247 EDE regulatory limit of 10 millirems/year to the offsite MEI.

For comparison, the total projected EDE to the offsite MEI using 40 CFR 61, Appendix D Methodology is 0.477 millirem/year and is presented in Table 4-2. The radionuclides listed include those that could contribute 10 percent or more of the EDE to the MEI.

The dose resulting from all Hanford Site operations in 1992, was determined to be 0.004 millirem/year (PNL 1993). The emissions as a result of the full year's stabilization activities at the basin, in conjunction with previous operations at the Hanford Site, will not result in a violation of the National Emission Standard of 10 millirem/year (40 CFR 61).

Table 4-1. Effective Dose Equivalent to Individual Receiving Maximum Exposure to Radiological Emissions Based on Good Engineering Judgement.

Nuclide	Unabated emission (Ci/yr)	Abated emission (Ci/yr)	Modeled dose factor (mrem/Ci)*	Unabated MEI dose (mrem/yr)	% of Unbated MEI dose	Abated MEI dose (mrem/yr)	% of Abated MEI dose
Am-241	4.04 E-03	2.02 E-06	4.25 E+00	1.71 E-02	6.42	8.57 E-06	3.96
Ce-144	1.43 E-01	7.13 E-05	4.27 E-03	6.09 E-04	0.23	3.05 E-07	0.14
Co-60	7.58 E-03	3.79 E-06	5.71 E-01	4.33 E-03	1.62	2.16 E-06	1.00
Cs-134	6.57 E-04	3.29 E-07	3.14 E-02	2.06 E-05	0.01	1.03 E-08	0.00
Cs-137	1.01 E-01	5.06 E-05	5.96 E-02	6.04 E-03	2.26	3.02 E-06	1.40
H-3	1.19 E+01	1.19 E+01	6.98 E-06	8.29 E-05	0.03	8.29 E-05	38.30
K-40	3.38 E-07	1.69 E-10	4.53 E-02	1.53 E-08	0.00	7.65 E-12	0.00
Mn-54	2.07 E-05	1.04 E-08	3.65 E-03	7.57 E-08	0.00	3.78 E-11	0.00
Pu-238	8.97 E-03	4.48 E-06	2.58 E+00	2.31 E-02	8.67	1.16 E-05	5.35
Pu-239/240	5.64 E-02	2.82 E-05	2.78 E+00	1.57 E-01	58.68	7.83 E-05	36.22
Pu-241	1.20 E+00	5.98 E-04	4.39 E-02	5.25 E-02	19.66	2.62 E-05	12.13
Ru-106	4.12 E-02	2.06 E-05	5.57 E-03	2.29 E-04	0.09	1.15 E-07	0.05
Sb-125	6.42 E-03	3.21 E-06	5.87 E-03	3.77 E-05	0.01	1.88 E-08	0.01
Sr-90	1.51 E-01	7.57 E-05	3.86 E-02	5.84 E-03	2.19	2.92 E-06	1.35
U-234	1.68 E-04	8.39 E-08	1.04 E+00	1.74 E-04	0.07	8.72 E-08	0.04
U-235	3.04 E-05	1.52 E-08	9.93 E-01	3.02 E-05	0.01	1.51 E-08	0.01
U-238	1.36 E-04	6.82 E-08	1.14 E+00	1.56 E-04	0.06	7.81 E-08	0.04
Total				2.67 E-01	100.00	2.16 E-04	100.00

*Clean Air Assessment Package 1988 dose estimates for releases from a 200-foot stack for an individual located 14.7 kilometers east of 100N (equivalent to actual offsite location 11.5 kilometers west).

MEI = maximally exposed individual.

Table 4-2. Effective Dose Equivalent to Individual Receiving Maximum Exposure to Radiological Emissions (based on 40 CFR 61, Appendix D Methodology).

Nuclide	Total emissions (Ci/yr)	Modeled dose factor (mrem/Ci)*	Appendix D dose (mrem/yr)
Am-241	6.82 E-03	4.25 E+00	2.90 E-02
Ce-144	2.42 E-01	4.27 E-03	1.03 E-03
Co-60	1.21 E-02	5.71 E-01	6.85 E-03
Cs-134	1.16 E-03	3.14 E-02	3.64 E-05
Cs-137	1.78 E-01	5.96 E-02	1.06 E-02
H-3	3.00 E+02	6.98 E-06	2.09 E-03
K-40	3.69 E-07	4.53 E-02	1.67 E-08
Mn-54	3.32 E-05	3.65 E-03	1.21 E-07
Pu-238	1.52 E-02	2.58 E+00	3.92 E-02
Pu-239/240	9.53 E-02	2.78 E+00	2.65 E-01
Pu-241	2.02 E+00	4.39 E-02	8.87 E-02
Ru-106	6.99 E-02	5.57 E-03	3.89 E-04
Sb-125	1.08 E-02	5.87 E-03	6.34 E-05
Sr-90	1.86 E-01	3.86 E-02	3.32 E-02
U-234	2.84 E-04	1.04 E+00	2.95 E-04
U-235	5.15 E-05	9.93 E-01	5.11 E-05
U-238	2.31 E-04	1.14 E+00	2.63 E-04
		Total	4.77 E-01

*Clean Air Assessment Package 1988 dose estimates for releases from a 200-foot stack for an individual located 14.7 kilometers east of 100N (equivalent to actual offsite location 11.5 kilometers west).

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5.0 BEST AVAILABLE RADIONUCLIDE CONTROL TECHNOLOGY

A Best Available Radionuclide Control Technology (BARCT) assessment is provided in Appendix D.

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6.0 REFERENCES

40 CFR 61, "National Emission Standards for Hazardous Air Pollutants," Title 40, *Code of Federal Regulations*, Part 61, U.S. Environmental Protection Agency, Washington, D.C.

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WHC, 1993b, *N Reactor Deactivation Program Plan*, December 1993, WHC-SP-0615, Rev. 4, Westinghouse Hanford Company, Richland, Washington.

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APPENDIX A-1

CORRESPONDENCE - N REACTOR EFFLUENT PROJECTION

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Westinghouse
Hanford Company

Internal
Memo

From: N Reactor Deactivation
Phone: 373-4164
Date: November 19, 1993
Subject: N REACTOR EFFLUENT PROJECTION

85600-93-55

To: Gary Wells H6-26

cc:	Warren Cohen	H6-26	M. R. Morton	R2-77
	Ella Coenberg	H6-25	B. D. Schilperoort	X8-20
	Jerry Hunacok	X0-41	D. Schilperoort	X8-29
	J. L. Laurenz	R2-77	Jerry Turnbaugh	H6-25
	Chris Lucas	X0-35	John Walsh	X0-57
	F. N. McDonald	R3-45	Dave Watson	A5-55

Attached is an updated estimate of the water volumes and removal sequence of N Reactor contaminated effluents. This data should be used for future planning and analysis purposes rather than the former quantities identified in the BAT documents.

Should you have any questions please contact John Walsh (373-1408) of my staff.

R. J. Gimera
N Reactor Manager

ak

attachment

N REACTOR DEACTIVATION
WATER VOLUME/REMOVAL QUANTITIES
PROGRAM SUMMARY
Thousands of Gallons

	FY-94	FY-95	FY-96	FY-97	SUMMARY
*PROGRAM SUMMARY					
Fiscal Year Starting Volume	1603	1558	1498	498	1603
Misc. (leaks to sumps, etc.)	10	10	10	0	30
Evaporation	(292)	(292)	(292)	(61)	(937)
Water Additions - Makeup	232	202	200	10	644
Water Additions - Decon	5	20	15	10	50
Volume Subtotal	1558	1498	1431	457	N/A
Removal Volume	0	0	(933)	(457)	(1390)
Fiscal Year End Volume	1558	1498	498	0	0

* Does not include clearwell, potable water, or fire system water. Since these are not contaminated, they will discharge to the NPDES permitted outfall (009~).

N REACTOR DEACTIVATION
WATER VOLUME/REMOVAL QUANTITIES
Thousands of Gallons

	FY-94	FY-95	FY-96	FY-97	SUMMARY
PROGRAM SCHEDULE DATES					
Drain Fuel Basins					
Transfer Other pits/sumps/ piping to Lift Station					
Drain Lift Station					
Drain/Remove Emergency Dump Basin					
*WATER VOLUME & REMOVAL					
FUEL BASIN COMPLEX					
Fiscal Year Starting Vol.	1000	955	895	0	1000
Evaporation	(80)	(80)	(80)	0	(240)
Water Addition - Make-up	30	0	0	0	30
Water Addition - Decon	5	20	15	0	40
Volume Subtotal	955	895	830	0	N/A
Removal Volume	0	0	(830)	0	(830)
Fiscal Year End Volume	955	895	0	0	0
LIFT STATION					
Fiscal Year Starting Vol.	15	15	15	23	15
Evaporation	(2)	(2)	(2)	(1)	(7)
Water Addition - Make-up	2	2	0	0	4
Volume Subtotal	15	15	13	22	N/A
Transfer from -Misc. leaks etc.(10);Sumps(35)(Less Evap. of 3)	0	0	42	0	42
Transfer from - 'C' Pit (43 less evap. of 5)	0	0	38	0	38
Transfer from - Piping	0	0	25	35	60
Water Addition - Rod Room Pit (10 less evap. of 2)	0	0	8	0	8
Available for Removal	15	15	126	57	N/A
Removal Volume	0	0	103	(57)	(160)
Fiscal Year End Volume	15	15	23	0	0

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N REACTOR DEACTIVATION
WATER VOLUME/REMOVAL QUANTITIES
Thousands of Gallons

	FY-94	FY-95	FY-96	FY-97	SUMMARY
EMERGENCY DUMP BASIN					
Fiscal Year Starting Volume	440	440	440	440	440
Evaporation	(200)	(200)	(200)	(60)	(660)
Water Addition - Makeup	200	200	200	10	610
Water Addition - Decon	0	0	0	10	10
VOLUME SUBTOTAL	440	440	440	400	N/A
Removal Volume	0	0	0	(400)	(400)
Fiscal Year End Volume	440	440	440	0	0
OTHER PITS/SUMPS/PIPING					
Fiscal Year Starting Vol. 'C' Pit (43), Piping (60), Rod Room Pit (10), Sumps (35)	148	148	148	35	148
Misc. (leaks to sumps, etc.)	10	10	10	0	30
Evaporation	(10)	(10)	(10)	(0)	(30)
Water Addition - Makeup	0	0	0	0	0
Volume Subtotal	148	148	148	35	N/A
****Transfer To Lift Station**	0	0	(113)	(35)	(148)
Fiscal Year End Volume	148	148	35	0	0

* Does not include non-contaminated clearwell, potable water, or fire system water. These will be discharged to NPDES permitted outfall.

** See Lift Station analysis for removal volume.

APPENDIX A-2

CORRESPONDENCE - UNDERWATER PLASMA CUTTING

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PCI EnergyServices

FIELD MACHINING, WELDING,
TOOL DESIGN & ENGINEERING

December 21, 1993

Linda Mihalik
CH 2 M Hill
1933 Jadwin
Suite 225
Richland, WA 99352

Subject: Underwater Plasma Cutting

Dear Linda:

It was a pleasure speaking with you the other day about underwater plasma cutting. PCI has been involved with this process for the past 14 years. We have worked at nuclear plants such as North Anna, Three Mile Island, Connecticut Yankee, Shoreham, Yankee Rowe, and Fort St. Vrain. In my opinion, we are the leader of underwater plasma cutting in the nuclear industry.

During our conversation you asked me about the particulates which escape into the atmosphere during the underwater plasma cutting process. I spoke with our technicians, who are at Yankee Rowe and are working with the decommissioning of their reactor vessel internals. The following is some information which may be helpful to you.

When plasma cutting under water, PCI uses a specially designed floating vent hood to trap any particulates which escape into the air. This vent hood has a suction hose which is attached to a Hepa filtration unit. The information for this unit is as follows:

NFS/RPS
Model PFB - 2500 (SP)
Bag in / Bag out
Portable Hepa Filtration Module
Nominal flow rate 2,000 CFM

The capture velocity range under the hood is 163 - 230 feet per minute. Our floating hood size is approximately an 8 foot hexagon.

PCI EnergyServices

While at the Yankee Rowe Nuclear Plant, we have experienced the following conditions during the underwater plasma cutting operations.

Components - Reactor vessel internals
Depth of water - 12 feet to 20 feet

Time Range - For 11 weeks the dose rate for the material cut ranged from 1 REM to 15,000 REM. The HEPA filter started a 0 mR\hr and went to 4 mR/hr during this time.

Time Range - In 4 days the dose rate of the material cut ranged from 15,000 REM to 25,000 REM. The HEPA filter dose rate went up to 50 mR/hr. The majority of this increase in the dose rate was caused by the initial cut into the core plate. A significant amount of "crud" was trapped in this plate and was loosened during the initial cutting. A fine powder-like substance migrated to the surface causing an initial blast of airborne which was all collected by the hood. Other than this burst from the core plate cutting, the HEPA off gas collection was not necessary.

At this time I am not aware of any analysis which has been performed to the HEPA filters.

You also asked me about temperature ranges at the plasma torch. For more technical information I recommend you call Hypertherm, the company from which we purchase the plasma equipment from. Their phone number is 603-643-~~2~~3441.

Dr Dick Sanders
Manager Process Group

If there is any more information you might require please call me at 708-680-8100 and I will be happy to help you.

Sincerely,

Mike Simundza
Regional Account Manager

APPENDIX A-3

CORRESPONDENCE - 100N FUEL CANISTER ASSAYS

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Battelle
Pacific Northwest Laboratories

Provided by Daryl Schilperoort, 12/3/93

~~R P Denise~~ R Gano

Project Number

Internal Distribution

PR Boolen
JD Forsythe
L Garret
DL Haggard
RA Larson
RJ Pyzel
File/LB

Date July 24, 1990

To J. J. Jernberg

From R. L. Brodzinski Ron

Daryl

Subject 100N Fuel Canister Assays

A total of 11 canisters were neutron counted for residual TRU contamination before and after cleaning. In addition, canisters #7 amd #10 were gamma counted both before and after cleaning which should give a reasonably good measure of the decontamination factor afforded by the cleaning process. Assuming the TRU is weapons grade fuel, the first table below gives the TRU concentration before and after cleaning in units of nCi/g net weight of canister for the nine canisters which were neutron counted only. The second table gives both the TRU and the gamma emitters, before and after, for canisters #7 and #10. I am enclosing the hardcopies of the data for you to archive. We will archive the data on floppy disc here.

Canister #	TRU before	TRU after
2	725	<32
3	942	26.9
4	493	<30
5	1020	<30
6	827	9.02
8	849	37.3
9	1110	46.0
11	453	<31
12	2680	18.9

Isotope	#7 before	#7 after	#10 before	#10 after
TRU	989	89.8	853	46.3
^{54}Mn	4.27	1.91	0.994	0.865
^{60}Co	148	67.7	80.3	31.9
^{125}Sb	198	84.8	89.8	61.3
^{137}Cs	1330	222	387	154
^{154}Eu	56.3	28.4	46.5	25.0
^{155}Eu	<13	<4.2	27.6	<3.8

DRAFT

TEST REPORT OF FEASABILITY STUDY FOR HIGH PRESSURE CLEANING OF FUEL CANISTERS

1.0 INTRODUCTION

This test was conducted to demonstrate the feasability of using an Aquadyne hydro-blaster to remove the TRU which has accumulated on the K Basin fuel storage canisters during the years of storage. The TRU remaining on the decontaminated canister must be less than the 100 nCi/g limit in order for the canister to be considered non-TRU. The basis for a successful test is that the TRU remaining on the canisters after decontamination is below TRU limits (measured at 90% confidence).

Environmental and Engineering Demonstration Laboratory prepared a test plan titled "Feasability Study for High Pressure Cleaning of Fuel Canisters", WHC-SD-NR-AP-001 and directed the testing. Fuel and K/D Operations funded the test and provided operators and other support. The TRU measurements were made by Battelle Pacific Northwest Laboratories.

The testing started on June 25, 1990 and was completed July 24, 1990.

2.0 DESCRIPTION OF TEST

Twelve Mark 0 aluminum canisters were placed in 55 gallon drums. One of the twelve canisters had not been used. Irradiated N-Reactor fuel had been stored in the other eleven canisters for several years until it was processed by PUREX. The empty canisters were returned and stored in the basins. The canisters which were used in this test had been stored at N Basin for a number of years.

The canisters were packaged in plastic, surveyed, and each one was placed in a plastic-lined drum and the lid fastened on. Each drum was labeled with an identification number from one to twelve.

All of the 12 canisters were neutron counted for TRU contamination prior to cleaning. In addition, canisters #7 and #10 were gamma counted. Canister #1 was the unused canister and counted for the background measurement.

Canisters #7 and #10 were each hydro-blasted for 15 minutes. Then the TRU and gamma measurements were taken. About one-third of the visable corrosion products were removed. There was no reduction of the TRU.

Canisters #7 and #10 were cleaned again. Canister #7 was cleaned for about one hour more. The cleaning was

DRAFT

concentrated on the inside and inside bottom of the canister. When these canisters were remeasured, #7 had a marked reduction of TRU.

After reviewing these results, the cleaning method was revised. A wire chimney brush with the same diameter as the canister was obtained to scrub the insides and bottoms of the canisters. It was attached to a power drill with an extention about 7 feet in length so scrubbing could take place under water. The canisters were first scrubbed with the brush on the inside and bottom for several minutes, then hydro-blasted on all the surfaces with the Aquadyne power washer. Total cleaning time for each canister was 15 minutes.

Ten of the canisters were cleaned using this procedure. (All of the canisters except #7.) When the cleaning was completed TRU measurements were taken again.

3.0 TEST RESULTS

The residual transuranics on all of the canisters was reduced to less than the TRU limit of 100 nCi/g. Canister #7 had a TRU level of 89.9 nCi/g after decontamination. It was the only canister which had a TRU level close to the limit and was the only one which was not cleaned with the brush prior to the hydro-blasting. The TRU levels remaining on the canisters which were cleaned with the brush prior to hydro-blasting ranged from 9 to 46 nCi/g. Table 1 shows the TRU results.

Table 1 Canister TRU Concentration (nCi/g)

Canister #	Before Decon	After Decon
2	725	<32
3	942	26.9
4	493	<30
5	1020	<30
6	827	9.02
7	989	89.8
8	849	37.3
9	1110	46.0
10	853	46.3
11	453	<31
12	2680	18.9

Canisters #7 and #10 were also gamma counted. Table 2 shows the gamma emitters before and after decontamination.

DRAFT

Approximately 75% of these isotopes were removed from canister #7 and 56% were removed from canister #10.

Table 2 Gamma Emmitters Before and After Decontamination

Isotope	#7 before	#7 after	#10 before	#10 after
Mn-54	4.27	1.91	0.994	0.865
Co-60	148	67.7	80.3	31.9
Sb-125	198	84.8	89.8	61.3
Cs-137	1330	222	387	154
Eu-154	56.3	28.4	46.5	25.0
Eu-155	<13	<4.2	27.6	<3.8

4.0 CONCLUSIONS AND RECOMMENDATIONS

This feasability test has shown that the TRU on the canisters can be reduced to less than the TRU limits of 100 nCi/g by the cleaning method described in this test. It is recommended that a procedure and criteria be developed to apply this method for use on the remaining canisters.

APPENDIX A-4

EFFECTS OF HYDROSCRUBBING

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APPENDIX A-4

EFFECTS OF HYDROSCRUBBING

CONVERSATION RECORD

Speaker: John Karns, Westinghouse Hanford Company

Documented by: Linda Mihalik, CH2M HILL

Date of Conversation: December 12, 1993

Mr. Karns addressed the issue of aerosol generated during high-pressure washing activities. He stated that hydroscrubbing of contaminated surfaces had been performed frequently in the past in the 100 Areas. Operators were located 10 to 20 feet from the spray surface, and the air in the vicinity of the operators was monitored routinely by Health Physics Technicians. According to Mr. Karns, the only facial protection worn by the operators was a splash shield. Airborne contaminant levels at that distance were below levels at which respiratory protection was required.

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APPENDIX B

THERMAL EVALUATION OF PLASMA TORCH CUTTING

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APPENDIX B

THERMAL EVALUATION OF PLASMA TORCH CUTTING

A plasma torch generates temperatures of 11,093 to 27,760 degrees celsius in the ionized gas that is used for cutting (McGough and Knetl 1990). This temperature range is well above the boiling point of the radionuclides in the basin. For example, the boiling point of pure cesium is 678 degrees celsius; the oxide decomposes to the pure substance at 650 degrees celsius (CRC Press 1980). The boiling points for the pure substances and compounds of americium, cobalt, manganese, and plutonium, and strontium are higher, but still below the cutting temperatures.

The high cutting temperatures raise the following three questions:

- Can radionuclide gases be dispersed to the basin air space?
- To what extent could cutting increase evaporation of the basin water?
- What quantity of water could be heated to 100 degrees celsius or greater (required for 40 CFR 61 calculations)?

The following information was obtained during actual operation of a torch on the Lower Core Support Assembly at Three Mile Island (McGough and Knetl 1990):

> The materials being cut were 1 to 2.5 inches thick, and cutting was performed under 35 feet of water. Material that was 2 inches thick or more required 180 volts and resulted in a torch travel speed of 7 to 8 inches per minute. Material less than 2 inches thick required a voltage of 140 volts and resulted in a torch travel speed of 12 inches per minute. The cutting current ranged from 450 to 860 amperes.

Combining the upper-bound voltage and amperage, the total power consumed by the torch is:

$$180 \text{ V} \times 860 \text{ amps} = 154{,}800 \text{ watts}$$

As indicated by the travel speeds, plasma torch cutting proceeds rapidly. Assuming that the torch is operated in the basin continuously for 8 hours, the total energy consumption during that time would be:

$$154{,}800 \text{ watts} \times 1 \text{ Joule/sec/watt} \times 8 \text{ h} \times 60 \text{ min/hr} \times 60 \text{ sec/min} = 4.5 \times 10^9 \text{ Joule}$$

This energy dissipates by several mechanisms: the temperature of the metal and surrounding water increase, water, and other constituents (including radionuclides) vaporize, and water dissociates into hydrogen and oxygen gases.

If all of this energy could be isolated and directed into uniformly heating the surrounding water to 100 degrees celsius, the maximum quantity of water heated would be:

$$\frac{4.5 \times 10^9 \text{ Joule} \times 1 \text{ ml/g} \times 1 \text{ l}/1000 \text{ ml}}{4.2 \text{ Joule/g/}^\circ\text{C} \times (100 - 25\ ^\circ\text{C})} = 14{,}000 \text{ l } (14 \text{ m}^3)$$

Initial temperature: 25 °C (77 °F)
Heat capacity of water: 4.2 Joule/g/°C (CRC Press 1980)

A volume of 14 cubic meters would be the equivalent of a sphere having a radius of approximately 1.5 meters (5 feet). Thus, in this worst-case scenario temperatures of 100 degrees celsius would be achieved in 14,000 liters of water and at a distance of up to 5 feet from the cutting point. This assumes an ideal system where the heat energy is uniformly distributed in the water and no energy dissipates by other mechanisms. In reality, the heat energy could not be "confined" in a volume around the torch, a temperature gradient would be established (temperatures near the cutting point would be higher than 100 degrees celsius and temperatures at a distance of 5 feet would be lower), and some of the energy would dissipate by other mechanisms.

If all of the energy were used to vaporize water, the maximum quantity of water evaporated would be:

$$\frac{4.5 \times 10^9 \text{ Joule} \times 18 \text{ g/mol} \times 1 \text{ ml/g} \times 1 \text{ l}/1000 \text{ ml}}{4.1\text{E+04 Joule/mol} + 76 \text{ Joule/mol/}^\circ\text{C} * (100^\circ\text{C} - 25^\circ\text{C})} = 1{,}700 \text{ l}$$

Initial temperature: 25 °C
Heat capacity of water: 4.2 Joule/g/°C (76 Joule/mol/°C)
Heat of vaporization (at 100 °C): 4.1E+04 Joule/mol (Reference b)

Again, this is a worst-case scenario. In reality, some of the heat energy would dissipate by other mechanisms.

Finally, although cutting temperatures suggest that gaseous radionuclides can be generated at the point of cutting, the evaluation of water temperatures indicates that the gaseous species cannot be sustained as the gases transfer away from the torch. The radionuclides only exist as the gaseous species so long as they remain at the elevated temperatures. However, as soon as the gases move away from the torch, they contact much cooler water. A temperature differential of several hundred degrees exists between small gas bubbles and the cooler surrounding water for no more than a few seconds. As demonstrated previously, temperatures in the surrounding water will be less than 100 degrees celsius at distances of only 5 feet from the torch. Thus, radionuclides vaporized at the cutting surface will recondense before they can contribute as gases to the source term above the water. However, the condensed particulate from the radionuclides continues to be a potential source term contributor.

REFERENCES

CRC Press, 1980-81, *Handbook of Chemistry and Physics, 61st Edition.*

McGough, M. S., and G. J. Knetl/PCI Energy Services, 1990, "Underwater Plasma Cutting of the Lower Core Assembly and Metallurgical Sample of the Bottom head at Three Mile Island Unit 2," *Excellent and Economic Nuclear Plan Performance*, NE-Vol. 4, Book No. 100305.

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APPENDIX C

CAP-88 RESULTS 105N BASIN STABILIZATION

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January 19, 1994

To: Linda Mihalik

From: Paul D. Rittmann

Westinghouse Hanford Company HO-36

376-8715 FAX: 376-1293

Number of pages attached to this cover: **9**

The attached CAP88PC run uses wind data from 1983 to 1991 (9 years) at an elevation of 89 meters. This is appropriate for a 200 foot stack release.

The receptor location from the CAP88PC output is 14.7 km east of 100N Area. This is still onsite. The location is fictitious. It had to be entered this way because CAP88PC always picks the wrong wind transport direction. The actual worst case location is 11.5 km west of 100N Area.

The dose factors you need are listed on page 2 of the SUMMARY. I marked the nuclide chains on that page. The doses in a chain need to be added together. For example, the dose factor for Sr-90 is really the sum of 3.85E-02 and 7.17E-05 mrem, or 3.86E-02 mrem per curie released over one year.

I will send all of the CAP88PC output attached to a ccMail message.

C A P 8 8 - P C

Version 1.00

Clean Air Act Assessment Package - 1988

D O S E A N D R I S K E Q U I V A L E N T S U M M A R I E S

Non-Radon Individual Assessment
Jan 19, 1994 10:27 am

Facility: 100 Area - 200 ft Stack - Individual Nuclides
Address: Westinghouse Hanford Company
P.O. Box 1970
City: Richland
State: WA Zip: 99352-1970

Source Category:
Source Type: Stack
Emission Year: 1994

Comments: 14.7 km E is equivalent to 11.5 km W
Wind data generated at 100N from 1983 to 1991

Dataset Name: 100N for Linda
Dataset Date: Jan 19, 1994 10:27 am
Wind File: WNDFILES\JF10089.WND

ORGAN DOSE EQUIVALENT SUMMARY
(RN-222 Working Level Calculations Excluded)

Organ	Selected Individual (mrem/y)
GONADS	2.55E+00
BREAST	9.82E-01
R MAR	1.44E+01
LUNGS	4.97E+01
THYROID	9.87E-01
ENDOST	1.71E+02
RMNDR	8.00E+00
EFFEC	1.60E+01

Radon Decay Product Concentration (working level)

0.00E+00

PATHWAY EFFECTIVE DOSE EQUIVALENT SUMMARY
(RN-222 Working Level Calculations Excluded)

Pathway	Selected Individual (mrem/y)
INGESTION	1.94E+00
INHALATION	1.35E+01
AIR IMMERSION	2.64E-05
GROUND SURFACE	5.77E-01
INTERNAL	1.55E+01
EXTERNAL	5.77E-01
TOTAL	1.60E+01

Radon Decay Product Concentration (working level)

0.00E+00

NUCLIDE EFFECTIVE DOSE EQUIVALENT SUMMARY
(RN-222 Working Level Calculations Excluded)

Nuclide	Selected Individual (mrem/y)
AM-241	4.25E+00
CE-144	4.27E-03
PR-144	1.53E-07
CO-60	5.71E-02
CS-134	3.14E-02
CS-137	1.37E-02
BA-137M	4.59E-02
H-3	6.98E-06
K-40	4.53E-02
MN-54	3.65E-03
PU-238	2.58E+00
PU-239	2.78E+00
PU-241	4.39E-02
RU-106	5.57E-03
RH-106	5.65E-29
SB-125	5.78E-03
TE-125M	9.12E-05
SR-90	3.85E-02
Y-90	7.17E-05
U-235	9.93E-01
TH-231	7.21E-06
U-238	9.29E-01
TH-234	7.27E-04
PA-234	2.14E-01
U-234	1.04E+00
TH-230	1.86E+00
RA-226	1.68E-01
RN-222	0.00E+00
PO-218	0.00E+00
PB-214	3.08E-02
BI-214	1.63E-01
PO-214	0.00E+00
PB-210	5.30E-01
BI-210	1.44E-03
PO-210	1.84E-01
TOTAL	1.60E+01

Radon Decay Product Concentration (working level)

0.00E+00

CANCER RISK SUMMARY

Cancer	Selected Individual Total Lifetime Fatal Cancer Risk
LEUKEMIA	1.40E-05
BONE	7.67E-06
THYROID	3.58E-07
BREAST	3.02E-06
LUNG	8.46E-05
STOMACH	1.92E-06
BOWEL	1.06E-06
LIVER	2.41E-05
PANCREAS	1.32E-06
URINARY	2.19E-06
OTHER	1.62E-06
TOTAL	1.42E-04

	Selected Individual Cancer Risk
Radon Decay Product Lung Exposure	0.00E+00
Total Fatal Risk All Exposures	1.42E-04

PATHWAY RISK SUMMARY

Pathway	Selected Individual Total Lifetime Fatal Cancer Risk
INGESTION	1.22E-05
INHALATION	1.16E-04
AIR IMMERSION	6.34E-10
GROUND SURFACE	1.38E-05
INTERNAL	1.28E-04
EXTERNAL	1.38E-05
TOTAL	1.42E-04

	Selected Individual Cancer Risk
Radon Decay Product Lung Exposure	0.00E+00
Total Fatal Risk All Exposures	1.42E-04

NUCLIDE RISK SUMMARY

Nuclide	Selected Individual Total Lifetime Fatal Cancer Risk
AM-241	2.17E-05
CE-144	1.84E-07
PR-144	4.30E-12
CO-60	1.43E-06
CS-134	7.96E-07
CS-137	3.58E-07
BA-137M	1.10E-06
H-3	1.89E-10
K-40	1.14E-06
MN-54	8.76E-08
PU-238	2.17E-05
PU-239	2.15E-05
PU-241	1.67E-07
RU-106	2.40E-07
RH-106	1.36E-33
SB-125	1.38E-07
TE-125M	1.57E-09
SR-90	6.47E-07
Y-90	2.43E-09
U-235	1.29E-05
TH-231	2.10E-10
U-238	1.19E-05
TH-234	2.04E-08
PA-234	5.12E-06
U-234	1.32E-05
TH-230	1.52E-05
RA-226	1.98E-06
RN-222	0.00E+00
PO-218	0.00E+00
PB-214	7.28E-07
BI-214	3.94E-06
PO-214	0.00E+00
PB-210	3.78E-06
BI-210	3.84E-08
PO-210	1.86E-06
TOTAL	1.42E-04

	Selected Individual Cancer Risk
Radon Decay Product Lung Exposure	0.00E+00
Total Fatal Risk All Exposures	1.42E-04

INDIVIDUAL EFFECTIVE DOSE EQUIVALENT RATE (mrem/y)
(All Radionuclides and Pathways)

	Distance (m)
Direction	14700
N	6.9E+00
NNW	5.6E+00
NW	7.5E+00
WNW	8.8E+00
W	1.2E+01
WSW	6.0E+00
SW	5.3E+00
SSW	5.5E+00
S	8.3E+00
SSE	6.6E+00
SE	7.8E+00
ESE	1.0E+01
E	1.6E+01
ENE	1.0E+01
NE	7.4E+00
NNE	5.3E+00

INDIVIDUAL LIFETIME RISK (deaths)
(All Radionuclides and Pathways)

	Distance (m)
Direction	14700
N	6.1E-05
NNW	4.9E-05
NW	6.6E-05
WNW	7.8E-05
W	1.1E-04
WSW	5.3E-05
SW	4.7E-05
SSW	4.9E-05
S	7.3E-05
SSE	5.9E-05
SE	6.9E-05
ESE	9.1E-05
E	1.4E-04
ENE	9.0E-05
NE	6.5E-05
NNE	4.7E-05

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Version 1.00

Clean Air Act Assessment Package - 1988

S Y N O P S I S R E P O R T

Non-Radon Individual Assessment
Jan 19, 1994 10:27 am

Facility: 100 Area - 200 ft Stack - Individual Nuclides
Address: Westinghouse Hanford Company
P.O. Box 1970
City: Richland
State: WA Zip: 99352-1970

Effective Dose Equivalent (mrem/year)
1.60E+01

At This Location: 14700 Meters East

Source Category:
Source Type: Stack
Emission Year: 1994

Comments: 14.7 km E is equivalent to 11.5 km W
Wind data generated at 100N from 1983 to 1991

Dataset Name: 100N for Linda
Dataset Date: Jan 19, 1994 10:27 am
Wind File: WNDFILES\JF10089.WND

MAXIMALLY EXPOSED INDIVIDUAL
(RN-222 Working Level Calculations Excluded)

Location Of The Individual: 14700 Meters East
Lifetime Fatal Cancer Risk: 1.42E-04

ORGAN DOSE EQUIVALENT SUMMARY
(RN-222 Working Level Calculations Excluded)

Organ	Dose Equivalent (mrem/y)
GONADS	2.55E+00
BREAST	9.82E-01
R MAR	1.44E+01
LUNGS	4.97E+01
THYROID	9.87E-01
ENDOST	1.71E+02
RMNDR	8.00E+00
EFFEC	1.60E+01

RADIONUCLIDE EMISSIONS DURING THE YEAR 1994

Nuclide	Class	Size	Source #1 Ci/y	TOTAL Ci/y
AM-241	W	1.00	1.0E+00	1.0E+00
CE-144	Y	1.00	1.0E+00	1.0E+00
PR-144	Y	1.00	1.0E+00	1.0E+00
CO-60	Y	1.00	1.0E+00	1.0E+00
CS-134	D	1.00	1.0E+00	1.0E+00
CS-137	D	1.00	1.0E+00	1.0E+00
BA-137M	D	1.00	9.5E-01	9.5E-01
H-3	*	0.00	1.0E+00	1.0E+00
K-40	D	1.00	1.0E+00	1.0E+00
MN-54	W	1.00	1.0E+00	1.0E+00
PU-238	Y	1.00	1.0E+00	1.0E+00
PU-239	Y	1.00	1.0E+00	1.0E+00
PU-241	Y	1.00	1.0E+00	1.0E+00
RU-106	Y	1.00	1.0E+00	1.0E+00
RH-106	Y	1.00	1.0E+00	1.0E+00
SB-125	W	1.00	1.0E+00	1.0E+00
TE-125M	W	1.00	2.5E-01	2.5E-01
SR-90	D	1.00	1.0E+00	1.0E+00
Y-90	Y	1.00	1.0E+00	1.0E+00
U-235	Y	1.00	1.0E+00	1.0E+00
TH-231	Y	1.00	1.0E+00	1.0E+00
U-238	Y	1.00	1.0E+00	1.0E+00
TH-234	Y	1.00	1.0E+00	1.0E+00
PA-234	Y	1.00	1.6E-03	1.6E-03
U-234	Y	1.00	1.0E+00	1.0E+00
TH-230	Y	1.00	1.0E+00	1.0E+00
RA-226	W	1.00	1.0E+00	1.0E+00
RN-222	*	0.00	0.0E+00	0.0E+00
PO-218	W	1.00	0.0E+00	0.0E+00
PB-214	D	1.00	0.0E+00	0.0E+00
BI-214	W	1.00	0.0E+00	0.0E+00
PO-214	W	1.00	0.0E+00	0.0E+00
PB-210	D	1.00	1.0E+00	1.0E+00
BI-210	W	1.00	1.0E+00	1.0E+00
PO-210	W	1.00	1.0E+00	1.0E+00

SITE INFORMATION

Temperature: 12 degrees C
Precipitation: 16 cm/y
Mixing Height: 1000 m

SOURCE INFORMATION

Source Number: 1

Stack Height (m): 89.00
Diameter (m): 2.50

Plume Rise Pasquill Cat:	A	B	C	D	E	F	G
Fixed (m): (Fixed Rise)	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

AGRICULTURAL DATA

	Vegetable	Milk	Meat
Fraction Home Produced:	1.000	1.000	1.000
Fraction From Assessment Area:	0.000	0.000	0.000
Fraction Imported:	0.000	0.000	0.000

Food Arrays were not generated for this run.
Default Values used.

DISTANCES USED FOR MAXIMUM INDIVIDUAL ASSESSMENT

14700

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Clean Air Act Assessment Package - 1988

W E A T H E R D A T A

Non-Radon Individual Assessment
Jan 19, 1994 10:27 am

Facility: 100 Area - 200 ft Stack - Individual Nuclides
Address: Westinghouse Hanford Company
P.O. Box 1970
City: Richland
State: WA Zip: 99352-1970

Source Category:
Source Type: Stack
Emission Year: 1994

Comments: 14.7 km E is equivalent to 11.5 km W
Wind data generated at 100N from 1983 to 1991

Dataset Name: 100N for Linda
Dataset Date: Jan 19, 1994 10:27 am
Wind File: WNDFILES\JF10089.WND

HARMONIC AVERAGE WIND SPEEDS (WIND TOWARDS)

Dir	Pasquill Stability Class A	B	C	D	E	F	G	Wind Frequency
N	1.681	1.778	1.424	1.547	1.767	1.458	1.262	0.038
NNW	1.968	1.531	1.795	1.713	1.845	1.645	1.353	0.033
NW	2.199	1.558	1.683	1.770	2.091	1.660	1.389	0.048
WNW	1.743	1.586	1.520	1.605	1.941	1.756	1.504	0.054
W	1.653	1.561	1.581	1.474	1.730	1.785	1.574	0.073
WSW	1.628	1.611	1.430	1.637	1.865	1.683	1.539	0.039
SW	2.397	2.113	1.931	2.033	2.088	1.621	1.320	0.040
SSW	2.757	2.744	2.372	2.301	2.378	1.917	1.326	0.049
S	1.956	1.904	1.642	1.873	2.104	1.670	1.364	0.060
SSE	1.752	1.576	1.448	1.698	1.772	1.548	1.162	0.045
SE	2.159	1.960	1.827	2.129	2.310	1.586	1.244	0.062
ESE	2.703	2.430	2.307	3.002	3.525	1.811	1.351	0.104
E	2.605	2.417	2.292	2.912	3.890	2.171	1.615	0.166
ENE	2.612	2.309	1.957	2.644	3.097	2.128	1.705	0.094
NE	2.947	2.303	2.360	2.348	2.426	1.669	1.476	0.060
NNE	2.204	1.720	1.625	2.087	2.003	1.475	1.433	0.035

ARITHMETIC AVERAGE WIND SPEEDS (WIND TOWARDS)

Dir	Pasquill Stability Class A	B	C	D	E	F	G
N	3.055	4.038	2.806	3.127	3.657	2.710	2.001
NNW	3.197	2.857	3.302	3.315	3.810	2.914	2.224
NW	3.496	2.507	2.786	3.216	3.940	3.263	2.214
WNW	2.701	2.634	2.360	2.631	3.295	3.092	2.887
W	2.501	2.438	2.485	2.329	3.051	3.349	2.745
WSW	2.826	2.635	2.184	2.886	3.556	3.407	2.812
SW	4.081	4.060	3.535	3.917	4.984	3.311	2.033
SSW	4.798	5.882	5.439	5.408	6.371	5.504	2.492
S	3.393	4.088	3.766	4.096	5.307	3.608	2.699
SSE	2.734	2.531	2.241	3.279	3.882	3.090	1.773
SE	4.842	4.313	4.417	5.177	5.585	3.079	1.929
ESE	5.637	5.804	5.862	6.890	7.308	3.667	2.278
E	4.670	4.738	4.425	5.814	6.756	4.291	2.844
ENE	4.737	4.655	4.270	5.069	5.679	4.300	3.064
NE	5.452	5.401	5.543	5.575	5.305	3.159	2.555
NNE	4.244	3.531	4.136	5.411	4.776	2.743	2.319

FREQUENCIES OF STABILITY CLASSES (WIND TOWARDS)

	Pasquill Stability Class						
Dir	A	B	C	D	E	F	G
N	0.1082	0.0449	0.0475	0.3245	0.2797	0.1372	0.0580
NNW	0.1064	0.0547	0.0547	0.3283	0.2705	0.1337	0.0517
NW	0.1151	0.0460	0.0523	0.3326	0.2552	0.1485	0.0502
WNW	0.1065	0.0467	0.0449	0.3159	0.2505	0.1701	0.0654
W	0.0941	0.0409	0.0382	0.2428	0.2510	0.2360	0.0969
WSW	0.0876	0.0335	0.0438	0.2294	0.2500	0.2423	0.1134
SW	0.1796	0.0574	0.0599	0.2519	0.2319	0.1546	0.0648
SSW	0.2037	0.0679	0.0576	0.2840	0.2016	0.1337	0.0514
S	0.2020	0.0546	0.0579	0.2781	0.2086	0.1457	0.0530
SSE	0.1955	0.0517	0.0494	0.2584	0.1978	0.1708	0.0764
SE	0.2006	0.0546	0.0498	0.2648	0.2103	0.1509	0.0690
ESE	0.1675	0.0443	0.0404	0.2772	0.2810	0.1299	0.0597
E	0.1210	0.0361	0.0301	0.2541	0.3311	0.1481	0.0795
ENE	0.1223	0.0383	0.0319	0.2660	0.2915	0.1734	0.0766
NE	0.1581	0.0516	0.0499	0.3111	0.2429	0.1248	0.0616
NNE	0.1203	0.0372	0.0458	0.3410	0.2550	0.1404	0.0602
TOT	0.1424	0.0457	0.0438	0.2781	0.2621	0.1579	0.0698

ADDITIONAL WEATHER INFORMATION

Average Air Temperature: 12.0 degrees C
285.2 K
Precipitation: 16.0 cm/y
Lid Height: 1000 meters
Surface Roughness Length: 0.010 meters
Height Of Wind Measurements: 10.0 meters
Average Wind Speed: 4.401 m/s

Vertical Temperature Gradients:
STABILITY E 0.073 k/m
STABILITY F 0.109 k/m
STABILITY G 0.146 k/m

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Clean Air Act Assessment Package - 1988

G E N E R A L D A T A

Non-Radon Individual Assessment
Jan 19, 1994 10:27 am

Facility: 100 Area - 200 ft Stack - Individual Nuclides
Address: Westinghouse Hanford Company
P.O. Box 1970
City: Richland
State: WA Zip: 99352-1970

Source Category:
Source Type: Stack
Emission Year: 1994

Comments: 14.7 km E is equivalent to 11.5 km W
Wind data generated at 100N from 1983 to 1991

Dataset Name: 100N for Linda
Dataset Date: Jan 19, 1994 10:27 am
Wind File: WNDFILES\JF10089.WND

VALUES FOR RADIONUCLIDE-DEPENDENT PARAMETERS

Nuclide	Clearance Class	Particle Size (microns)	Scavenging Coefficient (per second)	Dry Deposition Velocity (m/s)
AM-241	W	1.0	1.60E-06	1.80E-03
CE-144	Y	1.0	1.60E-06	1.80E-03
PR-144	Y	1.0	1.60E-06	1.80E-03
CO-60	Y	1.0	1.60E-06	1.80E-03
CS-134	D	1.0	1.60E-06	1.80E-03
CS-137	D	1.0	1.60E-06	1.80E-03
BA-137M	D	1.0	1.60E-06	1.80E-03
H-3	*	0.0	0.00E+00	0.00E+00
K-40	D	1.0	1.60E-06	1.80E-03
MN-54	W	1.0	1.60E-06	1.80E-03
PU-238	Y	1.0	1.60E-06	1.80E-03
PU-239	Y	1.0	1.60E-06	1.80E-03
PU-241	Y	1.0	1.60E-06	1.80E-03
RU-106	Y	1.0	1.60E-06	1.80E-03
RH-106	Y	1.0	1.60E-06	1.80E-03
SB-125	W	1.0	1.60E-06	1.80E-03
TE-125M	W	1.0	1.60E-06	1.80E-03
SR-90	D	1.0	1.60E-06	1.80E-03
Y-90	Y	1.0	1.60E-06	1.80E-03
U-235	Y	1.0	1.60E-06	1.80E-03
TH-231	Y	1.0	1.60E-06	1.80E-03
U-238	Y	1.0	1.60E-06	1.80E-03
TH-234	Y	1.0	1.60E-06	1.80E-03
PA-234	Y	1.0	1.60E-06	1.80E-03
U-234	Y	1.0	1.60E-06	1.80E-03
TH-230	Y	1.0	1.60E-06	1.80E-03
RA-226	W	1.0	1.60E-06	1.80E-03
RN-222	*	0.0	0.00E+00	0.00E+00
PO-218	W	1.0	1.60E-06	1.80E-03
PB-214	D	1.0	1.60E-06	1.80E-03
BI-214	W	1.0	1.60E-06	1.80E-03
PO-214	W	1.0	1.60E-06	1.80E-03
PB-210	D	1.0	1.60E-06	1.80E-03
BI-210	W	1.0	1.60E-06	1.80E-03
PO-210	W	1.0	1.60E-06	1.80E-03

VALUES FOR RADIONUCLIDE-DEPENDENT PARAMETERS

	DECAY CONSTANT (PER DAY)			TRANSFER COEFFICIENT	
Nuclide	Radio-active (1)	Surface	Water	Milk (2)	Meat (3)
AM-241	0.00E+00	5.48E-05	0.00E+00	4.00E-07	3.50E-06
CE-144	0.00E+00	5.48E-05	0.00E+00	2.00E-05	7.50E-04
PR-144	5.78E+01	5.48E-05	0.00E+00	2.00E-05	3.00E-04
CO-60	0.00E+00	5.48E-05	0.00E+00	2.00E-03	2.00E-02
CS-134	0.00E+00	5.48E-05	0.00E+00	7.00E-03	2.00E-02
CS-137	0.00E+00	5.48E-05	0.00E+00	7.00E-03	2.00E-02
BA-137M	3.91E+02	5.48E-05	0.00E+00	3.50E-04	1.50E-04
H-3	0.00E+00	5.48E-05	0.00E+00	0.00E+00	0.00E+00
K-40	0.00E+00	5.48E-05	0.00E+00	7.00E-03	2.00E-02
MN-54	0.00E+00	5.48E-05	0.00E+00	3.50E-04	4.00E-04
PU-238	0.00E+00	5.48E-05	0.00E+00	1.00E-07	5.00E-07
PU-239	0.00E+00	5.48E-05	0.00E+00	1.00E-07	5.00E-07
PU-241	0.00E+00	5.48E-05	0.00E+00	1.00E-07	5.00E-07
RU-106	0.00E+00	5.48E-05	0.00E+00	6.00E-07	2.00E-03
RH-106	2.00E+03	5.48E-05	0.00E+00	1.00E-02	2.00E-03
SB-125	0.00E+00	5.48E-05	0.00E+00	1.00E-04	1.00E-03
TE-125M	1.20E-02	5.48E-05	0.00E+00	2.00E-04	1.50E-02
SR-90	0.00E+00	5.48E-05	0.00E+00	1.50E-03	3.00E-04
Y-90	2.60E-01	5.48E-05	0.00E+00	2.00E-05	3.00E-04
U-235	0.00E+00	5.48E-05	0.00E+00	6.00E-04	2.00E-04
TH-231	6.52E-01	5.48E-05	0.00E+00	5.00E-06	6.00E-06
U-238	0.00E+00	5.48E-05	0.00E+00	6.00E-04	2.00E-04
TH-234	2.88E-02	5.48E-05	0.00E+00	5.00E-06	6.00E-06
PA-234	2.48E+00	5.48E-05	0.00E+00	5.00E-06	1.00E-05
U-234	0.00E+00	5.48E-05	0.00E+00	6.00E-04	2.00E-04
TH-230	0.00E+00	5.48E-05	0.00E+00	5.00E-06	6.00E-06
RA-226	0.00E+00	5.48E-05	0.00E+00	4.50E-04	2.50E-04
RN-222	1.81E-01	5.48E-05	0.00E+00	0.00E+00	0.00E+00
PO-218	3.27E+02	5.48E-05	0.00E+00	3.50E-04	9.50E-05
PB-214	3.72E+01	5.48E-05	0.00E+00	2.50E-04	3.00E-04
BI-214	5.02E+01	5.48E-05	0.00E+00	5.00E-04	4.00E-04
PO-214	3.66E+08	5.48E-05	0.00E+00	3.50E-04	9.50E-05
PB-210	0.00E+00	5.48E-05	0.00E+00	2.50E-04	3.00E-04
BI-210	1.38E-01	5.48E-05	0.00E+00	5.00E-04	4.00E-04
PO-210	0.00E+00	5.48E-05	0.00E+00	3.50E-04	9.50E-05

FOOTNOTES:

(1) Effective radioactive decay constant in plume; set to zero if less than 1.0E-2

(2) Fraction of animal's daily intake of nuclide which appears in each L of milk (days/L)

(3) Fraction of animal's daily intake of nuclide which appears in each kg of meat (days/kg)

VALUES FOR RADIONUCLIDE-DEPENDENT PARAMETERS

	CONCENTRATION UPTAKE FACTOR		GI UPTAKE FRACTION	
Nuclide	Forage (1)	Edible (2)	Inhalation	Ingestion
AM-241	5.50E-03	1.07E-04	1.00E-03	1.00E-03
CE-144	1.00E-02	1.71E-03	3.00E-04	3.00E-04
PR-144	1.00E-02	1.71E-03	3.00E-04	3.00E-04
CO-60	2.00E-02	3.00E-03	5.00E-02	3.00E-01
CS-134	8.00E-02	1.28E-02	9.50E-01	9.50E-01
CS-137	8.00E-02	1.28E-02	9.50E-01	9.50E-01
BA-137M	1.50E-01	6.42E-03	1.00E-01	1.00E-01
H-3	0.00E+00	0.00E+00	9.50E-01	9.50E-01
K-40	1.00E+00	2.35E-01	9.50E-01	9.50E-01
MN-54	2.50E-01	2.14E-02	1.00E-01	1.00E-01
PU-238	4.50E-04	1.93E-05	1.00E-03	1.00E-03
PU-239	4.50E-04	1.93E-05	1.00E-04	1.00E-03
PU-241	4.50E-04	1.93E-05	1.00E-03	1.00E-03
RU-106	7.50E-02	8.56E-03	5.00E-02	5.00E-02
RH-106	1.50E-01	1.71E-02	5.00E-02	5.00E-02
SB-125	2.00E-01	1.28E-02	1.00E-02	1.00E-01
TE-125M	2.50E-02	1.71E-03	2.00E-01	2.00E-01
SR-90	2.50E+00	1.07E-01	3.00E-01	3.00E-01
Y-90	1.50E-02	2.57E-03	1.00E-04	1.00E-04
U-235	8.50E-03	1.71E-03	2.00E-03	2.00E-01
TH-231	8.50E-04	3.64E-05	2.00E-04	2.00E-04
U-238	8.50E-03	1.71E-03	2.00E-03	2.00E-01
TH-234	8.50E-04	3.64E-05	2.00E-04	2.00E-04
PA-234	2.50E-03	1.07E-04	1.00E-03	1.00E-03
U-234	8.50E-03	1.71E-03	2.00E-03	2.00E-01
TH-230	8.50E-04	3.64E-05	2.00E-04	2.00E-04
RA-226	1.50E-02	6.42E-04	2.00E-01	2.00E-01
RN-222	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PO-218	2.50E-03	1.71E-04	1.00E-01	1.00E-01
PB-214	4.50E-02	3.85E-03	2.00E-01	2.00E-01
BI-214	3.50E-02	2.14E-03	5.00E-02	5.00E-02
PO-214	2.50E-03	1.71E-04	1.00E-01	1.00E-01
PB-210	4.50E-02	3.85E-03	2.00E-01	2.00E-01
BI-210	3.50E-02	2.14E-03	5.00E-02	5.00E-02
PO-210	2.50E-03	1.71E-04	1.00E-01	1.00E-01

FOOTNOTES: (1) Concentration factor for uptake of nuclide from soil for pasture and forage (in pCi/kg dry weight per pCi/kg dry soil)

(2) Concentration factor for uptake of nuclide from soil by edible parts of crops (in pCi/kg wet weight per pCi/kg dry soil)

H-3 DOSE CONVERSION FACTOR FOR WATER INGESTION (rem-cc/pCi-y): 5.70E-02

DECAY CHAIN INGROWTH FACTORS

Nuclide	Parent(s)	Ingrowth Factor(s)
BA-137M	CS-137	3.209E+06
PA-234	TH-234	2.961E+04
RA-226	TH-230	1.490E-02
PB-214	TH-230	6.915E+03
	RA-226	4.635E+05
	PO-218	1.138E-01
BI-214	TH-230	9.313E+03
	RA-226	6.240E+05
	PO-218	1.532E-01
	PB-214	1.347E+00
PO-210	PB-210	3.560E+01
	BI-210	3.621E-02

VALUES FOR RADIONUCLIDE-INDEPENDENT PARAMETERS

HUMAN INHALATION RATE	
Cubic centimeters/hr	9.17E+05
SOIL PARAMETERS	
Effective surface density (kg/sq m, dry weight) (Assumes 15 cm plow layer)	2.15E+02
BUILDUP TIMES	
For activity in soil (years)	1.00E+02
For radionuclides deposited on ground/water (days)	3.65E+04
DELAY TIMES	
Ingestion of pasture grass by animals (hr)	0.00E+00
Ingestion of stored feed by animals (hr)	2.16E+03
Ingestion of leafy vegetables by man (hr)	3.36E+02
Ingestion of produce by man (hr)	3.36E+02
Transport time from animal feed-milk-man (day)	2.00E+00
Time from slaughter to consumption (day)	2.00E+01
WEATHERING	
Removal rate constant for physical loss (per hr)	2.90E-03
CROP EXPOSURE DURATION	
Pasture grass (hr)	7.20E+02
Crops/leafy vegetables (hr)	1.44E+03
AGRICULTURAL PRODUCTIVITY	
Grass-cow-milk-man pathway (kg/sq m)	2.80E-01
Produce/leafy veg for human consumption (kg/sq m)	7.16E-01
FALLOUT INTERCEPTION FRACTIONS	
Vegetables	2.00E-01
Pasture	5.70E-01
GRAZING PARAMETERS	
Fraction of year animals graze on pasture	4.00E-01
Fraction of daily feed that is pasture grass when animal grazes on pasture	4.30E-01

VALUES FOR RADIONUCLIDE-INDEPENDENT PARAMETERS

ANIMAL FEED CONSUMPTION FACTORS	
Contaminated feed/forage (kg/day, dry weight)	1.56E+01
DAIRY PRODUCTIVITY	
Milk production of cow (L/day)	1.10E+01
MEAT ANIMAL SLAUGHTER PARAMETERS	
Muscle mass of animal at slaughter (kg)	2.00E+02
Fraction of herd slaughtered (per day)	3.81E-03
DECONTAMINATION	
Fraction of radioactivity retained after washing for leafy vegetables and produce	5.00E-01
FRACTIONS GROWN IN GARDEN OF INTEREST	
Produce ingested	1.00E+00
Leafy vegetables ingested	1.00E+00
INGESTION RATIOS:	
IMMEDIATE SURROUNDING AREA/TOTAL WITHIN AREA	
Vegetables	1.00E+00
Meat	1.00E+00
Milk	1.00E+00
MINIMUM INGESTION FRACTIONS FROM OUTSIDE AREA (Minimum fractions of food types from outside area listed below are actual fixed values.)	
Vegetables	0.00E+00
Meat	0.00E+00
Milk	0.00E+00
HUMAN FOOD UTILIZATION FACTORS	
Produce ingestion (kg/y)	1.76E+02
Milk ingestion (L/y)	1.12E+02
Meat ingestion (kg/y)	8.50E+01
Leafy vegetable ingestion (kg/y)	1.80E+01
SWIMMING PARAMETERS	
Fraction of time spent swimming	0.00E+00
Dilution factor for water (cm)	1.00E+00

APPENDIX D

DISCUSSION OF BEST AVAILABLE RADIONUCLIDE CONTROL TECHNOLOGY

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APPENDIX D

DISCUSSION OF BEST AVAILABLE RADIONUCLIDE CONTROL TECHNOLOGY

With approval from the U.S. Department of Energy Richland Operations Office, the Washington State Department of Health (DOH) was contacted on January 8, 1994 by representatives of Westinghouse Hanford Company to discuss the BARCT assessment for basin stabilization. Westinghouse Hanford Company requested that the discussion of BARCT be prepared in a format similar to the discussion of BARCT as presented in the Notice of Construction entitled, "Radioactive Air Emissions Program Notice of Construction, Rotary Mode Core-Sampling Truck and Exhauster," (DOE/RL-93-40), submitted to DOH in May 1993. This format was requested based on the following points.

- The reroute of the air flow from the basin roof vents through the 117N Filter Building and discharge to the 116N Stack was approved by DOH on September 2, 1993. All radioactive air emissions, as a result of the basin stabilization activity, will be routed through the 117N HEPA Filter Building.

- From the total abated dose (2.16 E-04 millirems/year) contributing to the MEI located 14.7 kilometers east of 100-N, tritium contributes 38 percent. For tritium at such low levels (8.29 E-05 millirems/year ^{3}H), there are no controls economically achievable.

- In October 1993, DOH approved BARCT for a similar fuel basin located in the 100-K Area of the Hanford Site. The 105-KE Basin contains approximately 1,150 MTUs of irradiated fuel. 105N Basin does not contain any fuel elements. BARCT for basin stabilization, using the 117N HEPA Filter Building, exceeds the requirements for BARCT accepted at 105-KE Basin.

The DOH approved the use of the same format as used in the "Radioactive Air Emissions Program Notice of Construction, Rotary Mode Core-Sampling Truck and Exhauster," (DOE/RL-93-40), pending that the dose contribution from tritium was as presented in Chapter 4.0 of this Notice of Construction.

Prior to October 1993, the basin air space was exhausted directly to the atmosphere via roof exhausters and through the Zone II exhaust system. After receiving approval from DOH, the roof vents were closed and the basin air space was rerouted to Zone I. Zone I is vented through a HEPA filtration system located in the 117N Filter Building prior to discharge through the 116N Stack.

As a result, it is proposed that the filtration system as described Section 2.3 of this Notice of Construction be approved as BARCT for basin stabilization. This discussion of BARCT does not present a detailed evaluation of all controls economically achievable. The DOH has stated that HEPA filters are generally accepted as BARCT for particulate radionuclide air emissions. HEPA filter systems have been and are used extensively at the Hanford Site in the control of particulate radionuclide air emissions.

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