ARF-023608

Savannah River
NUCLEAR SOLUTIONS℠
FLUOR • NEWPORT NEWS NUCLEAR • HONEYWELL

Removal Site Evaluation Report / Engineering Evaluation/Cost Analysis for the F-Area Material Storage Building (235-F) (U)

SRNS-RP-2021-00001

Revision ~~0~~1 Redline

~~August 2021~~February 2022

DOES NOT CONTAIN
OFFICIAL USE ONLY INFORMATION

Name/Org: Dena Brett/SRNS Date: 2/9/2022

DISCLAIMER

Printed in the United States of America

Prepared for
U.S. Department of Energy
and
Savannah River Nuclear Solutions, LLC
Aiken, South Carolina

EXECUTIVE SUMMARY

The United States Department of Energy is proposing to perform a non-time critical removal action at the F-Area Material Storage Building (FAMS)(235-F), herein referred to as Building 235-F. Building 235-F is listed in the Savannah River Federal Facility Agreement on Appendix K.1: *D&D Facilities to be Decommissioned*. In 2019, the United States Department of Energy, United States Environmental Protection Agency, and South Carolina Department of Health and Environmental Control reached agreement that a non-time critical removal action under the Comprehensive Environmental Response, Compensation, and Liability Act is an appropriate regulatory mechanism to decommission Building 235-F. This Removal Site Evaluation Report/Engineering Evaluation/Cost Analysis identifies the objectives of the removal action for the Building 235-F, evaluates removal action alternatives that address the potential threats from release of contaminants to the environment, and provides a vehicle for public comment per the National Oil and Hazardous Substances Pollution Contingency Plan, 40 Code of Federal Regulations § 300.415.

Building 235-F is a windowless, two-story, reinforced-concrete structure located in F Area near the center of Savannah River Site. The building is approximately 68 meters (222 feet) long, 33 meters (109 feet) wide, and 8.5 meters (28 feet) high. Building 235-F was constructed in the 1950s as part of the original Savannah River Site project and used for a variety of missions, primarily processing, storage, and disbursement of radioactive materials in support of Savannah River Site and the Department of Energy complex. Building 235-F and support facilities are currently in a reduced surveillance and maintenance state and are undergoing deactivation activities and preparations for decommissioning. Building 235-F is designated a Hazard Category 2 non-reactor facility according to DOE STD-1027-92 due to the radiological contamination within the building, consisting primarily of neptunium-237 in the Actinide Billet Line and plutonium-238 in the Plutonium Fuel Form Facility, Old Metallurgical Laboratory, and the Plutonium Experimental Facility.

Two additional structures exterior to Building 235-F are also included in the scope of this proposed non-time critical removal action. They include an abandoned capped stack (293-F) located on the east side of the building and an underground storage tank connected by a pipe trench on the north side of the building. The underground storage tank previously contained radionuclide contamination (plutonium and tritium) and hazardous waste constituents (cadmium and chromium). The tank was emptied in 1991 by pumping out the liquid and sludge and cleaned by scraping and mopping the walls and bottom of the tank and flushing the tank. The tank inlet pipe has been capped.

A streamlined human health risk assessment was conducted for Building 235-F on the radiological inventory present in the building. The risk to the hypothetical future industrial worker was calculated to be 2.3E+09, primarily from plutonium-238 and neptunium-237 contamination in the process areas. The human health risk estimate for exposure to radiological contamination is significantly higher than the United States Environmental Protection Agency acceptable risk range of 1E-04 to1E-06. The contamination is considered principal threat source material (risk >1E-03). Other hazardous materials such as asbestos, lead, and polychlorinated biphenyls will remain in the building following deactivation. A risk evaluation for polychlorinated biphenyls and lead, based on maximum detected concentrations in building paint, was conducted to demonstrate the negligible risk contribution from these hazardous inventories when compared to the primary radiological risk drivers. Risk from exposure to hazardous materials is well bounded by the risk from exposure to plutonium-238 and neptunium-237 in the process areas.

Fate and transport modeling was conducted to simulate the migration of contaminants from Building 235-F through the vadose zone and groundwater over a long period of time. The model evaluated five different points of assessment (i.e., 1-meter [3-feet] beyond the outside perimeter of Building 235-F; 100-meters [328-feet] beyond the outside perimeter of Building 235-F; 360-meters [1,181-feet] [F Area industrial area boundary fence where it cross the projected plume path]; 683-meters [2,241-feet] [closest groundwater-fed seepline to Building 235-F along the plume path]; and the surface water in an unnamed tributary of the Upper Three Runs creek). Modeling results were reported for assessment periods of 0-1,000 years, 0-10,000 years, and 0-100,000 years. The United States Department of Energy, United States Environmental Protection Agency, and South Carolina Department of Health and Environmental Control agreed to consider the identification of contaminant migration problems warranting action based on an assessment period of 0-10,000 years at the 360-meter (1,181-feet) point of assessment.

The following removal action objectives were identified for Building 235-F to protect human health and the environment: (1) Prevent exposure of the hypothetical future industrial worker to radiological contaminants present in Building 235-F that exceed 1E-06 risk thresholds (including principal threat source material); and (2) Prevent the migration of radionuclide contamination from Building 235-F to groundwater at concentrations that exceed maximum contaminant levels ~~at the 360-meter (1,181-feet) point of assessment in less than 10,000 years~~ to the extent practicable.

Four non-time critical removal action alternatives were evaluated in this report.

- **Alternative A-1, No Action**: Building 235-F, underground storage tank, and abandoned capped stack (293-F) will remain as currently exist after completion of the deactivation scope. The roof is assumed to collapse at 150 years at which time additional response action(s) will likely be necessary to mitigate exposure/spread of contaminants. No Action is representative of the Building 235-F deactivated state.
- **Alternative A-2, In-situ Decommissioning of First and Second Level Process Areas/Engineered Roof**: First and second level process areas will be grouted. An engineered roof (sloped concrete reinforced roof slab with integral crystalline waterproofing) designed to last 1,000 years will be installed. The underground storage tank will be grouted/capped, and the abandoned capped stack (293-F) will be permanently sealed.
- **Alternative A-3, In-situ Decommissioning of Entire Building/Engineered Roof**: First and second levels (all areas) will be grouted. An engineered roof (sloped concrete reinforced roof slab with integral crystalline waterproofing) designed to last 1,000 years will be installed. The underground storage tank will be grouted/capped, and the abandoned capped stack (293-F) will be permanently sealed.
- **Alternative A-4, Complete Building 235-F Removal/Soil Cover**: Building 235-F will be demolished to the building slab. The abandoned capped stack (293-F) will be removed. The underground storage tank will be grouted/capped. A soil cover will be applied over both the building slab and grouted underground storage tank.

In accordance with United States Environmental Protection Agency guidance on conducting non-time critical removal actions, the four proposed removal action alternatives were evaluated against three broad criteria: effectiveness, implementability, and cost. The No Action alternative was found to be ineffective at preventing exposure of the hypothetical future industrial worker to radiological contaminants present in Building 235-F and ineffective at preventing groundwater contamination above maximum contaminant levels at the point of assessment in less than 10,000 years. Alternatives A-2, A-3, and A-4 all achieve the removal action objectives to prevent human exposure and contaminant migration to groundwater, although the short-term effectiveness for Alternative A-4 is low due to the risk to a decommissioning worker from potential exposure of airborne or surface contamination during demolition of the building. With respect to implementability, Alternatives A-1, A-2, and A-3 were technically and administratively feasible with Alternative A-2 offering more flexibility for a phased implementation strategy.

Based on the evaluation in this Removal Site Evaluation Report/Engineering Evaluation/Cost Analysis, the lead agency's preferred removal action alternative for Building 235-F is Alternative A-2, In-situ Decommissioning of First and Second Level Process Areas/Engineered Roof. Alternative A-2 will meet the removal action objectives and will be protective of human health and the environment in the short term and long term. Grouting the process rooms and sealing all building openings will remove exposure pathways for future hypothetical industrial workers to contaminants within Building 235-F. Installation of a sloped concrete roof will prevent rainwater infiltration and, combined with entombing the contaminants within Building 235-F in grout, will prevent groundwater migration in exceedance of maximum contaminant levels.

TABLE OF CONTENTS

TABLE OF CONTENTS *(Continued)*

LIST OF APPENDICES

LIST OF FIGURES

LIST OF FIGURES *(Continued)*

LIST OF TABLES

LIST OF TABLES *(Continued)*

LIST OF ABBREVIATIONS AND ACRONYMS

~	approximate, approximately
ABL	Actinide Billet Line
ACM	asbestos transite-containing material
ags	above ground surface
ARAR	applicable or relevant and appropriate requirement
BPRG	building preliminary remediation goal
BDCC	building dose compliance concentration
CERCLA	Comprehensive Environmental Response Compensation and Liability Act
CFR	Code of Federal Regulations
cm	centimeter
DNFSB	Defense Nuclear Facilities Safety Board
EC&ACP	Environmental Compliance/Area Completion Projects
F&T	Fate and Transport
FAMS	F-Area Material Storage
FAOU	F-Area Operable Unit
FFA	Federal Facility Agreement
ft	foot, feet
ft^2	square feet
ft^3	cubic feet
g	grams
GSA	General Separations Area
HELP	Hydrological Evaluation of Landfill Performance
HHRA	human health risk assessment
in	inch
ISD	In-situ Decommissioing
kg	kilogram
km	kilometer
km^2	square kilometer
lbs	pounds
LLC	Limited Liability Company
LLW	Low-Level Waste
LLWF	Low Level Waste Facility
LUCAP	Land Use Control Assurance Plan
m	meter
m^2	square meter
MAR	Material at Risk
MCL	maximum contaminant level
mi	mile
mi^2	square mile
msl	mean sea level
NCP	National Oil and Hazardous Substances Pollution Contingency Plan
NEPA	National Environmental Policy Act
NTC	Non-Time Critical

LIST OF ABBREVIATIONS AND ACRONYMS *(Continued/End)*

OML	Old Metallurgical Lab
ORR	Operational Readiness Review
PCB	polychlorinated biphenyl
PEF	Plutonium Experimental Facility
pCi, pCi/g, pCi/L	picocuries, picocuries per gram, picocuries per liter
PPE	Personal Protective Equipment
POA	point of assessment
PTSM	principal threat source material
PuFF	Plutonium Fuel Form
RAO	removal action objectives
RBC	Reactor Building Complex
RCRA	Resource Conservation and Recovery Act
rem	roentgen equivalent man
RFI	RCRA Facility Investigation
RI	Remedial Investigation
ROD	Record of Decision
ROI	region of interest
ROM	Rough Order of Magnitude
RME	reasonable maximum exposure
RSER/EE/CA	Removal Site Evaluation Report/Engineering Evaluation/Cost Analysis
RSL	regional screening level
RTG	Radioisotope Thermoelectric Generator
S&M	Surveillance and Maintenance
SCDHEC	South Carolina Department of Health and Environmental Control
SEMS	Superfund Enterprise Management System
SPFF	Special Products Fabrication Facility
SRNL	Savannah River National Laboratory
SRNS	Savannah River Nuclear Solutions
SRS	Savannah River Site
TBC	to be considered
TRU	Transuranic
TSCA	Toxic Substances Control Act of 1976
USDOE	United States Department of Energy
USEPA	United States Environmental Protection Agency
USFS	United States Forest Service
UTR	Upper Three Runs
UTRA	Upper Three Runs Aquifer
WSRC	Westinghouse Savannah River Company

1.0 INTRODUCTION

The U.S. Department of Energy (USDOE) is proposing to perform a non-time critical (NTC) removal action at the F-Area Material Storage Building (FAMS)(235-F), herein referred to as Building 235-F. Building 235-F is listed in the Savannah River Site (SRS) Federal Facility Agreement (FFA) on Appendix K.1: *D&D Facilities to be Decommissioned* (FFA 1993). The USDOE, United States Environmental Protection Agency (USEPA), and South Carolina Department of Health and Environmental Control (SCDHEC) reached agreement in 2019 that a NTC removal action under Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) is an appropriate regulatory mechanism to decommission Building 235-F (USDOE 2019a).

The USDOE, USEPA, and SCDHEC held scoping meetings in October 2019 and April 2021 and reached agreement on the criteria for evaluation of human health risk and contaminant migration for Building 235-F, the NTC removal alternatives for evaluation, and the comparative analysis of the removal alternatives (SRNS 2021a). This Removal Site Evaluation Report/Engineering Evaluation/Cost Analysis (RSER/EE/CA) identifies the objectives of the NTC removal action for Building 235-F, describes removal action alternatives that address the potential threats from release of contaminants to the environment, and provides a vehicle for public comment in accordance with the National Oil and Hazardous Substances Pollution Contingency Plan (NCP), 40 Code of Federal Regulations (CFR) § 300.415. The NTC removal action will also address the Metallurgical Building Stack (293-F), also referred to as the abandoned capped stack (293-F), located on the east side of the building and an underground storage tank connected by a pipe trench on the north side of the building.

The SRS encompasses 803 square kilometers (km^2)(310 square miles [mi^2]) of land adjacent to the Savannah River, principally in Aiken and Barnwell counties of South Carolina. SRS is located approximately 40 km (25 mi) southeast of Augusta, Georgia, and 32 km (20 mi) south of Aiken, SC (Figure 1). SRS is owned by USDOE while Savannah River Nuclear Solutions, LLC (SRNS) provides management and operating services. SRS has historically produced tritium, plutonium, and other special nuclear materials for national defense. Chemical and radioactive wastes are by-products of nuclear material production processes. Hazardous substances, as defined by CERCLA, are present in the SRS environment.

The public is encouraged to comment on the removal action alternatives presented in this RSER/EE/CA. Following the public comment period, an Action Memorandum will be prepared by USDOE and added to the SRS Administrative Record, which is accessible by the public. All responses to the public comments will be included in an Action Memorandum.

Copies of this RSER/EE/CA and the Administrative Record for SRS are available at the following locations:

U.S. Department of Energy
Public Reading Room
Gregg-Graniteville Library
University of South Carolina-Aiken
471 University Parkway
Aiken, South Carolina 29803
(803) 641–3504

Thomas Cooper Library
Government Information and Maps Department
University of South Carolina
1322 Greene Street
Columbia, South Carolina 29208
(803) 777–4841

Hard copies of this RSER/EE/CA are available at the following locations:

Reese Library
Government Information Department
Augusta University
2500 Walton Way
Augusta, Georgia 30904
(706) 737–1744

Asa H. Gordon Library
Savannah State University
2200 Tompkins Road
Savannah, Georgia 31404
(912) 358–4324

To submit comments or request a public meeting during the public comment period, contact:

Angie Benfield
Savannah River Nuclear Solutions, LLC
Public Involvement
Savannah River Site
Building 730–1B
Aiken, South Carolina 29808
(803) 952-9830
angela.benfield@srs.gov

2.0 SITE CHARACTERIZATION

2.1 Site Description and Background

Building 235-F is a windowless, two-story, reinforced-concrete structure located in F Area near the center of SRS (Figure 2). Building 235-F is designated a Hazard Category 2 nonreactor nuclear facility according to DOE STD-1027-92. Other facilities in the vicinity of Building 235-F include nuclear facilities and standard industrial buildings. F-Canyon (Building 221-F) is located approximately 270 meters (m) (886 feet [ft]) west of Building 235-F, and the F-Area Tank Farm (241-F) is located approximately 500 m (1,640 ft) southwest of Building 235-F. The former Mixed Oxide Fuel Facility Project administrative building is located approximately 200 m (656 ft) north of Building 235-F (Figure 3).

Building 235-F is approximately 68 m (222 ft) long, 33 m (109 ft) wide, and 8.5 m (28 ft) high. Excluding ancillary structures, the building has a footprint of approximately 2,230 square meters (m^2) (24,000 square ft [ft^2]). A historical photo (date unknown) of the building is shown in Figure 4. The two-story structure has double-reinforced 36-centimeter (cm) (14-inch [in]) thick exterior walls supported by a 1.5-m (5-ft) wide perimeter grade beam. The first building level consists of a 20-cm (8-in) reinforced concrete slab on grade. Pier footings and columns support the second building level. The roof is supported by reinforced concrete beam and girder systems and includes a 23-cm (9-in) high perimeter curb or parapet. Drainage off the roofs is directed through roof drains. Some interior walls are reinforced concrete load bearing walls. The Metallurgical Building Stack (293-F), also referred to as the "abandoned capped stack," is located on the east side of the building, and an underground storage tank is connected by a pipe trench on the north side of the building.

Within the Building 235-F perimeter fence, there are three buildings in addition to Building 235-F which have the potential for residual radiological contamination, primarily in the form of plutonium-238 (Pu-238) and neptunium-237 (Np-237). The three buildings include the Sand Filter Fan House (292-2F), Sand Filter (294-2F), and the Exhaust Stack (291-2F) (Figure 5). These three buildings are not part of the NTC removal action but are important because they support the ventilation systems that will remain operational during decommissioning of Building 235-F. The below-grade Sand Filter (294-2F), primarily of concrete construction, receives exhaust air from

Building 235-F and is linked to the building through a 213-cm (84-inch) diameter underground duct/tunnel constructed of 1.8 m (6 ft) and 2.4 m (8 ft) segments of reinforced concrete pipe. The underground duct/tunnel is considered part of the Sand Filter (294-2F). The Sand Filter Fan House (292-2F) is a separate structure that contains a fan room, diesel generator room, and electrical control room which serves the exhaust fans and other building loads. Currently, the exhaust fans draw air from radiologically contaminated process areas/enclosures within Building 235-F through double HEPA filtration to the below-grade Sand Filter (294-2F) before discharge to the metal constructed Exhaust Stack (291-2F). The Sand Filter Fan House (292-2F), Sand Filter (294-2F), and the Exhaust Stack (291-2F) have not been assayed. However, the Sand Filter (294-2F) has been surveyed per the SRS Radiological Controls Program. Based on historical operations, the radiological inventories in all three buildings are assumed to be significantly less than the estimate of radiological holdup (i.e., residual radiological contamination) within Building 235-F. The Sand Filter Fan House (292-2F) and Sand Filter (294-2F) have been dispositioned to FFA Appendix K.1: *D&D Facilities to be Decommissioned.* The Exhaust Stack (291-2F) is identified as an ancillary structure and will be evaluated with the Sand Filter Fan House (292-2F) and Sand Filter (294-2F). ~~Other b~~Buildings located within the Building 235-F perimeter fence that are designated for eventual deactivation and decommissioning in accordance with FFA Appendix K.1: *D&D Facilities to be Decommissioned* are shown in Figure 5.

Building 235-F Process History

Building 235-F was constructed in the 1950s as part of the original SRS project and used for a variety of missions. The original mission slated for Building 235-F was “C-Line” designed to manufacture nuclear triggers from Pu-239 metal. The mission was cancelled before any equipment was installed, and the building unused until the mid-1960s. During the mid- to late 1960s, Building 235-F was used to house the Special Products Fabrication Facility (SPFF) that supported facilities processing uranium-238 (U-238), Np-237, and Pu-239 oxide. These facilities were known as the “Slug Facility” and the “Alloy Line.” The Slug Facility extended almost the entire length of the building (east to west), occupying space later taken over by the Actinide Billet Line (ABL) and the Plutonium Fuel Form (PuFF) Facility. The Slug Facility processed uranium, plutonium, and other actinide-bearing materials into irradiation target components. The Alloy Line was a set of gloveboxes running north and south on the first level of the building at the east end and had blending, welding, machining, and decontamination capabilities. The Alloy Line included an induction furnace that was used to cast uranium alloy pieces for targets/slugs. The Alloy Line cabinets were removed in 1984, and the New Metallography Lab was built in its place in 1986. The New Metallography Lab was never placed into operation.

In the mid-1970s, the SPFF was decontaminated and decommissioned, and the building reconfigured to support the ABL mission and make room for the PuFF Facility (Figures 6 and 7). As part of these modifications, the Slug Facility cabinets were truncated to the current ABL configuration. The ABL consisted of a glovebox line and associated equipment, and initially fabricated plutonium billets for special reactor applications. ABL was later modified to produce special billets from Np-237 oxide powder for extrusion into reactor targets. The ABL continued operations until 1991.

The next mission of Building 235-F was the PuFF Facility and Plutonium Experimental Facility (PEF) including a Metallography Laboratory (commonly known as the Old Metallurgical Lab [OML]) (Figures 6 through 8). These facilities produced heat sources from Pu-238 oxide powder for the National Aeronautics and Space Administration program. The radioactive pellets (i.e., encapsulated Pu-238 heat source) were shipped to the USDOE Mound Facility for final assembly into a system referred to as a Radioisotope Thermoelectric Generator (RTG). The RTG acts as a power source to convert heat from the radioactive decay of the Pu-238 pellets into electricity.

The PuFF Facility was installed on the first building level and consisted of Process Cells 1-5 (east line) and their attached gloveboxes, Process Cells 6-9 (west line), and the auxiliary systems located on the second building level (Figures 7 through 8). The PuFF Facility produced Pu-238 heat source pellets from 1977 until 1983. The PuFF Facility cells and gloveboxes contain significant amounts of residual Pu-238 oxide contamination. The cells and associated gloveboxes are maintained at negative pressure by the process exhaust ventilation system for confinement of this residual material.

The PEF contains a line of 12 gloveboxes and 2 hoods (Figure 7). PEF activities included research and development on processes that manufactured Pu-238 heat sources. Pu-238 was processed in the PEF from 1979 to 1981 to develop the fuel pellet fabrication process performed in the PuFF

Facility. The Pu-238 inventory was removed when the facility was shut down, but residual Pu-238 contamination, in dust form, remains throughout the glovebox line. The glovebox line is maintained at negative pressure by the process exhaust ventilation system for confinement of the residual plutonium oxide. The layout of the first building level is shown in Figure 7.

The OML facility, located on the second building level, contains a glovebox line that was used to test Pu-238 heat source pellets for conformance to design specifications (Figure 8). OML was used to examine the iridium welds and was also used to cut iridium from failed welding runs so the Pu-238 oxide could be recycled back to the SRS HB-Line. The glovebox line is maintained at negative pressure by the process exhaust ventilation system for confinement of the residual material. The layout of the second level is shown in Figure 8 and depicts areas where the ductwork supports the ABL, PuFF Facility, and PEF.

All metallurgical processes within Building 235-F (including PEF, PuFF Facility, OML, and ABL) were shut down by 1991. The last mission of the Building 235-F provided for the receipt, storage (within vaults), and repackaging of containerized plutonium- and uranium-bearing materials (i.e., specialized nuclear materials) to support both the SRS and the USDOE complex. The original vault areas and other rooms within Building 235-F were equipped to provide storage space for nuclear materials as well as designated material repacking areas. The storage and repackaging mission ended in 2006 and all special nuclear materials, except for radiological holdup, were removed from the building (i.e., the vaults were de-inventoried, and all stored plutonium and shipping containers were removed). During deactivation, Building 235-F will be prepared for Long Term Safe Storage, which involves minimal surveillance and maintenance (S&M) for containing and monitoring the residual radiological contamination within the process areas.

The original 22.9 m (75 ft) high stack (293-F) was part of the 1950s ventilation system and is located approximately 9 m (30 ft) east of Building 235-F and connected by a pipe trench (Figure 6). The stack was abandoned in place in the 1970s after a change in facility mission. It was recognized that the abandoned capped stack was vulnerable to natural phenomena hazard events which could cause it to fall on the ventilation exhaust duct on the Building 235-F roof. In 2010, the stack height was reduced to 8.6 m (28 ft), and an aluminum cap was installed to prevent rainwater infiltration and ecological habitat (Figure 9). The height reduction was accomplished by chipping away the stack walls and dropping the debris inside the remaining stack. The interior volume of the remaining stack, 19 m^3 (680 ft^3), is filled with approximately 10 m^3 (357 ft^3) of debris. Void space resulting from possible large pieces of debris is unevenly distributed and this condition was taken into consideration in the structural evaluation of the reduced height stack. The 8.6 m (28 ft) debris filled stack was found to be structurally adequate to resist design basis tornado and seismic loads.

The underground storage tank is a 560-gallon (nominal) tank housed in a 2.13 m (7 ft) by 2.13 m (7 ft) by 3.96 m (13 ft) deep concrete pit. A 0.457-m (1.5-ft) wide underground pipe trench connects the concrete pit to Building 235-F, and a single 5 cm (2 in) diameter pipe connects the underground storage tank to the building. The underground tank initially supported the ABL by receiving liquid waste solutions in the 1960s and 1970s. The underground storage tank was repurposed in the late 1970s to receive liquids from decontamination activities in the PuFF Facility cells and condensate from radiological monitors.

2.2 Previous Action

In 2006, Building 235-F was de-inventoried of all special nuclear material with the exception of legacy radiological holdup. Extensive deactivation activities to remove as much Material At Risk (MAR) as possible from each of the PuFF Facility process cells and gloveboxes associated with the PuFF Facility Cells 1 through 9 were completed in June 2019. MAR in PuFF Facility Cells 1 and 2, which included the glovebox with the highest concentration of radiological holdup, was removed to the greatest extent practical. As removal activities progressed, it became apparent that further MAR removal did not justify the significant risk of exposure to deactivation workers. On July 10, 2019, the USDOE instructed SRNS to stop removal of MAR from Building 235-F and proceed with the activities necessary to establish a deactivation end state for a “Cold and Dark Transition Surveillance and Maintenance Facility” (USDOE 2019b).

No additional MAR removal from any of the enclosures is planned, and all remaining activities associated with final deactivation and S&M are non-intrusive to the process areas (i.e., no internal penetration of the exhaust ventilation confinement boundary of a cell, glovebox, hood, cabinet, or enclosure). Building 235-F is designated a Hazard Category 2 nonreactor nuclear facility according to DOE STD-1027-92 (i.e., residual radiological holdup exceeds the 3.6 curies [Ci] threshold for Pu-238). Building 235-F is expected to be downgraded to less than a Hazard Category 3 radiological facility once a documented safety analysis confirms that the radiological inventory is unreleasable following decommissioning.

Prior to the height reduction of the abandoned capped stack (293-F), cores from the stack were collected at approximately 13.7 m (45 ft) and were sampled for radioactivity and polychlorinated biphenyl (PCB) contamination. The radioactivity of the waste material was less than measured between 200-400 picocuries/gram (pCi/g) and was characterized as low-level waste (LLW) for disposal in the SRS E-Area Low-Level Waste Facility (LLWF). No PCB contamination was not detected.

In 1990, the underground storage tank connected by a pipe trench on the north side of the building was identified during a SRS Hazardous Waste Survey as potentially containing hazardous waste. The bottom of the underground tank was sampled in December 1990 as part of a Settlement Agreement with SCDHEC (SCDHEC 1990). Sampling confirmed the presence of radionuclides (plutonium and tritium) and hazardous waste constituents (cadmium and chromium). In May 1991, the tank was emptied and cleaned of liquid and sludge and the inlet pipe capped. The liquid was sent to the SRS H-Area Tank Farm, and the sludge collected in drums and sent to the SRS Mixed Waste Storage Facility. The tank walls and bottom were scraped and mopped, and the tank flushed with approximately 1,893 liters (L) (500 gallons [gal]) of water and caustic. The tank was inspected to verify all material was removed in accordance with the SCDHEC Settlement Agreement.

Building 235-F and support facilities are in a reduced S&M state, and deactivation activities to support decommissioning are under way. Residual radiological contamination remains in the PuFF Facility process area (Pu-238), PEF (Pu-238 oxide), OML (Pu-238 oxide), and ABL (Pu-238 oxide and Np-237 oxide). S&M activities include monitoring of residual radiological contamination and support for the electrical system, diesel generator, ventilation, instrument air, alarm monitoring, steam supply, chilled water systems, and confinements. The Building 235-F structure, process enclosures (cells and gloveboxes), and exhaust ventilation currently function to provide confinement of the residual radiological holdup. Rooms within the building are maintained at a slight vacuum as compared to the atmosphere outside the building. Also, the air atmospheres within the process enclosures (cells and gloveboxes) are maintained at a slight vacuum as compared to the rooms. The current deactivation scope includes de-energizing supply and exhaust fans within the building, but not the exhaust fans external to the building. During and after deactivation, these external fans will maintain vacuum inside Building 235-F and inside the process enclosures. A ventilation strategy will be developed during the design of the selected NTC removal action to determine the appropriate ventilation to prevent contaminant release to the environment and to protect workers. Building 235-F is not occupied and is monitored remotely.

2.3 Land Use

Building 235-F is located within the F Area fence line in an area currently designated for industrial use. Future land use in F Area is expected to remain industrial and will be controlled in accordance with the SRS Land Use Control Assurance Plan (LUCAP) (WSRC 1999). An industrial land use scenario was selected as the baseline risk assessment exposure scenario for Building 235-F for the protection of human health and the environment (USDOE 1996).

Groundwater is not part of the NTC removal action scope for Building 235-F. There is no current or projected future use of the groundwater as a drinking water source. The groundwater in F Area is currently monitored by the SRS General Separations Area Western Groundwater Operable Unit monitoring network.

2.4 Environmental Setting

The surface elevation across Building 235-F is approximately 93.4 m (306 ft) above mean sea level (msl). The Upper Three Runs Aquifer (UTRA) is the shallow-most aquifer beneath Building 235-F. The water table elevation is approximately 67.4 m (221 ft) above msl (i.e., 26 m [85 ft] below ground surface), and groundwater flow is north/northeast towards the UTR creek and its tributaries.

2.5 Nature and Extent of Contamination

Extensive characterization of Building 235-F has been performed to identify hazardous and radiological contamination that is expected to remain in the building following deactivation activities. The radiological inventory is based on the most recent material removal campaign in 2019 and is supplemented with data following the shutdown of the final mission of the building in 2006 (SRNS 2020). Characterization methods included non-destructive analysis of radiological holdup, smear data for radiological contamination, dose measurements, hazardous material sampling results, and engineering/process knowledge. Hazardous material inventories are predominately from reports following the shutdown of the final mission of the building in 2006. The detailed characterization activities and radiological and hazardous material inventories for Building 235-F were assembled into a characterization report in 2020 in support of decommissioning activities (SRNS 2020). The radiological inventory identified that 85 percent (%) of the Pu-238 contamination remaining in the building is located in the PuFF Facility cells and 87% of Np-237 is in ABL. The complete radiological inventory for the building is provided in Table 1. The location of radiological holdup on the first and second building levels is depicted in Figures 10 and 11, respectively.

Building 235-F was thoroughly inspected for asbestos in 2006 using facility knowledge, visual inspection completed by certified asbestos inspectors, document review, and the collection of bulk samples. This inspection identified 9,643 m^2 (103,791 ft^2) of asbestos transite-containing material (ACM) in the building. There is no intent to remove ACM during deactivation unless it is determined to be friable or is disturbed by deactivation activities.

Lead is present in the building in PuFF Facility gloveboxes, leaded glass on glovebox windows, lead washers, brass valves, lead-acid batteries such as in emergency lights, lead shielding, lead solder in sewer line joints, and lead counterweights. The accessible lead counterweights and lead-acid batteries will be removed during deactivation. The largest contributor of lead is shielding in the walls of gloveboxes and shield doors of the process areas, which will not be removed during deactivation. An estimated total of 17,208 kilograms (kg) (37,937 pounds [lbs]) of lead remains in processing areas following deactivation. In addition, lead-based paint is likely present throughout both the process areas and non-process areas. The maximum detected concentration of lead in lead-based paint was 0.940 g/kg (940 mg/kg) (SRNS 2020). Paint that remains intact on the wall will be abandoned-in-place, while any lead disturbed by deactivation activities will be removed. Paint removal during deactivation will not result in a significant change in the overall lead inventory remaining in the building.

Other hazardous materials in the building may include the following:

- Mercury in lamps, switches, batteries, thermostats, etc. These features will be removed if accessible during deactivation.

- ~~Used~~ Residual oils in locations throughout Building 235-F. During inventory assessments it has been determined that all transformers are ~~dry and void of dielectric fluid~~of the dry type and did not contain oil reservoirs or PCBs. Hydraulic oil in the main reservoirs for the hot and cold presses have been drained ~~aside from~~ and any oil remaining in the small lines to the presses and the hydraulic cylinders will be drained during deactivation. The hydraulic oils have no history of PCBs. The accessible used oil will be drained from the chillers (Room 209) and air compressors (Room 152) during deactivation.

- Chiller/cooling water contain chemicals that inhibit biological activity (i.e. sodium nitrate), prevent freezing (i.e. ethylene glycol) and/or inhibit corrosion within the piping and equipment. The cooling water will be drained and recovered as wastewater.

- PCBs may be present in the materials of construction such as paint, joint compounds, insulation, ~~capacitors, hydraulic oil,~~ and light ballasts. ~~Accessible PCB ballasts will be removed during deactivation.~~ All ballasts, regardless of the type (non-PCB or PCB-containing) will be removed during deactivation and no known ballasts will remain within the facility. PCBs in paint on the first and second levels, assuming two surface coats of paint applied, was conservatively estimated at 2.38 kg (7.44 lbs). Paint samples were taken with the highest concentration of PCBs of 3,900 mg/kg (SRNS 2020). PCBs in building components, paint, etc., will remain in place and be addressed by the NTC removal action alternative.

- Other hazards include refrigerant that will be removed from the heating, ventilation, and air conditioning units and chiller units during deactivation. Lead and chromium may be present in items throughout the facility such as circuit boards and smoke detectors that will not be removed during deactivation. Photographic equipment in a former dark room has been removed, but a sewer connection remains. ~~Residual silver or other photographic chemicals may be present in the sewer line or associated trap, which will be sampled during~~

deactivation. The drain and trap in the former dark room were sampled in August 2021. Residual silver was present in the drain and trap with a maximum concentration of 180 mg/kg, well below the industrial worker USEPA Regional Screening Level of 5,800 mg/kg. Building 235-F will be isolated from the sanitary sewer system during deactivation by sealing all floor connections and grouting the manhole nearest to the building.

- Historical storage and handling of beryllium components were conducted on the first level. Beryllium-containing items were removed from containers for visual inspection and then placed back into storage. No cutting, grinding, or other activities were performed on the components that would have generated beryllium dust. All beryllium-containing items have been removed from Building 235-F.

2.6 Human Health Risk Assessment

A streamlined human health risk assessment (HHRA) was conducted on the inventory presented in Table 1 for Pu-238 and Np-237 and is presented in Appendix A. It is recognized that other contaminants are present within the facility due to process impurities and radioactive decay (i.e., daughter products). In addition, hazardous materials such as asbestos, lead, and PCBs will remain in the building following deactivation as discussed in Section 2.5. Because any risk from exposure to the radioactive decay products and hazardous materials are well bounded by the risk from exposure to Pu-238 and Np-237 in the process areas, the streamlined risk evaluation focuses on Pu-238 and Np-237 as the primary risk drivers.

The HHRA was performed on PuFF Facility Process Cells 1-5 where the majority of Pu-238 (85%) contamination is located, and the ABL since it contains the majority of Np-237 (87%) contamination. The HHRA was also performed for an Entire Building 235-F scenario which includes the Pu-238 and Np-237 inventory from PuFF Facility Process Cells 1-5, PuFF Facility Process Cells 6-9, ABL, PEF, OML, and various exhaust and ductwork components. The contribution of the other process areas and building components (excluding PuFF Facility Process Cells 1-5 and ABL) was 14% of Pu-238 and 13% of Np-237.

The default indoor worker receptor scenario, herein referred to as the hypothetical future industrial worker, is a standard USEPA scenario used to estimate long-term risks to workers who are exposed to building contaminants within an industrial setting. The exposure assumptions for this default scenario are 25 years, 250 days per year, and 8 hours per day. The hypothetical future industrial worker is exposed to contamination in the building via two exposure routes, i.e., inhalation and external exposure (submersion). The HHRA conservatively assumes that no personal protective equipment is used, and that all of the contamination remains within the building. Source material at SRS is considered principal threat source material (PTSM) when the cumulative risk exceeds 1E-03 for carcinogens or a hazard index greater than 10 for noncarcinogens. As shown in Appendix A, the risk to the hypothetical future industrial worker (i.e., Entire Building Risk = 2.3E+09, PuFF Facility Cells 1-5 risk = 5.9E+10, ABL risk = 6.9E+08) is significantly higher than the USEPA acceptable risk range of 1E-04 to 1E-06, and the contamination is considered PTSM (>1E-03).

Although a viable route of human exposure to hazardous materials present in building components is unlikely, exposure to PCBs and lead in paint is a recognized health hazard. A risk evaluation of PCBs and lead, based on maximum detected concentrations in building paint, is provided in Appendix A to demonstrate the negligible risk contribution from these hazardous inventories when compared to the primary risk drivers. There is no intent to remove ACM that is present in the building during deactivation unless it is determined to be friable or is disturbed by deactivation activities. The USEPA does not provide a screening level for asbestos for the purpose of estimating risk. No ecological constituents of concern are associated with Building 235-F because the industrial setting does not provide suitable habitat (i.e., incomplete exposure pathway) for ecological receptors, and the building interior is not accessible for ecological habitat.

2.7 Contaminant Migration Risk to Groundwater

The potential for migration of the radionuclide inventory in Table 1, the elemental lead inventory, and the PCB inventory through the vadose zone to groundwater was performed using a one-dimensional (1-D) GoldSim fate and transport (F&T) model and is presented in Appendix B. The inventories and F&T modeling results are representative of the deactivated end state for Building 235-F. The F&T model assumes decommissioning in 2025; however, sensitivity analysis shows that the actual year of decommissioning (up to 2040) does not impact the modeling results.

For the F&T modeling, the estimated total elemental lead inventory of 17,208 kg (37,937 lbs) was conservatively placed in PuFF Facility Cell 3. PCBs were assumed to be contained within the surface paint throughout Building 235-F. For modeling, the estimated total PCB inventory of 2.38 kg (7.44 lbs) in paint was assumed to be equally distributed in PuFF Facility Cells 1-5.

The 1-D GoldSim F&T model was used to evaluate four scenarios: a No Action scenario, ~~to represent the Building 235-F deactivated state~~Grout First Building Level Scenario, Grout First and Second Level Scenario, and Grout First and Second Levels with Engineered Roof Scenario. As discussed in further detail in Appendix B, the second and third modeling scenarios, which do not include an engineered roof, were not considered in the NTC removal action evaluation. The ~~model also evaluated a scenario to grout the first and second level process areas to~~ No Action Scenario represents the Building 235-F deactivated state, and the Grout First and Second Levels with Engineered Roof Scenario supports the in-situ decommissioning (ISD) removal action alternatives. The following conditions were assumed for ~~each~~ these two modeling scenarios.

- No Action Scenario - Represents Building 235-F deactivated state (i.e., baseline conditions). The F&T model assumes the existing roof collapses at 150 years with no modifications to prevent ponding of rainwater. This modeling scenario supports the No Action removal action alternative presented in Section 4.0.
- Grout First and Second Levels with Engineered Roof Scenario - First and second building level process areas are grouted. An engineered roof is placed over Building 235-F. The engineered roof is assumed to last over 1,000 years. This modeling scenario supports the two ISD removal action alternatives presented in Section 4.0.

The F&T evaluation simulated the migration of contaminants from the building through the vadose zone, the UTRA, and an unnamed tributary of the UTR creek at five different points of assessment (POA) as follows: 1-m (3-ft) beyond the outside perimeter of Building 235-F; 100-m (328-ft) beyond the outside perimeter of Building 235-F; 360-m (1,181-ft) (F Area industrial area boundary fence where it crosses along projected plume path); 683-m (2,241-ft) (closest groundwater-fed seepline to Building 235-F along the plume path); and the surface water in an unnamed UTR tributary. The POAs and groundwater flow paths are depicted in Appendix B, Figures B-5 and B-6. The assessment periods evaluated in the model include 0-1,000 years (0-1K years), 0-10,000 years (0-10K years), and 0-100,000 years (0-100K years). The results of the F&T analysis are provided in Appendix B, Tables B-1 and B-3.

The USDOE, USEPA, and SCDHEC agreed to consider the baseline condition for the identification of contaminant migration problems warranting action based on an assessment period of 0-10K years at the 360-m (1,181-ft) POA. The POA at the 360-m (1,181-ft) is expected to be consistent with the POA for all F Area sources in support of the eventual F-Area Operable Unit (FAOU) Record of Decision (ROD), while an assessment period of 0-10K allows for the ingrowth of radioactive decay products and travel time for the longer-lived radionuclides.

Based on the radionuclide inventories in Table 1, the following parent radionuclides and their full-chain progeny were considered:

Np-237 – key parent nuclide in the ABL $^{237}Np \rightarrow {}^{233}U \rightarrow {}^{229}Th$

Pu-238 – key parent nuclide in the PuFF Facility $^{238}Pu \rightarrow {}^{234}U \rightarrow {}^{230}Th \rightarrow {}^{226}Ra \rightarrow {}^{210}Pb$

The dashed line in each abbreviated decay chain indicates the radionuclide that contributes the most dose and concentration in the ABL and PuFF Facility. Pu-238 and Np-237 are used to represent the modeling results from both the parent radionuclides and their progeny. Each parent was modeled as a separate plume.

For the No Action scenario at the 360-m (1,181-ft) POA and 0-10K years assessment period (i.e., baseline deactivated building state), Np-237 exceeds the maximum contaminant level (MCL) threshold for gross alpha. Beta-gamma, radium, uranium, elemental lead, and PCBs did not exceed their MCLs at the 360-m (1,181-ft) POA and 0-10K years assessment period under the No Action scenario.

Under the modeling scenario to grout the first and second level process areas and place an engineered roof, there were no exceedances of Np-237, Pu-238, radium, uranium, elemental lead, or PCBs at the 360-m (1,181-ft) POA and 0-10K years assessment period. As discussed in Appendix B, differences between grouting the entire building (including the process areas) or grouting only the process areas would be undiscernible in the model results due to the relative percentage of contaminants in the process areas and the 1-D nature of GoldSim. The results of this modeling scenario therefore support the ISD removal action alternatives evaluated in this RSER/EE/CA that include grouting only the process areas within the building or grouting the entire building.

3.0 REMOVAL ACTION SCOPE AND OBJECTIVES

3.1 Justification for the Proposed Removal Action

USDOE, the lead agency, is mandated to take action to reduce the adverse effects of man-made contamination on human health and the environment. The NCP states that if the lead agency determines a release or potential release poses a threat to public health or welfare or the environment, the lead agency may take any appropriate removal action to abate, prevent, minimize, stabilize, mitigate, or eliminate the release or threat of release. This determination should be based on the factors identified in 40 CFR § 300.415(b) (2).

Potential exposure to residual radiological contamination that will remain in Building 235-F following deactivation exceeds the hypothetical future industrial worker risk threshold (risk greater than 1E-06) and PTSM levels (risk greater than 1E-03). In addition, residual radiological contamination has the potential to leach to groundwater at levels that would exceed MCLs at the POA (F Area industrial area boundary fence) in less than 10,000 years. The conceptual site model for Building 235-F is shown in Figure 12 and demonstrates that a removal action is needed to break the human health exposure and contaminant migration pathways.

On March 6, 2019, the USDOE presented an acceleration strategy for decommissioning Building 235-F to the USEPA and SCDHEC. The three agencies agreed that a NTC removal action under CERCLA is an appropriate regulatory mechanism to support decommissioning of Building 235-F. On May 14, 2019, the SRS Citizens Advisory Board published Recommendation #364 to accelerate the timeline for decommissioning Building 235-F and preparation of the required regulatory documents in parallel with the deactivation process. A follow-up information meeting was held with the USDOE, USEPA, and SCDHEC on October 30, 2019, and agreement was reached to proceed with the development of a RSER/EE/CA for Building 235-F (USDOE 2019a). A NTC removal action will provide the most appropriate level of analysis, oversight, public participation, and flexibility to conduct decommissioning in a cost-effect manner that achieves risk reduction and protects human health and the environment without unnecessary delay.

3.2 Removal Action Objectives

The removal action objectives (RAOs) for Building 235-F to protect human health and the environment include the following:

- Prevent exposure of the hypothetical future industrial worker to radiological contaminants present in Building 235-F that exceed 1E-06 risk thresholds (including PTSM).
- Prevent the migration of radionuclide contamination from Building 235-F to groundwater at concentrations that exceed MCLs ~~at the 360-m (1,181-ft) POA in less than 10,000 years~~ to the extent practicable.

4.0 IDENTIFICATION OF REMOVAL ACTION ALTERNATIVES

In accordance with CERCLA, the following alternatives for the Building 235-F removal action were examined:

- **Alternative A-1, No Action**: Building 235-F, underground storage tank, and abandoned capped stack (293-F) will remain as currently exist after completion of the deactivation scope. The roof is assumed to collapse at 150 years at which time additional response action(s) will likely be necessary to mitigate exposure/spread of contaminants. No Action is representative of the Building 235-F deactivated state.
- **Alternative A-2, ISD of First and Second Level Process Areas/Engineered Roof**: First and second level process areas will be grouted. An engineered roof (sloped concrete reinforced roof slab with integral crystalline waterproofing) designed to last 1,000 years will be installed. The underground storage tank will be grouted/capped, and the abandoned capped stack (293-F) will be permanently sealed.
- **Alternative A-3, ISD of Entire Building/Engineered Roof**: First and second levels (all areas) will be grouted. An engineered roof (sloped concrete reinforced roof slab with integral

crystalline waterproofing) designed to last 1,000 years will be installed. The underground storage tank will be grouted/capped, and the abandoned capped stack (293-F) will be permanently sealed.

- **Alternative A-4, Complete Building 235-F Removal/Soil Cover**: Building 235-F will be demolished to the building slab. The abandoned capped stack (293-F) will be removed. The underground storage tank will be grouted/capped. A soil cover will be applied over both the building slab and grouted underground storage tank.

4.1 Alternative A-1, No-Action, Activity Description

Under Alternative A-1, Building 235-F will remain in the deactivation end state and will require continued S&M. Under deactivation, most of the residual radiological contamination will be contained within the process enclosures which will be sealed to prevent contaminant migration. The E5 fan exhaust ventilation system, located external to Building 235-F in Building 292-2F (see Figure 5), will remain in operation and will draw unconditioned air into the building and will remain connected to the process enclosures to provide a negative pressure. To minimize failure of the gypsum walls and corrosion of steel components due to condensation build up, measures will be taken to reduce the air flow to the enclosure rooms and adjacent corridors. To prevent the potential spread of contamination, the process enclosure seals will be evaluated and enhanced where necessary, contamination in the process rooms will be fixed via a coating or removed to the extent practical, and the north door will be resealed. Building 235-F will be isolated from all other utilities (e.g., domestic water, electricity, steam, chilled water, sanitary sewer, telecommunications, etc.). Non-radiological material will be removed to the extent practicable. Hazardous material, such as lead shielding, leaded glass, lead based paints, PCB-based paints, and non-friable asbestos will remain in the building. Most ceiling tiles and other transient combustibles will be removed to reduce the combustible loading of Building 235-F. The Sand Filter (294-2F) which filters air from the E5 fan exhaust ventilation system, the Building 292-2F Sand Filter Fan House (292-2F), and the 291-2F Exhaust Stack (291-2F) will remain in operation. The abandoned capped stack (293-F) will remain in the current configuration.

The deactivated state for Building 235-F represents the No Action scenario. Ongoing S&M activities are included in the No Action scenario but will be limited to infrequent entry to monitor

Building 235-F and corrective maintenance as needed to prevent rainwater infiltration, structural deterioration/collapse, and spread of contamination. The No Action scenario assumes a roof collapse at 150 years at which time additional response action(s) will be necessary to mitigate exposure and spread of contamination.

4.2 Alternative A-2, ISD of First and Second Level Process Areas/Engineered Roof, Activity Description

Table 2 provides a list of the general activities that will be necessary to grout the first and second level process areas (approximately 6,600 m^3 [8,600 cubic yards {yd^3}] of grout), grout the underground storage tank, permanently seal the capped stack, and construct a sloped reinforced concrete roof over the entire Building 235-F structure (Figures 13 and 14). For the process areas, a summary of the overall removal action activities for Alternative A-2, includes:

- installing recombiners in the process enclosures to mitigate hydrogen build up;
- sealing of process enclosures to prevent contaminant release;
- installing covers over glass windows to protect them from the grout pressure;
- grouting below-grade transfer trench between PuFF cells;
- placement of formwork in process areas such that zones can be grouted in a phased approach, if needed;
- applying fixative for contamination control and moving ventilation ducts to the floor or penetrating the ductwork to mitigate voids and allow grout infiltration where possible;
- installing core holes in the concrete above the process rooms for grout placement via slick lines and tremies;
- managing ventilation via temporary equipment (huts, fans/blowers, filtration, supplied air, etc.) for containment and personnel safety; and
- grouting the process areas.

Some non-process areas will also be grouted either because they contain contaminated equipment or ventilation ducts (e.g., Zones 7, 8, and 9) or to eliminate void spaces directly above or below process areas that will be grouted (e.g., Zones 12 and 13). Grouting these non-process areas represents the facility conditions in the F&T model scenarios and will prevent contaminant migration by minimizing preferential pathways for water intrusion and provide structural support for the additional loading from the grout. All areas to be grouted are shown on Figures 13 and 14.

Because Alternative A-2 does not include grouting all rooms, structural supports will be needed for the roof loads created by the reinforced concrete roof that will be placed over the entire building. Structural supports will be needed in the first and second level rooms that are not grouted. Interior areas that are not grouted may require decontamination or application of a fixative.

Once the process enclosures are no longer ventilated and are sealed, there is a potential to accumulate hydrogen concentrations at the lower flammability limit after a significant period of time (varies by enclosure from approximately 1.5 years to >250 years). Ignition of the hydrogen inside the sealed and entombed process enclosure, while unlikely, could result in a deflagration[1]. Uncertainties associated with this potential will be managed by several layers of defense. The robust defenses used to mitigate the potential for a hydrogen deflagration, as discussed in the following paragraphs, are adequate and far outweigh the risks that would be incurred to decommissioning workers if grouting the interior of the enclosures was included in this alternative. The risk associated with worker exposure and potential release of residual radiological contamination is extremely high because contamination remaining in PuFF Cells 1-5 and in PEF is mostly submicron and distributed throughout the cell ventilation system as well as on all surfaces inside the cells. These cells and process enclosures are under a negative pressure to prevent migration, so opening the cells to effectively place grout (requires several lifts separated by a sufficient time interval) significantly increases the risk of an uncontrolled release and was not evaluated as a removal alternative.

The first layer of defense in preventing a flammable concentration of hydrogen in the process enclosures is to add a combustible gas control system such as recombiners prior to sealing. Recombiners include a perforated housing that contains a precious metal catalyst to efficiently and safely recombine hydrogen and oxygen gas to form water vapor. A desiccant is used to control

[1] Deflagration is combustion which propagates through a gas or across a surface at subsonic speeds. Because there are no flammable materials within the cells/process enclosures, a deflagration would result in relatively low pressures.

moisture. There is no chemical consumption, so the recombiners will not deplete over time. Design calculations will be required to determine the spacing and number of recombiners needed.

Although deactivation will remove many ignition sources, the second level of defense is to eliminate ignition sources remaining after deactivation activities in the unlikely event the recombiners fail or degrade. Decommissioning activities will remove ignition sources originating from equipment via electrical isolation. Grouting around the process enclosures and sealing the exterior openings eliminates possible ignition sources associated with human activity such as use of sparking tools, operation of gas-powered equipment, etc.

Finally, the entombment of the process enclosures within several tens of feet of grout will reduce impacts of and contain the release should a deflagration occur within a cell. Although deflagration is not likely, the grout acts as a containment barrier, eliminating contaminant releases from the process areas.

A preconceptual design was developed for Alternative A-2 and Alternative A-3 to support the removal alternative evaluation (SRNS 2021b). Preliminary design for the reinforced concrete roof indicates that the concrete ventilation duct and all other structures, handrails, guard shack, stairs, etc., will need to be removed. If this alternative is selected, a detailed design of the roof construction will be prepared, and consideration will be given to leaving the concrete roof ducts in place and filling with grout to avoid risks to decommissioning workers during demolition and reduce risks associated with the handling and transportation of the demolition waste. For purposes of the removal alternative comparison, however, removal of the concrete duct is assumed to be required. Contaminated material removed from the roof will presumably be disposed of as low-level waste (LLW). All roof penetrations will be grouted and sealed prior to constructing the roof. Roof penetrations above areas that are not grouted will require formwork to construct a grout plug.

Exterior penetrations, doors, and any openings created for construction equipment placement will be sealed with reinforced concrete, similar to what was done for openings in the 105-P Reactor Building Complex (RBC) and the 105-R RBC after ISD was complete (SRNS 2012a and SRNS 2012b).

The underground storage tank will be filled with grout in lifts to prevent airborne release of contaminants from the tank. Enclosure huts will be required to prevent airborne contaminant migration. The fill and vent lines will be capped by filling them with grout. The pipe trench leading from Building 235-F to the underground storage tank will also be grouted.

Currently, the 8.5 m (28 ft) above ground surface (ags) abandoned capped stack (293-F) is sealed with an aluminum cover. The aluminum cover will be replaced with a concrete plug that will require formwork and grout placement at the same elevation of approximately 8.5 m (28 ft) ags.

4.3 Alternative A-3, ISD of Entire Building/Engineered Roof, Activity Description

Table 2 provides the general activities that will be necessary to grout all rooms within Building 235-F (approximately 18,000 m^3 [23,000 yd3] of grout), grout the underground storage tank (Figures 15 and 16), and construct a sloped reinforced concrete roof over the entire Building 235-F structure. The major difference between Alternative A-2 and Alternative A-3 is that in addition to all process areas, all non-process areas will be grouted in Alternative A-3. A summary of the overall activities for Alternative A-3, includes:

- installing recombiners in the process enclosures to mitigate hydrogen build up;
- sealing of process enclosures to prevent contaminant release;
- installing covers over glass windows to protect them from the grout pressure;
- grouting the below grade transfer trench between the PuFF cells;
- applying fixative for contamination control and moving ventilation ducts to the floor or penetrating the ductwork to mitigate voids and allow grout infiltration where possible;
- installing core holes in the concrete above the rooms for grout placement via slick lines and tremies;
- managing ventilation via temporary equipment (huts, fans/blowers, filtration, supplied air, etc.) for containment and personnel safety; and
- grouting all rooms in Building 235-F.

The first level will be entirely grouted prior to grouting the second level. Alternative A-3 will allow for a grouting strategy to include simultaneous lifts in adjoining areas. Combining areas for simultaneous grout pours may eliminate the need for some formwork placement.

The roof design is the same for Alternatives A-2 and Alternative A-3. Alternative A-3 will eliminate the need for formwork associated with sealing roof penetrations prior to constructing the reinforced concrete roof because there will be no void spaces below the roof. Additionally, filling all rooms with grout will eliminate the need for structural supports for the roof loading as required for Alternative A-2.

Exterior penetrations, doors, and any openings created for construction equipment placement will be sealed with reinforced concrete as described for Alternative A-2. As discussed in Alternative A-2, the underground storage tank will be grouted, and the fill and vent lines will be capped by filling with grout. The underground pipe trench will also be grouted. The aluminum cap on the abandoned capped stack (293-F) will be replaced with a concrete plug.

4.4 Alternative A-4, Complete Building 235-F Removal/Soil Cover, Activity Description

Complete demolition of Building 235-F will first require removal of the gloveboxes, process enclosures, ventilation ducts, etc., that are contaminated with Pu-238 and Np-237 holdup. Previous evaluations have determined that the risk to the deactivation worker is too great to warrant removal of the material in the process cells and enclosures; however, for purposes of alternative comparison, minimum activities that would be required to implement this alternative scenario are described below.

Prior to removing the process enclosures and contaminated piping and equipment, a fixative will be applied to the contaminated surfaces to mitigate airborne release during demolition/removal activities. Removal of these items will require construction of containment huts, temporary ventilation, several changes of personal protective equipment (PPE) per shift, and supplied air for workers.

Under the complete Building 235-F removal scenario, the below grade transfer trenches between the PuFF cells will be grouted and the building will be demolished to the building slab. The abandoned capped stack (293-F) will be removed, and the underground storage tank and pipe trench will be grouted. A soil cover will be constructed over the building slab and the grouted underground tank and will encompass the area beneath the abandoned capped stack (293-F).

Removal of the process enclosures and contaminated equipment, as well as demolition of the structure, will result in significant costs associated with the potential for TRU waste, TRU mixed waste, LLW, and LLW mixed waste disposal. It is only within discrete process areas that may contain transuranic isotope holdup/concentrations that would exceed the 100 nanocuries/g threshold and result in a TRU and/or TRU mixed waste determination upon disposal. In addition to radiological holdup, All radiologically contaminated equipment/ventilation ducts, etc., will be characterized (to include hazardous waste determination), packaged, and transported off-site to a TRU waste disposal facility. Hazardous material and low-level radionuclide material will be removed and transported to the appropriate disposal facility. The majority of the building rubble will is expected to be disposed of as LLW.

5.0 ANALYSIS AND COMPARISION OF REMOVAL ACTION ALTERNATIVES

Four removal action alternatives are presented in this RSER/EE/CA for evaluation. According to the NCP, the No Action Alternative, Alternative A-1, must be evaluated as a baseline. Building 235-F is undergoing deactivation as directed by the USDOE; therefore, the deactivation end state is considered to be representative of the conditions for the No Action Alternative. Alternative A-2 proposed actions include grouting the first and second level process areas, constructing a sloped reinforced concrete roof over the entire Building 235-F structure, grouting the underground storage tank, and permanently sealing the abandoned capped stack (293-F) with a concrete plug. Alternative A-3 proposed actions include grouting all rooms within Building 235-F (i.e., both process and non-process areas), constructing a sloped reinforced concrete roof over the entire Building 235-F structure, grouting the underground storage tank, and permanently sealing the abandoned capped stack (293-F) with a concrete plug. Alternative A-4 proposed actions include grouting the below grade PuFF cell transfer trenches, demolishing Building 235-F to the building slab, removing the abandoned capped stack (293-F), grouting the underground storage tank, and constructing a soil cover over the remaining building slab, abandoned capped stack (293-F) area, and underground storage tank.

USEPA guidance on conducting NTC removal actions under CERCLA recommends that each alternative be reviewed against three broad criteria: effectiveness, implementability, and cost.

Regulatory acceptance and community acceptance are usually not known until after the comment periods for each. However, during the removal alternative analysis, a judgment as to acceptance may be included based on previous regulatory decisions or on public comment to other related documents. The final impact of these modifying criteria can be assessed only after the comment periods and after responses are developed.

5.1 Effectiveness

Long-term effectiveness is evaluated for each alternative on the basis of the magnitude of residual risk and/or the adequacy and reliability of controls used to manage contaminated media that remain after response objectives have been achieved. Alternatives that offer long-term effectiveness halt or mitigate any potential for offsite contaminant transport and minimize the need for future engineered controls. The degree of uncertainty with regard to treatment effectiveness is also evaluated. Evaluation of alternatives for short-term effectiveness takes into account the protection of decommissioning workers, members of the community, and the environment during implementation of the removal action and the time required to achieve removal action objectives.

5.1.1 Alternative A-1, No Action

Alternative A-1, No Action**,** does not meet the effectiveness criteria. Leaving the radiological holdup and other contaminants in the current process enclosures and radiological contamination in place outside of the process enclosures does not reduce the risk to the hypothetical future industrial worker and does not provide overall protection to human health or the environment as evaluated in the HHRA. Using the default indoor worker receptor scenario (i.e., hypothetical future industrial worker), the risk from exposure to residual radiological contamination exceeds 2E+09. This alternative does not contribute to a reduction of toxicity, mobility, or volume through treatment.

F&T modeling of the migration of Building 235-F contaminants, as discussed in Appendix B, was performed for each of the No Action scenarioremoval action alternatives. The No Action scenario was found to be ineffective at preventing groundwater contamination above MCLs at the POA(s) within the 0 to 10K years assessment period. Under the No Action scenario, the existing roof was assumed to collapse at 150 years. Modeling indicates that Np-237 will exceed the MCL threshold for beta-gamma (4 mrem/yr) and gross alpha (15 pCi/L), and Pu-238 will exceed the MCL for beta-gamma, gross-alpha, and radium (5 pCi/L) for the No Action scenario at the 1-m (3-ft) POA within the 0 to 10K years assessment period. For the No Action scenario at the 360-m (1,181-ft) POA and the 0 to 10K years assessment period (i.e., baseline building deactivated state), Np-237 is predicted to exceed the MCL threshold for gross alpha.

Short-term effectiveness will be achieved administratively by limiting and controlling access to Building 235-F. SRS workers will monitor and inspect Building 235-F periodically with appropriate PPE. Ventilation to the process enclosures and rooms will be maintained by drawing in unconditioned air. Maintenance will be performed as necessary to mitigate rainwater infiltration and contaminant migration. Over time, conditions within Building 235-F will deteriorate and will increase risk to the SRS workers performing inspections and maintenance.

This alternative leaves radiological contamination and hazardous substances within a deactivated facility with limited utilities. Although S&M activities will be performed periodically, the eventual deterioration of the building increases risks to co-located workers with respect to a potential release from the facility due to an accident scenario or natural phenomena. Future missions may also be impacted by the presence of this potentially hazardous condition.

5.1.2 Alternative A-2, ISD of First and Second Level Process Areas/Engineered Roof

Alternative A-2 meets the effectiveness criteria. ISD, grouting of the process rooms and underground storage tank, as well as the construction of a reinforced concrete roof and permanent seal for the abandoned capped stack (293-F), provides overall protection to human health and the environment as discussed in the following paragraphs.

Long-term effectiveness will be achieved by this alternative via eliminating access to the radiological and hazardous contamination in the process rooms. ISD of the process rooms will entomb the contamination in several feet of grout and prevent access to the process enclosures. Additionally, the building exterior openings will be grouted, and the entrances will be permanently sealed to prevent entry into the building and exposure to contamination.

Construction of a sloped reinforced concrete roof, in addition to the grouting of the process rooms, will provide long-term effectiveness with respect to contaminant migration. F&T modeling of the migration of Building 235-F contaminants, as discussed in Appendix B, was performed for this removal alternative scenario. Alternative A-2 was found to be effective at preventing groundwater contamination above MCLs at the UTR tributary POA within the 0 to 10K years assessment period. Under Alternative A-2, the reinforced concrete roof was anticipated to remain intact for 1,000 years. Modeling indicates that neither Np-237 nor Pu-238 will exceed the MCL thresholds at any POA (i.e., 1-m [3-ft], 100-m [328-ft], 360-m [1,181-ft], seep line, and UTR tributary) within the 0 to 10K years assessment period.

In addition to eliminating human exposure to contaminants via grouting and preventing migration of contaminant to the groundwater above MCLs, Alternative A-2 will provide long-term effectiveness of exposure to non-radiological hazardous material including asbestos, lead, and PCBs. These non-radiological materials will be encapsulated in grout within the process areas, and exposure to these materials will be eliminated in non-process areas by preventing access by sealing all exterior openings to Building 235-F. ACM in the existing roof will be covered by the newly constructed sloped reinforced concrete roof.

Uncertainties associated with the long-term impacts of leaving PTSM and hazardous substances entombed within Building 235-F will be managed with long term inspections and maintenance. The presence of these contaminants left in place will preclude unrestricted use of this site.

Alternative A-2 provides the greatest potential to achieve short-term effectiveness. Entombing the process enclosures with grout minimizes the need for decommissioning workers to enter the cells or the enclosures and thus reduces exposure to, and mitigates the potential for, an airborne release of radioactive materials. In this alternative, the PuFF facility could be grouted first, thereby preventing exposure to 85% of the Pu-238 inventory that accounts for the majority of the risk. Risk estimates for just the PuFF facility based on the hypothetical future industrial worker scenario are 5.9E+10. Similarly, once the ABL process room which contains 87% of the Np-237 inventory is grouted, risks will be significantly reduced. Risk estimates for just the ABL process room based on the hypothetical future industrial worker scenario were 6.9E+08 from Pu-238 and 1.6E+06 due to Np-237. If a phased approach is needed to implement the removal action (e.g., funding availability or resource limitations), significant risk reduction could be achieved in the short-term by focusing the removal action activities on the first-floor process rooms.

In addition to eliminating exposure to the areas with the highest radionuclide inventories, Alternative A-2 will provide short-term effectiveness by minimizing the potential to spread contamination during grouting. Implementation of this alternative will rely on structural barriers (i.e., existing concrete walls and newly constructed formwork) to limit the flow of grout within the confines of the zone being grouted. This implementation strategy will mitigate the spread of contamination that could be transported with the grout flow. The potential for radiation exposure to decommissioning workers will be slightly higher due to increased presence and activity associated with constructing the formwork, but will be managed by using temporary ventilation, PPE, and rotating decommissioning workers to limit their exposure time.

Additional decommissioning worker risk will be present due to radiological exposure and industrial hazards associated with the removal of the concrete ventilation duct on the Building 235-F roof. A pre-conceptual design for the construction of a sloped reinforced concrete roof requires that all obstructions on the existing roof be removed. If residual contamination is present in the concrete ventilation duct, demolition will need to be performed under containment huts with personnel in the appropriate level PPE. Standard industrial hazards associated with chipping and size-reducing concrete, and crane operation required to remove the waste from the 8.5 m (28 ft) high roof will reduce short-term effectiveness with respect to decommissioning worker hazards.

5.1.3 Alternative A-3, ISD of Entire Building/Engineered Roof

Alternative A-3 meets the effectiveness criteria. ISD, grouting of the entire Building 235-F and underground storage tank, as well as the construction of a reinforced concrete roof and permanent seal for the abandoned capped stack (293-F), provides overall protection to human health and the environment as discussed in the following paragraphs.

Long-term effectiveness will be achieved by this alternative via preventing access to the radiological and hazardous contamination in the process rooms where the radiological holdup is contained, as well as preventing access to areas that are slightly contaminated or where no contamination exists. ISD of the entire building will entomb the contamination in several feet of grout and prevent access to the process enclosures. Like Alternative A-2, the building exterior openings will be grouted, and the entrances will be permanently sealed to prevent entry into the building.

Construction of a sloped reinforced concrete roof will provide long-term effectiveness with respect to contaminant migration. As presented in Appendix B, grouting the entire Building 235-F provides the same effectiveness as Alternative A-2 in preventing groundwater contamination above MCLs at the POAs within the 0 to 10K years assessment period.

Uncertainties associated with the long-term impacts of leaving PTSM and hazardous substances entombed within Building 235-F will be managed as described for Alternative A-2. The presence of these contaminants left in place will preclude unrestricted use of this site.

As discussed for Alternative A-2, entombing the process enclosures in grout minimizes the decommissioning worker entry into the cells and increases short-term effectiveness with respect to exposure to and release of contaminants. This alternative has the potential to achieve short-term effectiveness should a phased approach to implementing the removal action be needed due to funding availability or resource limitations. As previously discussed, the structural loading from the addition of the grout will require that the entire first building level be grouted prior to grouting the second level. The PuFF and ABL process areas located on the first level contain 85% and 87% of the Pu-238 and Np-237 residual contamination, respectively. By encapsulating these areas, short-term effectiveness will be achieved by significantly reducing risk based on the hypothetical future industrial worker scenario.

During implementation, short-term effectiveness with respect to radiation exposure to decommissioning workers may be greater if the grouting is performed with reduced formwork. However, less formwork would also result in grouting the entire first level in lifts across large areas which has the potential to spread contamination within Building 235-F via grout flow and thus increase exposure to decommissioning workers within the building. Short-term effectiveness with respect to decommissioning worker exposure and hazards associated with removal of the concrete ventilation ducts on the Building 235-F roof will be impacted as described in Section 5.1.2 for Alternative A-2.

5.1.4 Alternative A-4, Complete Building 235-F Removal/Soil Cover

Alternative A-4 meets the long-term effectiveness criteria but provides a significant risk in terms of the short-term effectiveness. Complete removal of Building 235-F to the bottom slab, removal of the abandoned capped stack (293-F), grouting the underground storage tank as well as the construction of a soil cover over the building slab, abandoned capped stack (293-F) footprint, and grouted underground tank provides overall long-term protection to human health and the environment by removing the residual radiological contamination and contaminated equipment, ventilation piping, concrete, etc. Any contamination entrained within the concrete slab and grouted PuFF cell transfer trenches will be covered by a minimum of 0.6 m (2 ft) soil and vegetative cover. There is no long-term impact to groundwater associated with this alternative because the contamination source will be removed.

Residual contamination within the grouted transfer trenches will be left in place under this alternative scenario and will preclude unrestricted land use. Long-term effectiveness will be monitored with soil cap inspections and maintenance.

Prior to demolition, the highly contaminated process enclosures will need to be removed from the building which will greatly reduce the short-term effectiveness due to the potential to release contamination and expose decommissioning workers. As previously mentioned, the radiological inventory identified that 85% of the Pu-238 remaining in the building is located in the PuFF cells and 87% of Np-237 is in ABL. Residual radiological contamination remaining in PuFF Cells 1-5 and in PEF is mostly submicron and distributed throughout the cell ventilation system as well as on all surfaces inside the cells. These cells and process enclosures are under a negative pressure to prevent migration, so opening the cells to prepare for removal and demolition significantly increases the risk of an uncontrolled release. High specific activity and small particle size give Pu-238 oxide particles substantial mobility that significantly adds to the decommissioning worker risk during demolition activities. During deactivation, some of the radiological holdup was removed; however, risk to the deactivation worker was deemed too great (e.g., for a puncture/laceration accident, a single exposure dose of >100 rem would exceed the DOE-STD-3009 consequence threshold for a facility) to justify further removal of any additional radiological contamination. In order to remove the remaining radiological holdup, robotics will likely be needed to maneuver a spray wand around the cell interior to fog/apply a fixative, vacuum cell floors, move equipment within the cells, and cut cables and cap utilities. Even with the use of robotics, decommissioning worker exposure risk remains high due to handling of the contaminated material. This alternative requires significantly more personnel to perform demolition, waste handling, and other support functions during implementation and will result in the highest risk to decommissioning workers.

Short-term effectiveness is jeopardized by the potential for an airborne release during radiological holdup removal and demolition activities. The risk of the Pu-238 submicron particles becoming airborne is high, and airborne contamination is difficult to contain. SRS has successful experience in manipulating Pu-238 material within a controlled environment (e.g., sealed cells, using mechanical manipulator arms, lead, and other fixed shielding, etc.), but has less experience with employing temporary, multi-layered controls required to reduce the risk of release of airborne contamination during demolition activities.

In addition to the potential for exposure to highly mobile radioactivity, demolition will present decommissioning worker exposure to friable asbestos because this alternative requires dismantlement and removal of ACM. As discussed in Appendix A of this report, once ACM is disturbed, friable material becomes a significant breathing hazard to decommissioning workers.

Short-term effectiveness is also reduced by the amount of waste packaging, waste handling, and waste transportation. Waste will have to be segregated in accordance with appropriate acceptance criteria. TRU waste will require off-SRS disposal and will reduce short-term effectiveness with respect to specialized characterization, containerization, and shipping hazards.

5.2 Identification of Applicable or Relevant and Appropriate Requirements

In accordance with 40 CFR § 300.415(j) of the NCP, on-site removal actions conducted under CERCLA are required to attain applicable or relevant and appropriate requirements (ARARs) to the extent practicable, considering the exigencies of the situation. In determining whether compliance with ARARs is practicable, the lead agency may consider appropriate factors, including the urgency of the situation and the scope of the removal action. ARARs include only federal and state environmental or facility siting laws/regulations; they do not include occupational safety or worker protection requirements. Compliance with OSHA standards is required by 40 CFR § 300.150. For purposes of ease of identification, the USEPA has created three categories of ARARs: Chemical-, Location-, and Action-Specific. Additionally, per 40 CFR § 300.405(g)(3), other advisories, criteria, or guidance may be considered in determining remedies (to-be-considered [TBC] category). USDOE, the lead agency at the SRS, is expected to comply with ARARs and to-be-considered (TBC) guidance as set forth in the RSER/EE/CA when conducting this NTC removal action.

Applicable requirements, as defined in 40 CFR § 300.5, means those cleanup standards, standards of control, and other substantive requirements, criteria, or limitations promulgated under federal environmental or state environmental or facility siting laws that specifically address a hazardous substance, pollutant, or contaminant, remedial action, location, or other circumstance at a CERCLA site. Relevant and appropriate requirements, as defined in 40 CFR § 300.5, means those cleanup standards, standards of control, and other substantive requirements, criteria, or limitations promulgated under federal environmental or state environmental or facility siting laws that, while not "applicable" to a hazardous substance, pollutant, or contaminant, remedial action, location, or other circumstance at a CERCLA site, address problems or situations sufficiently similar to those encountered at a CERCLA site that their use is well suited to the particular site. Only those state standards that are identified by the state in a timely manner and that are more stringent than federal requirements may be relevant and appropriate.

Under Section 121 of CERCLA, any material remaining on site must reach a level or standard of control equal to that of any other applicable or relevant and appropriate standard or requirement promulgated under any federal or more stringent state environmental statute. The term "promulgated" means that the requirement generally is applicable and legally enforceable. The ARAR concept is pertinent only to onsite actions; offsite actions must comply with all applicable federal and state requirements. A requirement under other environmental laws may be either "applicable" or "relevant and appropriate," but not both. The first step in identifying ARARs is to determine if a requirement is applicable.

ARARs are identified for the Building 235-F NTC removal action in Appendix C. This RSER/EE/CA does not propose to waive any ARARs. Potential Action-Specific ARARs include Resource Conservation and Recovery Act (RCRA) waste characterization, storage, treatment and disposal requirements; asbestos abatement and disposal requirements; and Toxic Substances Control Act (TSCA) standards for PCB bulk product waste. The actions undertaken as part of the NTC removal action along with any relevant exposure information demonstrate that PCB bulk product materials that remain in the grouted building do not *pose an unreasonable risk or injury to health or the environment* as required by 40 CFR 761.62(c). The final NTC removal action (decommissioning) end state for Building 235-F will be documented as the CERCLA remedial action end state (i.e., requiring no further response action for protection of human health and the environment) in the FAOU ROD. Completion activities are identified in the FFA, a legally binding and enforceable tri-party agreement between USDOE and the two regulatory agencies, USEPA and SCDHEC.

Consideration of NEPA Values

This RSER/EE/CA conforms to USDOE policy (i.e., DOE Order 451.1B, "National Environmental Policy Act Compliance Program") to incorporate NEPA values in USDOE CERCLA documents. NEPA values, e.g., the analysis of cumulative, off-site, ecological, and socioeconomic impacts, and cultural resource impacts of the proposed action, are addressed below. A human health risk assessment has been performed as prescribed by USEPA risk assessment guidance and SRS protocols and is provided in Appendix A of this document. CERCLA risk assessments and the associated remedial clean up goals are protective of hypothetical future workers at the site, and thus are necessarily protective of offsite receptors. Any potential environmental releases resulting from implementation of the preferred alternative would be limited to the vicinity of the project area and would be managed via engineering controls. Impacts beyond the SRS boundary would be negligible, ensuring that there are no environmental justice concerns associated with the proposed removal action.

Ecological and Natural Resources

Since 1951, natural resource management practices and natural succession outside of construction and operation areas have increased the ecological complexity and diversity of the SRS. SRS was designated as the first National Environmental Research Park in 1972 and is one of the most extensively studied environments in the U.S. Forested areas support a diversity of wildlife habitats which have been largely restricted from public access or use. Open fields and pine and hardwood forests make up 73% of SRS, while 22% is wetlands, streams, and two large lakes. Production and support areas, roads, and utility corridors account for the remaining 5% of the land area (USDOE 2005). The U.S. Department of Agriculture - Forest Service (USFS), under an interagency agreement with the USDOE, manages natural resources on about 170,000 acres. USFS forest management practices include prescribed burning, harvesting of mature trees, and reforestation. Wildlife management includes public hunts on SRS for white-tailed deer (*Odocoileus virginianus*), feral hogs (*Sus scrofa*), wild turkeys (*Meleagris gallopavo*), and coyote (*Canis latrans*). Tree harvest and supervised hunts do not occur near Building 235-F. The proposed removal action is not anticipated to produce adverse effects on these ecological and natural resources.

Threatened or Endangered and Protected Species

Federal or South Carolina endangered or threatened plants and animals known to occur at SRS include the smooth purple coneflower (*Echinacea laevigata*), pondberry (*Lindera melissifolia*), bald eagle (*Haliaeetus leucocephalus*), wood stork (*Mycteria americana*), red-cockaded woodpecker (*Picoides borealis*), shortnose sturgeon (*Acipenser brevirostrum*), American swallow-tailed kite (*Elanoides* forficatus), gopher tortoise (*Gopherus polyphemus*), and Rafinesque's big-eared bat (*Corynorhinus rafinesquii)* (USDOE 2020). The location of Building 235-F in F Area is industrial, covered with buildings, parking lots, and industrial equipment. In a highly developed area such as F Area, the presence of threatened or endangered and protected plants and animals is unlikely. The proposed removal action is not anticipated to adversely affect the threatened or endangered and protected species listed above.

Floodplains and Wetlands

Wetlands are habitats dominated by hydrophytes, have saturated soils, or are periodically or permanently covered with water. The location of Building 235-F is in an existing industrial area that does not contain any wetlands. Aquatic plant and animal species do not occur near Building 235-F because of the absence of an aquatic habitat. The proposed removal action will not result in adverse impacts to wetlands or floodplains.

Socioeconomics

SRS is located ~40-km (25-mi) southeast of Augusta, Georgia and ~32-km (20-mi) south of Aiken, South Carolina. SRS's socioeconomic region of interest (ROI) is the four-county area of Columbia (GA), Richmond (GA), Aiken (SC), and Barnwell (SC). In 2019, the population in the ROI was estimated to be 550,970 according to the U.S. Census Bureau (USCB 2019). More than 86% of SRS employees reside in the four counties within the ROI (USDOE 2020). The ROI for environmental justice also considers parts of 28 counties throughout South Carolina and Georgia that make up an area within a 50-mile radius of SRS.

The workforce to be employed and the projected costs associated with the proposed removal action for Building 235-F are minimal when compared to the total SRS budget and employment and would have no socioeconomic impact in the ROI. Impacts beyond the SRS boundary would be negligible, ensuring that there are no environmental justice concerns associated with the proposed removal action.

Archaeological, Historical, and Cultural Resources

Concurrence was received in 2009 from the South Carolina Historical Preservation Office that all historical mitigating actions for Building 235-F were satisfied per the SRS's Cold War Built Environmental Cultural Resource Management Plan (CRMP) (USDOE 2005). Artifact selection and retrieval is complete, and historical release of Building 235-F has been received.

The site of the proposed removal action is located within an established industrial landscape. Implementation of the proposed removal action would have a negligible impact on SRS archaeological, cultural, or historical resources.

Cumulative Impacts

NEPA regulations define cumulative impacts as "…the incremental impact of [an] action when added to other past, present, and reasonably foreseeable future actions, regardless of what agency (Federal or non-Federal) or person undertakes such other actions. The cumulative impacts of an action can be viewed as the total effects on a resource (e.g., land, air, water, soil), ecosystem, or human community of that action and all other activities affecting that resource no matter what entity (Federal, non-Federal, or private) is taking the actions (USEPA 1999)." Cumulative effects are assessed by combining the effects of the proposed removal action with the other past, present, and reasonably foreseeable future actions in the ROI. As previously discussed, there are no notable incremental impacts to resources from the proposed removal action for Building 235-F.

5.3 Implementability

Implementability of each alternative was assessed against the criteria below:

- Technical feasibility with regard to available techniques and demonstrated methods for accomplishing the proposed alternative;
- Administrative feasibility with regard to operations personnel and other resources to complete the removal alternative's implementation and the availability of specific equipment and technical specialists;
- Regulatory acceptance of the preferred alternative; and
- Community acceptance of the preferred alternative. USDOE–Savannah River will provide a public comment period, and comments concerning the proposed remedy will be incorporated into the comment responses and included with the Action Memorandum.

5.3.1 Alternative A-1, No Action

The No Action alternative includes leaving Building 235-F in the deactivation end state. Because Building 235-F is a hHazard cCategory 2 nonreactor nuclear facility, a negative pressure must be maintained on the process enclosures to provide confinement of the radiological holdup, and minimum S&M activities will be required. It is unacceptable, from a nuclear facility hazard perspective, to eliminate the resources associated with minimal operation and S&M.

Implementation of Alternative A-1, No Action, will involve continued operation of the E5 fan exhaust ventilation system as well as operation and S&M of the Sand Filter Fan House (292-2F) and air compressors. Program support activity with respect to nuclear material hazards will continue to be implemented. Because other support systems (e.g., conditioned air, power supply, domestic water, etc.) will be deactivated, over time the facility will deteriorate and pose additional hazards to SRS workers. The levels of Pu-238 and Np-237, and other hazardous constituents to a lesser extent, in Building 235-F exceed the acceptable CERCLA risk to the hypothetical future industrial worker. The No Action alternative would not be acceptable to USDOE, USEPA, and SCDHEC.

Although implementation of Alternative A-1 is technically and administratively feasible, over time, conditions within Building 235-F will deteriorate and pose unacceptable risks. These risks have been recognized by other programs with respect to nuclear safety. In 2012, the Defense Nuclear Facilities Safety Board made recommendations to reduce the hazards associated with the Pu-238 holdup in Building 235-F (DNFSB 2012). Consequences of an unmitigated design basis event (natural phenomena event or accident) have been estimated to result in an unacceptable dose to co-located workers which impacts future missions at the SRS and further supports that the No Action alternative would not be acceptable.

5.3.2 Alternative A-2, ISD of First and Second Level Process Areas/Engineered Roof

ISD of the first and second level process areas is technically feasible as demonstrated in other ISD remediations performed at SRS such as the 105-P RBC, 105-R RBC, and the 105-C Disassembly Basin (SRNS 2012a, SRNS 2012b, SRNS 2013). The necessary resources to install the grout include common materials/resources that are readily available. A preconceptual design of the grouting process includes drilling cores through the concrete ceiling (second level floor and Building 235-F roof) above the areas to be grouted (SRNS 2021b). Where possible, ventilation ducts will either be removed and placed on the floor or holes will be drilled in the ventilation ducts to allow grout infiltration. The inside of the process enclosures will not be grouted due to high specific activity and substantial mobility of the radiological contamination that significantly add to the personnel risk and potential of release from breaching these enclosures. The grout placement will be designed to limit void spaces within the process rooms and outside of the process enclosures.

Structural supports will be needed in the areas that are not grouted to support the roof loads from the addition of a reinforced concrete roof. The structural supports require common construction techniques but will require analysis and design.

The preconceptual design also indicates that removal of the concrete duct work on the Building 235-F roof will be required to construct a concrete reinforced roof with a sufficient slope to prevent rainwater ponding (SRNS 2021b). Containment huts will likely be required to prevent the spread of contamination during its removal. The interior of the roof duct has not been sampled, but levels of contamination within this structure are thought to be such that the rubble will require disposal as LLW. Disposal quantity of debris associated with the concrete ventilation ducts is estimated to be approximately 650 tons (589,670 kg). SRS has experience in construction of reinforced concrete roofs, as demonstrated by similar construction activities on portions of the 105-P RBC and 105-R RBC. Grouting of contaminated tanks has also been performed at SRS with success. Personnel are readily available and technologies for grouting and roof construction are well defined.

Administratively, this alternative provides the most flexibility in implementation because it is possible to grout one process area at a time should a phased approach to implementing the removal action be needed due to funding or resource limitations. As discussed in Section 5.1.2, grouting of the process rooms in phases will reduce short-term risks by entombing the process enclosures that contain the greatest amount of radiological contamination.

5.3.3 Alternative A-3, ISD of Entire Building/Engineered Roof

Similar to Alternative A-2, ISD of the entire building is technically feasible as demonstrated in other ISD activities performed at SRS. The necessary resources to install the grout are common materials/resources that are readily available. A preconceptual structural design of this alternative indicated that very little formwork will be required to implement this alternative if nearly simultaneous lifts were poured throughout each level (SRNS 2021b). For structural support, the entire first level will have to be grouted prior to grouting the second level. An exterior access way to the second level will be created for equipment and placement of temporary piping (called 'slick lines") for grouting. Grouting of the first level will be conducted by drilling core holes through the second level concrete floor. Similar to Alternative A-2, the process enclosure interiors will not be grouted and void spaces in other areas, such as ventilation ducts, will be minimized by detaching the ductwork and placing in the floor or by drilling into ductwork to allow grout intrusion, where possible.

This alternative may be more technically feasible with respect to the need for formwork placement as compared to Alternative A-2 if grout could be poured throughout an entire level using simultaneous lifts. However, grouting an entire level as compared to grouting only the process areas by zones presents additional challenges. Resource availability such as grout production and truck availability may limit the ability to place grout in simultaneous lifts. Additionally, managing slick lines for simultaneous lifts throughout one entire level will be more difficult than Alternative A-2. Daily operation of multiple slick lines will be required and managing slick lines with grout delivery to avoid hardening of the grout within the lines increases in complexity with the number of slick lines required. Spread of contamination via the grout flow is possible within the interior of the building, although some concrete interior walls will limit movement within Building 235-F and exterior concrete walls will prevent release of contamination outside of the building.

Because all rooms will be grouted, implementation of this alternative eliminates the need for structural supports for the reinforced concrete roof.

Similar to Alternative A-2, it is assumed that the concrete ventilation ductwork on the roof will be required to be removed prior to installing a sloped reinforced concrete roof. Debris volumes, disposal, and implementation of containment measures associated with demolition of the existing roof structures (as described in Section 5.3.2) are the same for Alternative A-2 and Alternative A-3.

Administratively, this alternative provides flexibility in implementation should funding be allocated in phases rather than all at once. Implementation of Alternative A-3 could proceed by level, grouting the entire first level in Phase 1 and grouting the second level and roof modifications in subsequent phases.

5.3.4 Alternative A-4, Complete Building 235-F Removal/Soil Cover

Alternative A-4, Complete Building 235-F demolition, debris disposal and construction of a soil cover offers the most challenges in technical implementability as compared with the other alternatives. Although demolition of concrete buildings and debris disposal is not a unique construction activity, the containment of highly mobile radionuclides that have high specific activity and small particle size will require the use of specialty contractors, equipment and containment structures. Prior to demolition, the process enclosures and the contaminated equipment and structures containing radiological holdup will need to be removed and appropriately size reduced in order to be packaged and transported to an appropriate disposal facility. As discussed in Section 5.1.4, the use of robotics may be necessary to remove radiological holdup from within some of the cells, which will add to the complexity of the implementation in terms of design, schedule and resource availability.

Wastes to be disposed of from this alternative will include TRU waste, hazardous waste, mixed waste, and LLW. Offsite shipment and disposal agreements with other facilities will need to be coordinated and implemented. Safety concerns with offsite shipment will require robust nuclear safety analysis, characterization, and packaging requirements.

Alternative A-4 is the most restrictive alternative with respect to administrative implementability. The potential for airborne contamination outside Building 235-F will present challenges for air permitting and monitoring. Radiological holdup will need to be removed prior to demolishing the outside structure, as the building itself provides confinement and mitigation to releases outside of the facility. All contamination will not be able to be removed, so the demolition process will require physical controls such as containment structures and will need to proceed as expeditiously as possible to prevent release during demolition due to adverse weather conditions. Therefore, phased implementation (using incremental funding) will not be practicable.

Permitting will be required for the demolition or removal of ACM. Qualified asbestos workers will be necessary to perform much of the demolition or removal activities. Other permitting requirements include a Stormwater Pollution Prevention Plan due to the large area of land disturbance.

The construction of the soil cover over the remaining building slab will present a technical challenge with respect to design interferences above and below grade. Several structures still remain in the area as shown on Figure 5 (e.g., Sand Filter [294-2F] and Sand Filter Fan House [292-2F], Central Alarm Station [720-F], Refrigeration Building No.1 [235-1F], Refrigeration Building No. 2 [235-2F], etc.) which will inhibit the lateral extension needed for the toe of the cap to allow proper drainage. Although many of the structures shown on Figure 5 will eventually undergo decommissioning, the schedule for such activities may not align with the Alternative A-4 soil cover implementation. Other underground commodities, such as the underground ventilation tunnel to the sand filter are not being addressed by this removal action. The tunnel provides an underground interference which may impact compaction requirements and future access to the tunnel for eventual disposition.

5.4 Cost

The cost categories considered for the removal action alternatives were capital and S&M, as appropriate. Capital costs were summarized as baseline/preparatory costs and construction costs. The baseline/preparatory costs include design, implementation plans, nuclear safety basis documentation, operational readiness reviews, etc. Many of these costs are common among Alternatives A-2 through A-4. Alternative A-1, No Action, has the highest cost because the Building 235-F deactivated end state (i.e., baseline condition) requires operation of the ventilation system and exhaust structures to maintain a negative pressure as well as S&M activities for 150 years. This time period is consistent with the estimated time of roof collapse if no maintenance was performed. At or before that time, additional response action(s) would be necessary to mitigate exposure/spread of contamination; however, the costs for the additional response action(s) is not included in the removal alternative estimate.

The cost estimates were developed based on a generic description and preconceptual design (SRNS 2021b) of activities for each alternative with limited engineering data. Where possible, elements of cost estimates previously prepared for similar alternatives were used and were scaled to be representative of 2021 dollars. The cost estimates provided in this document were developed for comparative evaluation only with an expected level of accuracy of -30% to +50%. Once the removal alternative is selected, a formal cost analysis will be performed based on preliminary design information.

The estimated cost, including total and direct indirect capital costs, as well as the present worth of direct and indirect cost of S&M for 150 years are summarized below for each removal alternative:

• Alternative A-1.	No Action	~$971M
• Alternative A-2.	ISD of First and Second Level Process Areas/Engineered Roof	~$92M
• Alternative A-3.	ISD of Entire Building/Engineered Roof	~$100M
• Alternative A-4.	Complete Building 235-F Removal/Soil Cover	~$201M

Detailed costs for each alternative are presented in Appendix D.

5.5 Comparison of Removal Action Alternatives

A comparative analysis of the removal action alternatives is presented Table 3. The criteria of effectiveness, implementability, and cost were ranked as low, moderate, or high for each alternative. A ranking of "low" indicates the least favorable score in comparison with the other alternatives, and a ranking of "high" indicates the most favorable score.

6.0 PREFERRED REMOVAL ACTION ALTERNATIVE

Based on the evaluation of removal action alternatives in this RSER/EE/CA, the lead agency believes Alternative A-2, ISD of First and Second Process Areas/Engineered Roof, provides the best balance of tradeoffs among the other removal action alternatives with respect to the evaluation criteria. Alternative A-2 meets the effectiveness criteria by grouting of the process areas and sealing the doors and penetrations along the exterior walls of Building 235-F to prevent exposure to radionuclide and hazardous contamination contained within the building. Entombing the contamination within the building and grouting the underground storage tank will prevent release of contamination. Constructing a sloped reinforced concrete roof over the Building 235-F will prevent rainwater infiltration and contaminant transport to groundwater.

Both short-term and long-term effectiveness can be achieved with Alternative A-2. Alternative A-2 also provides the most flexibility with respect to implementation because the process areas may be grouted in phases to allow funding resources to be stretched over multiple years. The estimated cost for Alternative A-2 is lower than the estimated costs for any of the other removal action alternatives evaluated.

Use of a NTC removal action is an effective strategy for cost-effectively reducing obvious risks and supporting area closure by initiating early actions while meeting necessary ARARs. This alternative will not preclude any additional remediation of the FAOU and is consistent with the current and future land use.

The waste streams generated as part of the preferred removal alternative will be transported to the appropriate disposal facilities. The contaminated waste anticipated to be generated includes concrete rubble and steel material associated with removal of the roof ducts, job control waste, personal protective equipment, and miscellaneous items. Radioactively contaminated waste will be characterized in accordance with USDOE requirements for disposal and will be sent to the SRS E-Area ~~Low-Level Waste Facility (~~LLWF~~)~~ where CERCLA Off-Site Acceptability is approved. Prior to the transfer of these wastes to their final disposal facility, SRS will obtain an acceptability determination from the appropriate Regional Off-site Rule Coordinator for disposal of CERCLA waste.

7.0 IMPLEMENTATION SCHEDULE

This RSER/EE/CA will be submitted to USEPA and SCDHEC for review and comment. The RSER/EE/CA will be available for public comment following this review. The removal action schedule is presented below.

Issue RSER/EE/CA for Public Comment	January 5, 2022
Submit Action Memorandum to USEPA and SCDHEC	March 7, 2022

8.0 REFERENCES

10 CFR 20. “Standards for Protection Against Radiation.” Energy. Environmental Protection Agency. *Code of Federal Regulations*

40 CFR 300.5. “Definitions.” Protection of Environment. Environmental Protection Agency. *Code of Federal Regulations*

40 CFR 300.150. “Worker health and safety.” Protection of Environment. Environmental Protection Agency. *Code of Federal Regulations*

40 CFR 300.405. “Discovery of notification.” Protection of Environment. Environmental Protection Agency. *Code of Federal Regulations*

40 CFR 300.415. “Removal Action.” Protection of Environment. Environmental Protection Agency. *Code of Federal Regulations*

DNFSB, 2012. “Recommendation 2012-1 to the Secretary of Energy, Savannah River Site Building 235-F Safety, Pursuant to 42 U.S.C. 2286 a(a)(5), Atomic Energy Act of 1954, as Amended”, May 9, 2012

FFA, 1993. *Federal Facility Agreement for the Savannah River Site,* Administrative Docket No. 89-05-FF (Effective Date: August 16, 1993)

SCDHEC, 1990. SCDHEC Settlement Agreement 90-64-SW, September 5, 1990, pgs. 103 and 130

SRNS, 2012a. *Post Construction Report for the P-Area Operable Unit (U)*, SRNS-RP-2011-01582 Appendix A, Revision 1, July 2012, Savannah River Nuclear Solutions, LLC, Savannah River Site, Aiken, SC

SRNS, 2012b. *Post Construction Report for the R-Area Operable Unit (U)*, SRNS-RP-2011-01574 Appendix A, Revision 1, June 2012, Savannah River Nuclear Solutions, LLC, Savannah River Site, Aiken, SC

SRNS, 2013. *Removal Action Report (RAR) for the In Situ Decommissioning of the 105-C Disassembly Basin*, SRNS-RP-2012-00837, Revision 1, May 2013, Savannah River Nuclear Solutions, LLC, Savannah River Site, Aiken, SC

SRNS, 2020. *Characterization Report for Building 235-F, F-Area Material Storage Building (FAMS)*, G-ESR-F-00097, Revision 0, May 2020, Savannah River Nuclear Solutions, LLC, Savannah River Site, Aiken, SC

SRNS, 2021a. *Scoping Summary for the F-Area Material Storage Building (235-F) (Removal Site Evaluation Report/Engineering Evaluation/Cost Analysis – Removal Alternative Analysis Scoping)*, SRNS-RP-2020-00444, Final, April 2021, Savannah River Nuclear Solutions, LLC, Savannah River Site, Aiken, SC

SRNS, 2021b. *Pre-conceptual Design Estimate for Building 235-F In-Situ Decommissioning Alternatives*, T-ESR-F-00033, Revision 0, January 2021, Savannah River Nuclear Solutions, LLC, Savannah River Site, Aiken, SC

USCB, 2019. QuickFacts Population Estimates, July 1, 2019 (V2019), United States Census Bureau, www.census.gov/quickfacts/fact/table (accessed April 22, 2021)

USDOE, 1996. *Savannah River Site Future Use Project Report*, Stakeholder-Preferred Recommendations for SRS Land and Facilities, United States Department of Energy, Savannah River Operations Office, Aiken, SC

USDOE, 2005. Savannah River Site's Cold War Building Environment Cultural Resources Management Plan (CRMP), Revision 5, United States Department of Energy, Savannah River Operations Office, Aiken, SC

USDOE, 2019a. Letter, B.T. Hennessey (USDOE) to S.B. Fulmer (SCDHEC) and J. Richards (USEPA), *Building 235-F Regulatory Information Meeting (October 30, 2019)*, SEMS Number:88, IACD-20-113, dated November 15, 2019

USDOE, 2019b. NMPD-19-0024, *Savannah River Management and Operating Contract DE-AC09-08R22470, Direction Regarding Building 235-F Activity, July 10, 2019*

USDOE, 2020. *Final Environmental Impact Statement for Plutonium Pit Production at the Savannah River Site in South Carolina*, DOE/EIS-0541, USDOE National Nuclear Security Administration, Savannah River Site, Aiken, SC, September 2020

USEPA, 1999. *Consideration of Cumulative Impacts in EPA Review of NEPA Documents, US*EPA 315-R-99-002., United States Environmental Protection Agency, May 1999

WSRC, 1999. *Land Use Control Assurance Plan for the Savannah River Site*, WSRC-RP-98-4125, Rev. 1.1, August 1999, latest update, Savannah River Nuclear Solutions, LLC, Savannah River Site, Aiken, SC

9.0 GLOSSARY

Applicable or Relevant and Appropriate Requirement: The Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) requires compliance with any promulgated standard requirements, criteria, or limitation under Federal and more stringent State environmental laws. Examples include the Clean Water Act, Endangered Species Act, etc.

Comprehensive Environmental Response, Compensation and Liability Act: A Federal law, known as Superfund, passed in 1980 and reauthorized by the Superfund Amendments and Reauthorization Act (SARA) in 1986. The law authorizes the Federal government to respond directly to releases of hazardous substances that may endanger public health or the environment.

Curie: A unit of radioactivity that represents the amount of radioactivity associated with one gram of radium. To say that a sample of radioactive material exhibits one curie of radioactivity means that the element is disintegrating at the rate of 37 billion times per second.

Deactivation: The process of placing a facility in a stable and known condition including the removal of hazardous and radioactive materials to ensure adequate protection of the worker, public health and safety, and the environment, thereby limiting the long-term cost of S&M. Actions include the removal of fuel, draining and/or de-energizing nonessential systems, removal of stored radioactive and hazardous materials, and related actions. Deactivation does not include all decontamination necessary for the dismantlement and demolition phase of decommissioning, e.g., removal of contamination remaining in the fixed structures and equipment after deactivation.

Decommissioning: The final process of closing and securing a nuclear, radiologically contaminated, or radioactive material storage facility consistent with the established end state that provides adequate protection from radiation exposure and to isolate radioactive contamination from the human environment.

Decontamination: The removal or reduction of residual radioactive and hazardous materials by mechanical, chemical or other techniques to achieve a stated objective or end condition.

Low-Level Waste: Low-level radioactive waste is defined as any radioactive waste that does not belong in one of the following three categories for radioactive waste: high-level waste (spent nuclear fuel or the highly radioactive waste produced if spent fuel is reprocessed), uranium milling residues, and waste with greater than specified quantities of elements heavier than uranium. Low-level radioactive waste is generated at commercial facilities such as nuclear power plants, hospitals, and research institutions. It includes radioactive materials used in various processes as well as supplies and equipment that have been contaminated with radioactive materials.

Low-Level Waste Disposal Site: Low-level waste disposal occurs at facilities that are designed, constructed, and operated to meet safety standards. The operator of the facility must also extensively characterize the site on which the facility is located and analyze how the facility will perform for thousands of years into the future. Low-level waste from Building 235-F decommissioning will be characterized in accordance with USDOE requirements for disposal and will be sent to the SRS E-Area LLWF.

Material At Risk: The amount of hazardous material (curies or grams) used to calculate the source term for a release due to an accident or event.

National Oil and Hazardous Substances Pollution Contingency Plan: The federal government's blueprint for responding to both oil spills and hazardous substance releases. The NCP is the result of our country's efforts to develop a national response capability and promote overall coordination among the hierarchy of responders and contingency plans.

Non-Time Critical Removal Action: This is a type of response action recognized by the USEPA as appropriate for addressing hazardous substance threats where a planning horizon of six months or more is appropriate. Under an USEPA/USDOE agreement, USDOE uses a non-time critical removal action approach tailored for decommissioning USDOE facilities. That approach is comprised of a threat assessment; identification, analysis, and documentation of decommissioning alternatives; opportunities for public participation in the decommissioning decision; and planning and performance of decommissioning activities.

Principal Threat Source Material: Wastes are those source materials considered to be highly toxic or highly mobile and generally cannot be reliably contained or would present a significant risk to human health or the environment should exposure occur. If a contaminant poses a probability of more than 1 excess case of cancer in 1,000 (risk=1.0E-3) it is generally considered PTSM.

Radiological Holdup: A term used to describe process residues, consisting primarily of Pu-238 and Np-237 oxides, contained within the enclosures and ventilation exhaust system within Building 235-F. Radiological holdup is also referred to a residual radiological contamination in this RSER/EE/CA document.

Removal Action: When USDOE identifies a threat of exposure to, or migration of, hazardous substances that poses a risk to health, welfare, or the environment, USDOE is authorized by CERCLA to exercise removal action authority to implement an appropriate response to the risks posed. Activities that may be taken under CERCLA removal action authority include any activity that reduces risks or potential risks in a relatively short time frame and can be identified as appropriate with a relatively limited analysis of alternatives. Removal actions are not limited to immediate action, or action in response to an emergency. (See non-time critical removal action.)

Surveillance and Maintenance: These activities are conducted through-out the facility life cycle phase including when a facility is not operating and is not expected to operate again and continues until phased out during decommissioning. Activities include providing in a cost-effective manner periodic inspections and maintenance of structures, systems and equipment necessary for the satisfactory containment of contamination and protection of workers, the public, and the environment.

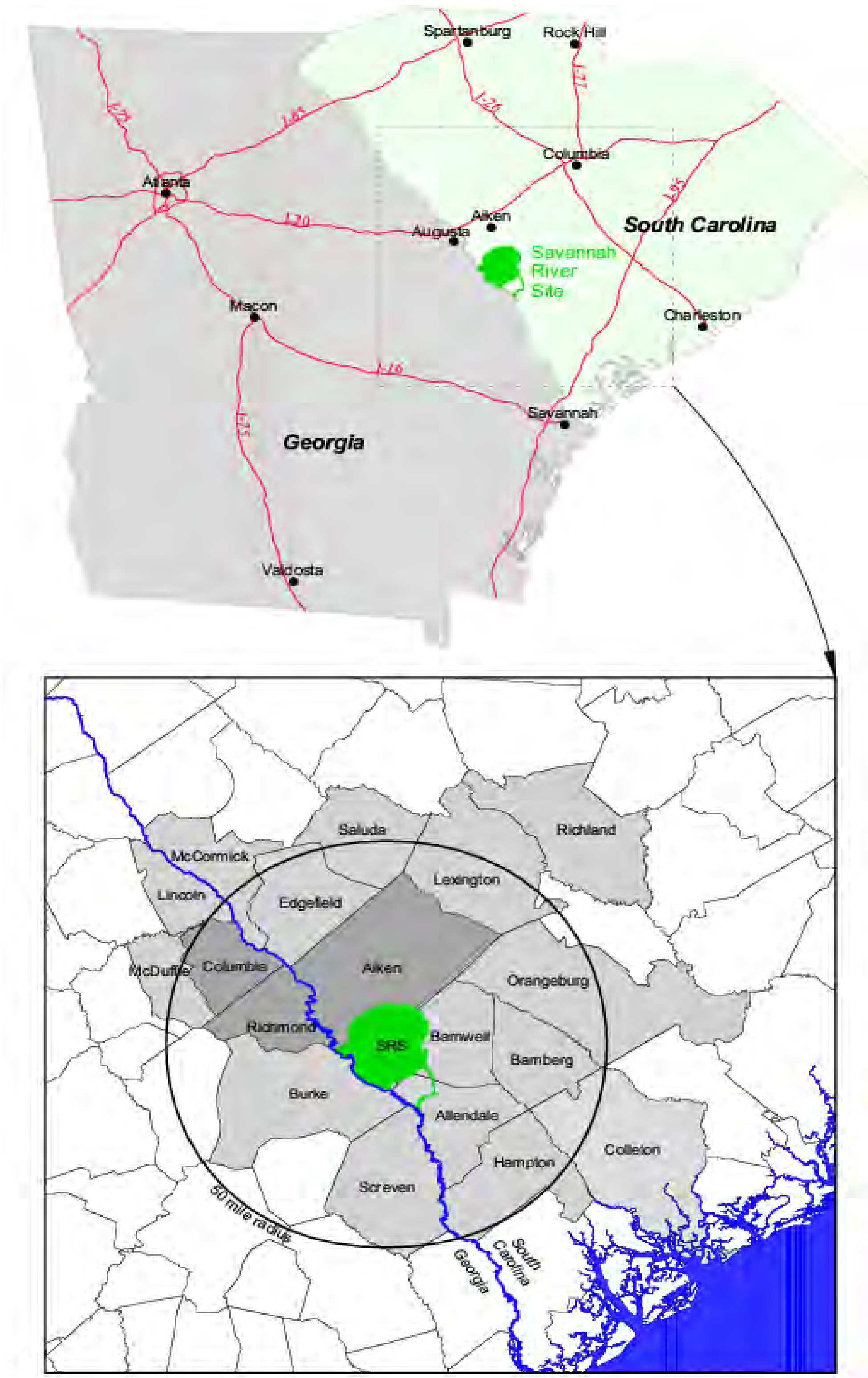


Figure 1. Geographic Proximity of the Savannah River Site

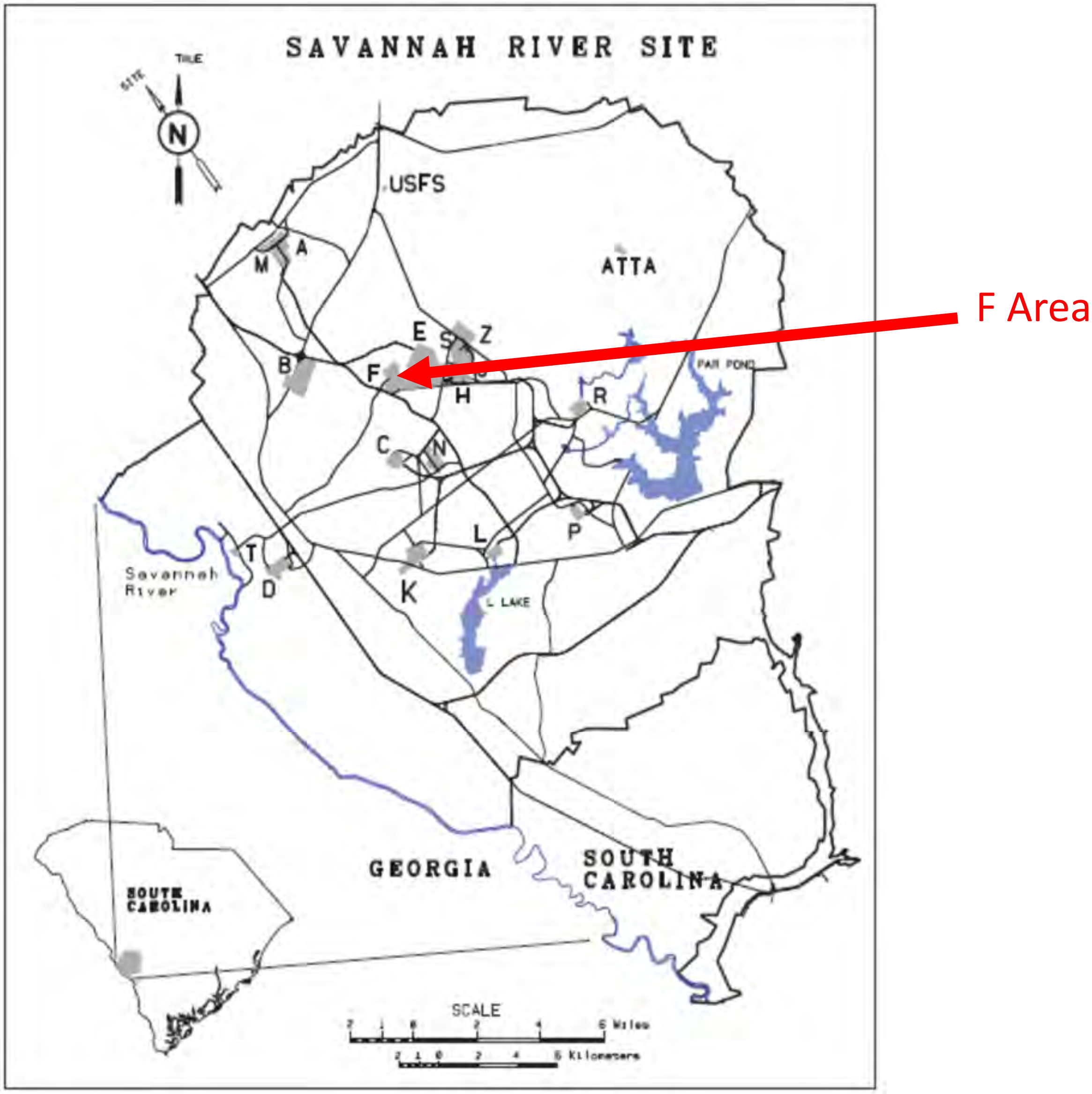


Figure 2. Location of F Area at Savannah River Site

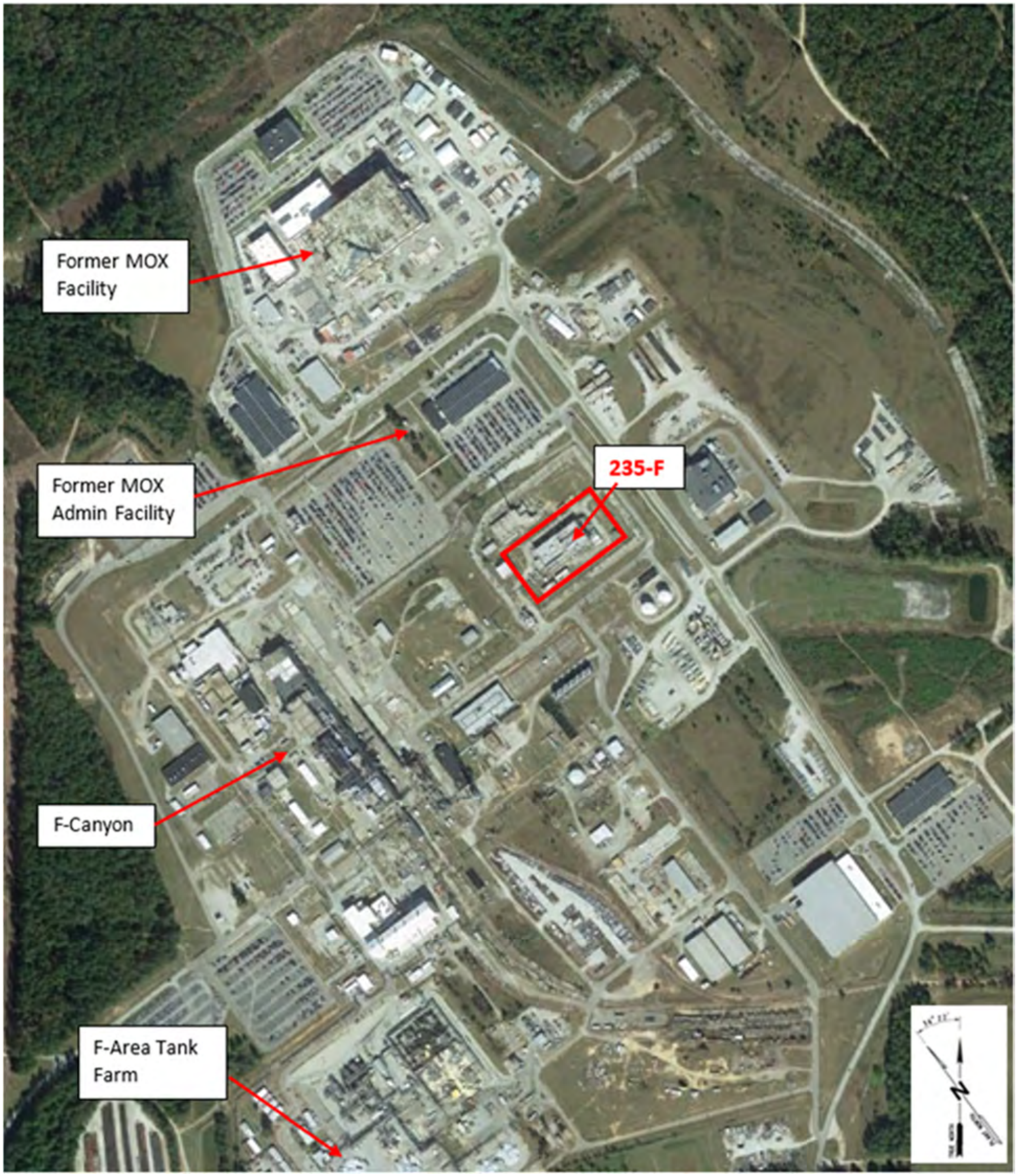


Figure 3. Aerial View of Building 235-F Location in F Area

This page was intentionally left blank.

Figure 4. Historical Photograph of Building 235-F (Prior to 293-F Stack Reduction)

Building Number	Facility
235-F*	Material Storage Building
235-12F	Storage Pad for B-25 Waste Boxes
235-17F	Emergency Storage Shed
235-1F*	Refrigeration Building No. 1
235-2F*	Refrigeration Building No. 2
235-11F	Office Trailer
245-6F	Office Trailer
252-10F	Secondary Transformer Station, for 292-F
252-11F	Secondary Transformer, 235-11F
252-20F	Substation for Building 235-F
252-9F	Secondary Transformer Station for 720-F
253-8F	15kV Outdoor Padmount Switchgear
281-28F	Spill Basin
285-11F	Induced Draft Cooling Tower
291-2F	Sand Filter Fan House Stack
292-2F*	Sand Filter Fan House
293-F*	Metallurgical Building Stack
294-2F*	Sand Filter for 235-F
607-3F	Sewage Lift Station
701-4F*	Gatehouse Entrance to 235-F
720-F*	Central Alarm Station (CAS)

* D&D Facilities on FFA Appendix K

Figure 5. Building 235-F and Surrounding Buildings and Structures

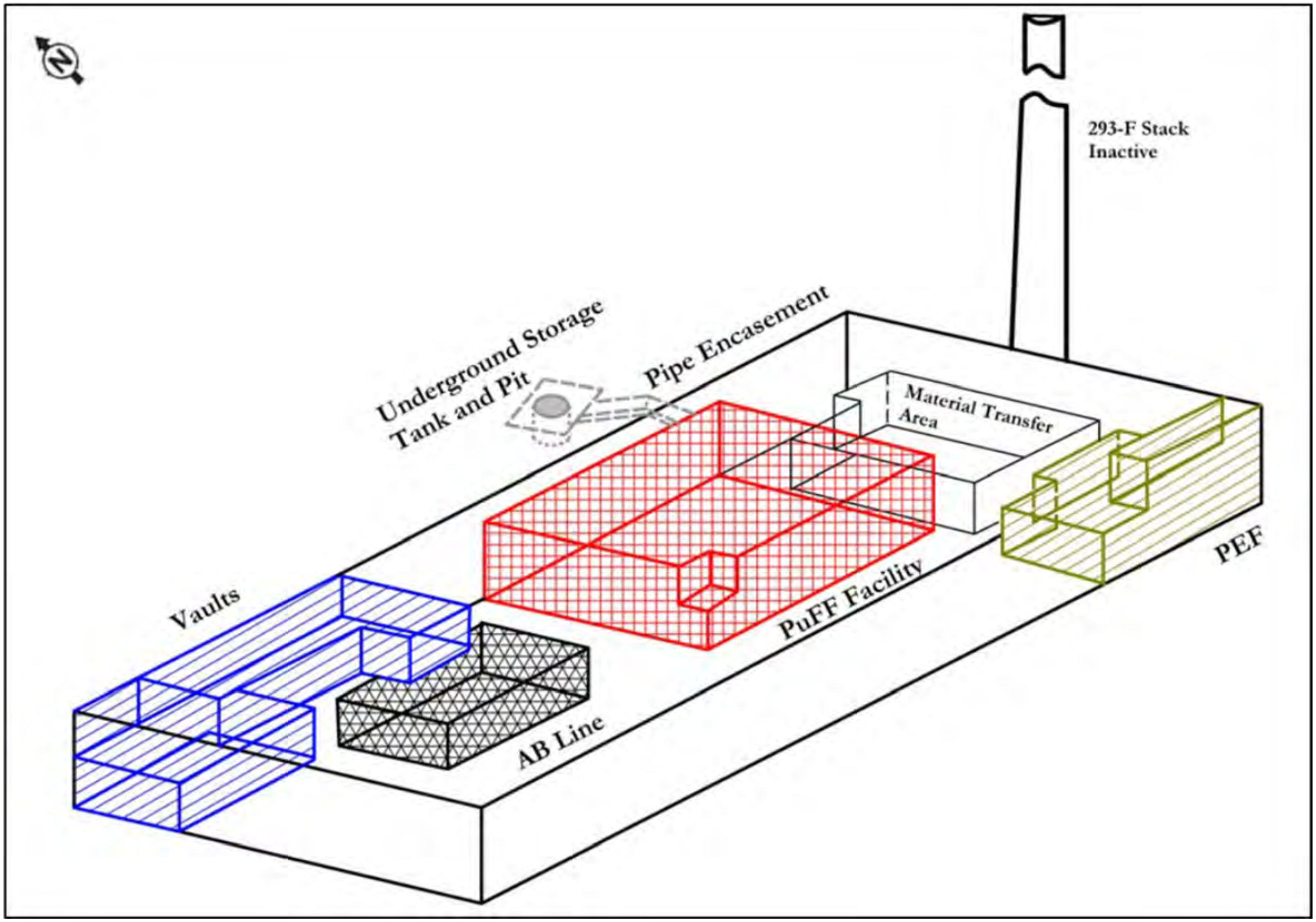


Figure 6. Conceptual View of Building 235-F First and Second Level Process Areas

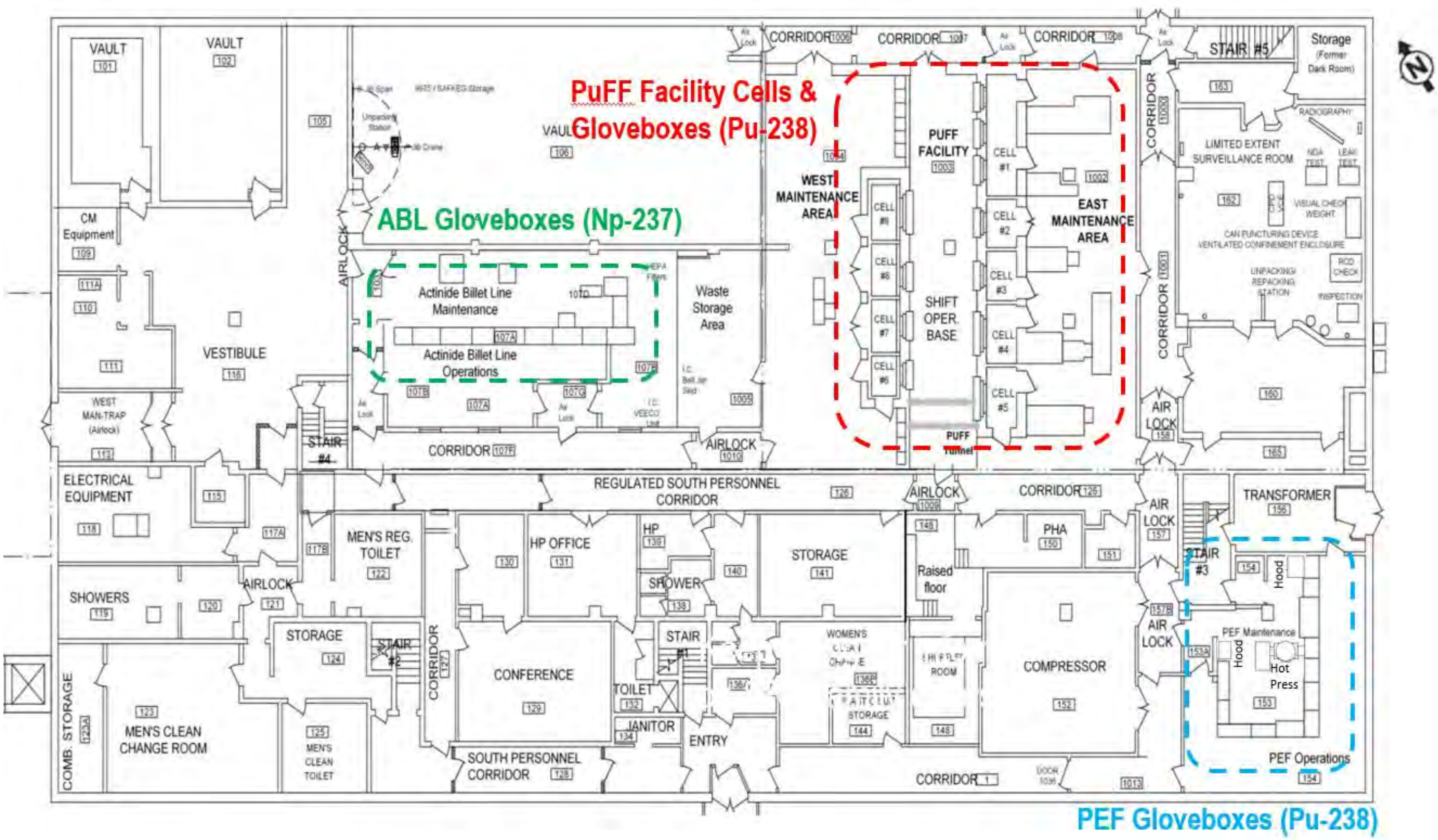


Figure 7. Layout of Building 235-F First Level

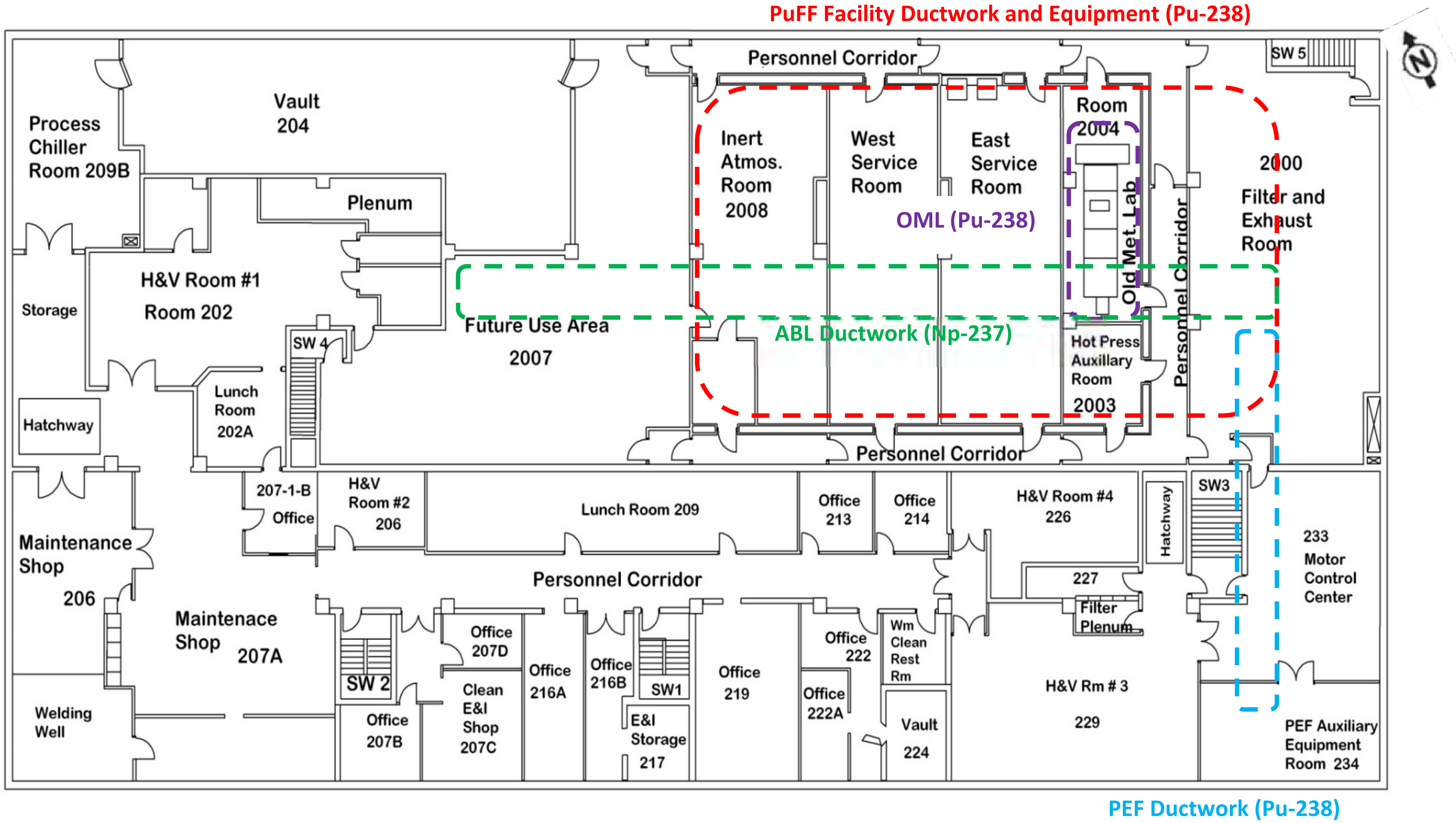


Figure 8. Layout of Building 235-F Second Level

Figure 9. Photograph of Height Reduced Abandoned Capped Stack (293-F)

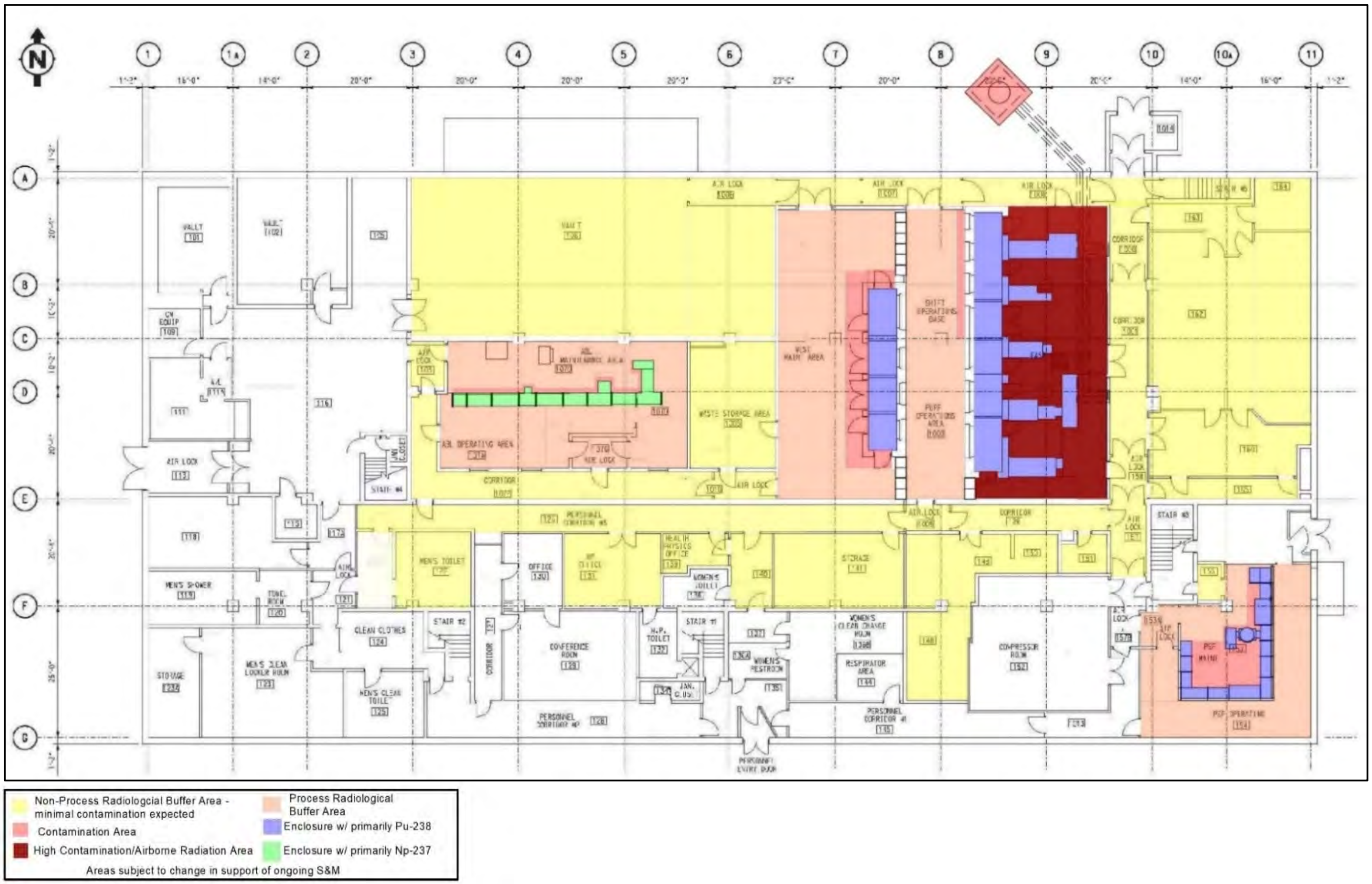


Figure 10. Building 235-F First Level Contamination (Radiological Holdup)

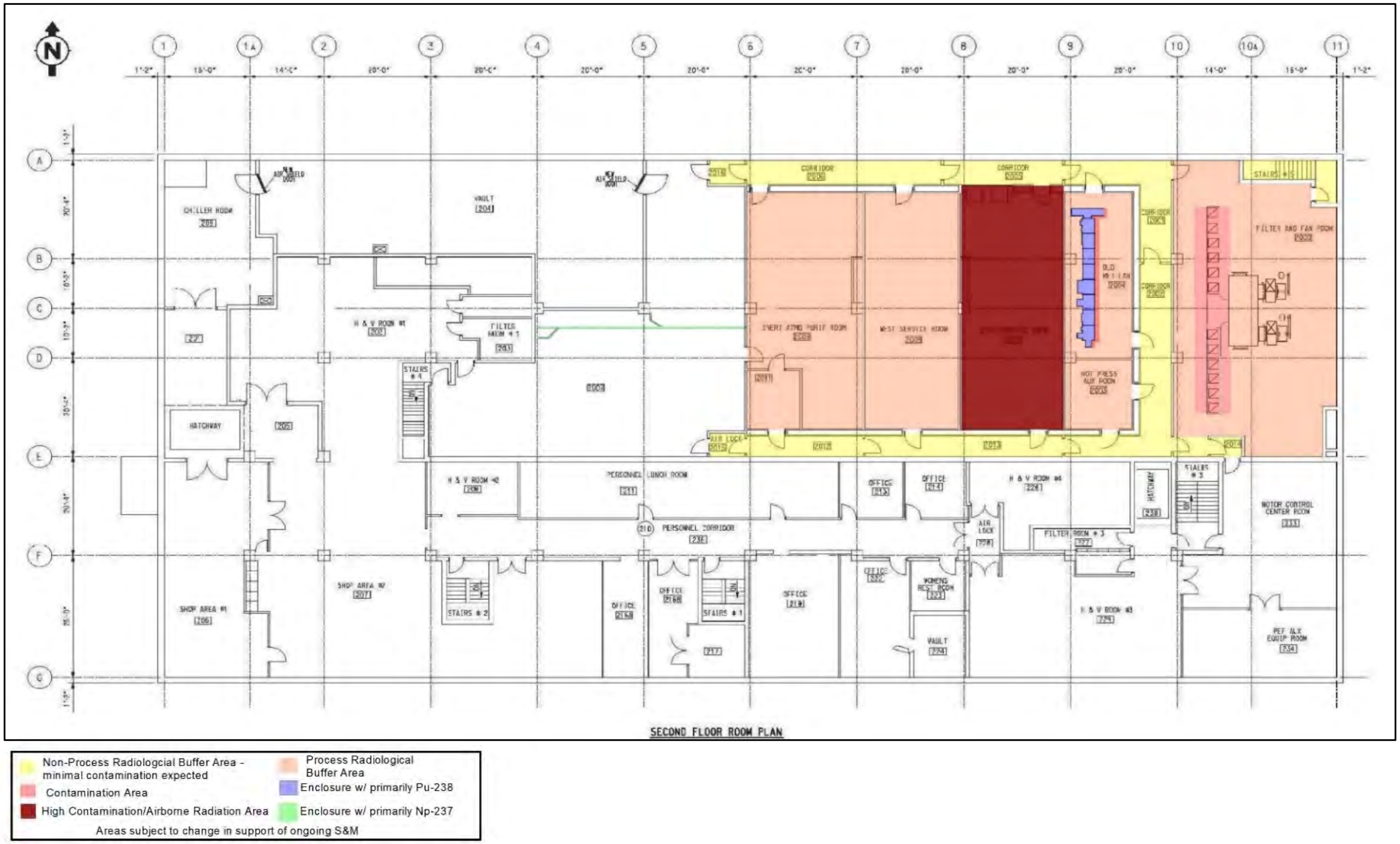


Figure 11. Building 235-F Second Level Contamination (Radiological Holdup)

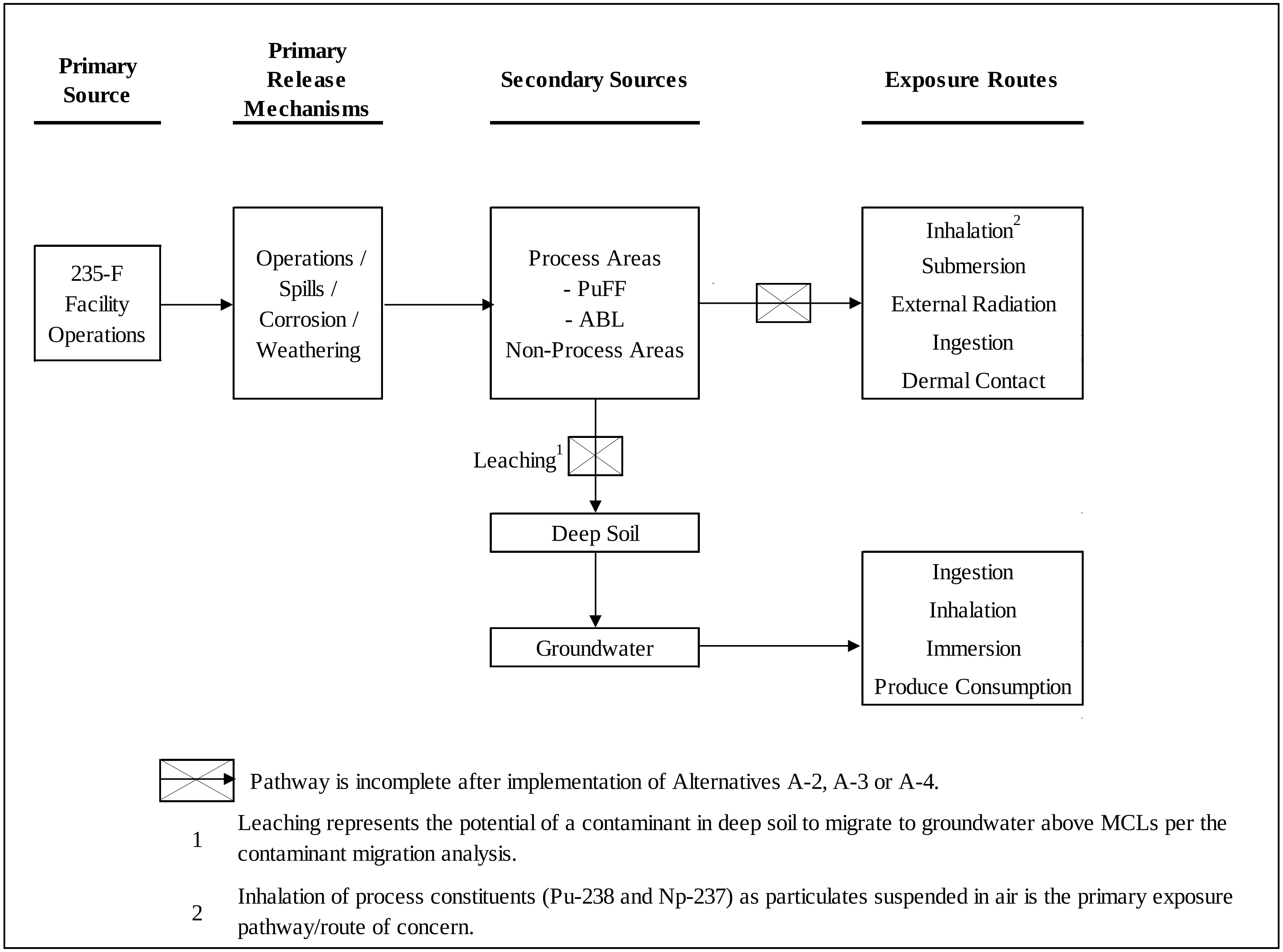


Figure 12. Conceptual Site Model of Building 235-F

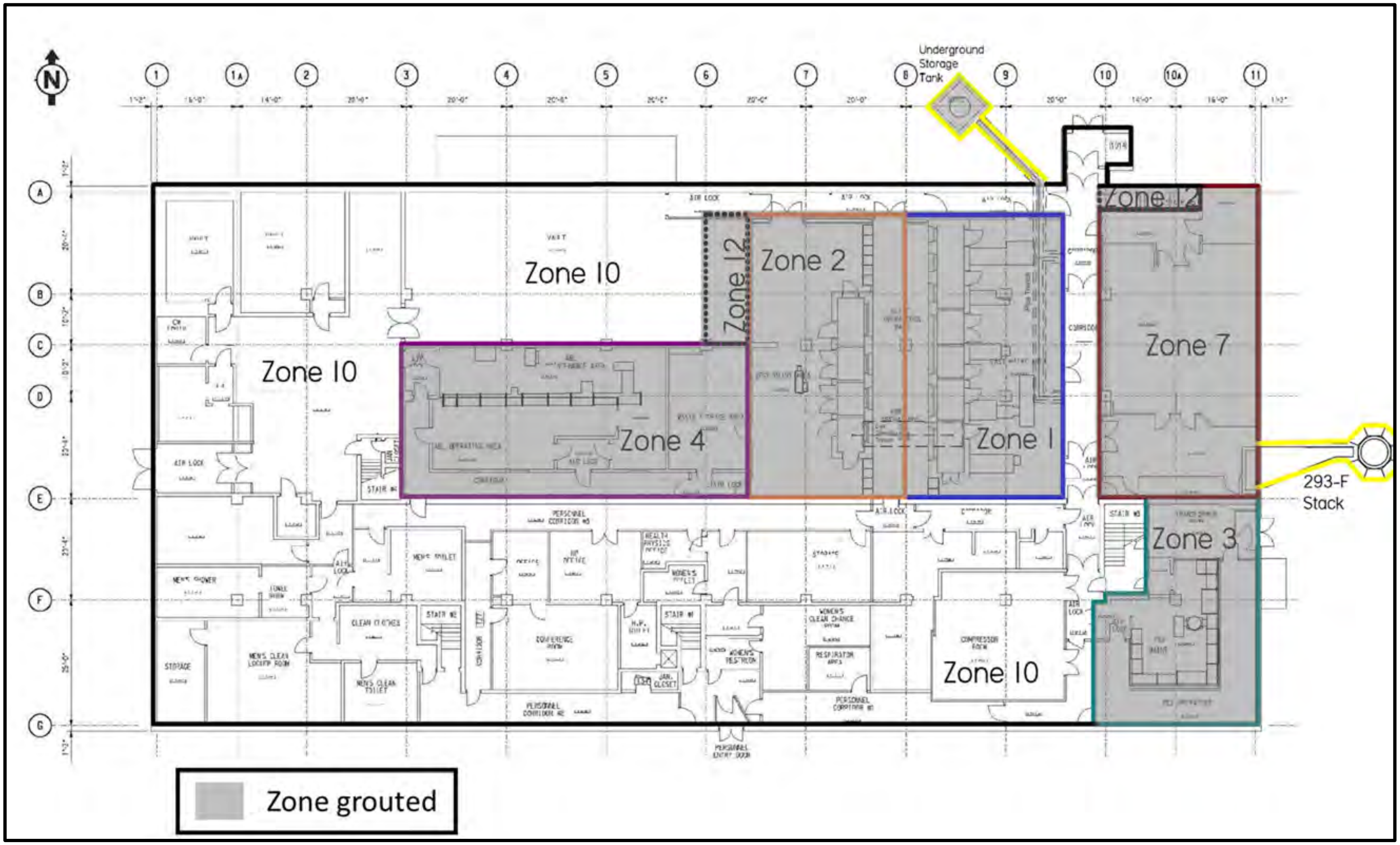


Figure 13. Alternative A-2 – ISD of First and Second Level Process Areas /Engineered Roof (First Level View)

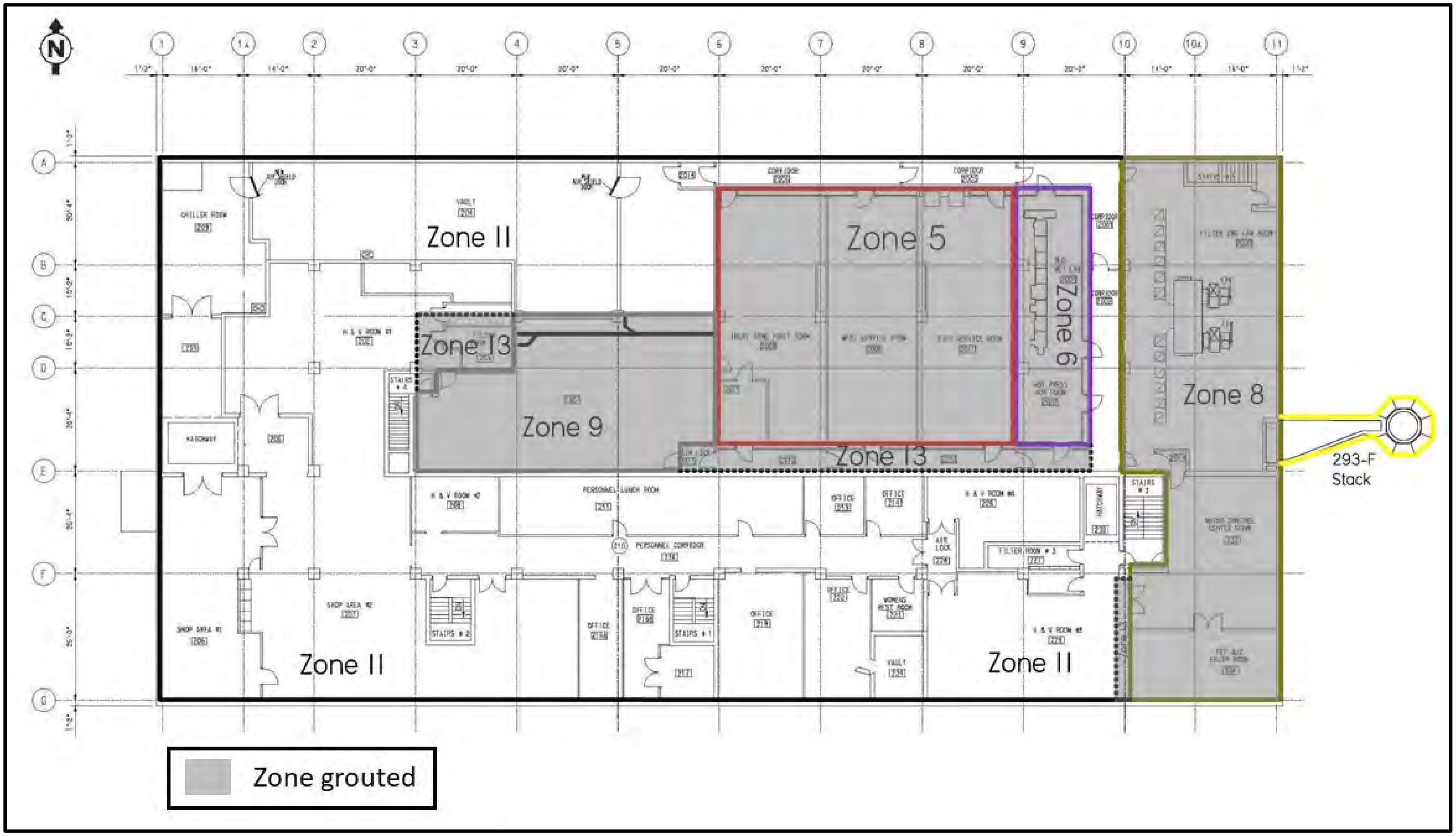


Figure 14. Alternative A-2 – ISD of First and Second Level Process Areas /Engineered Roof (Second Level View)

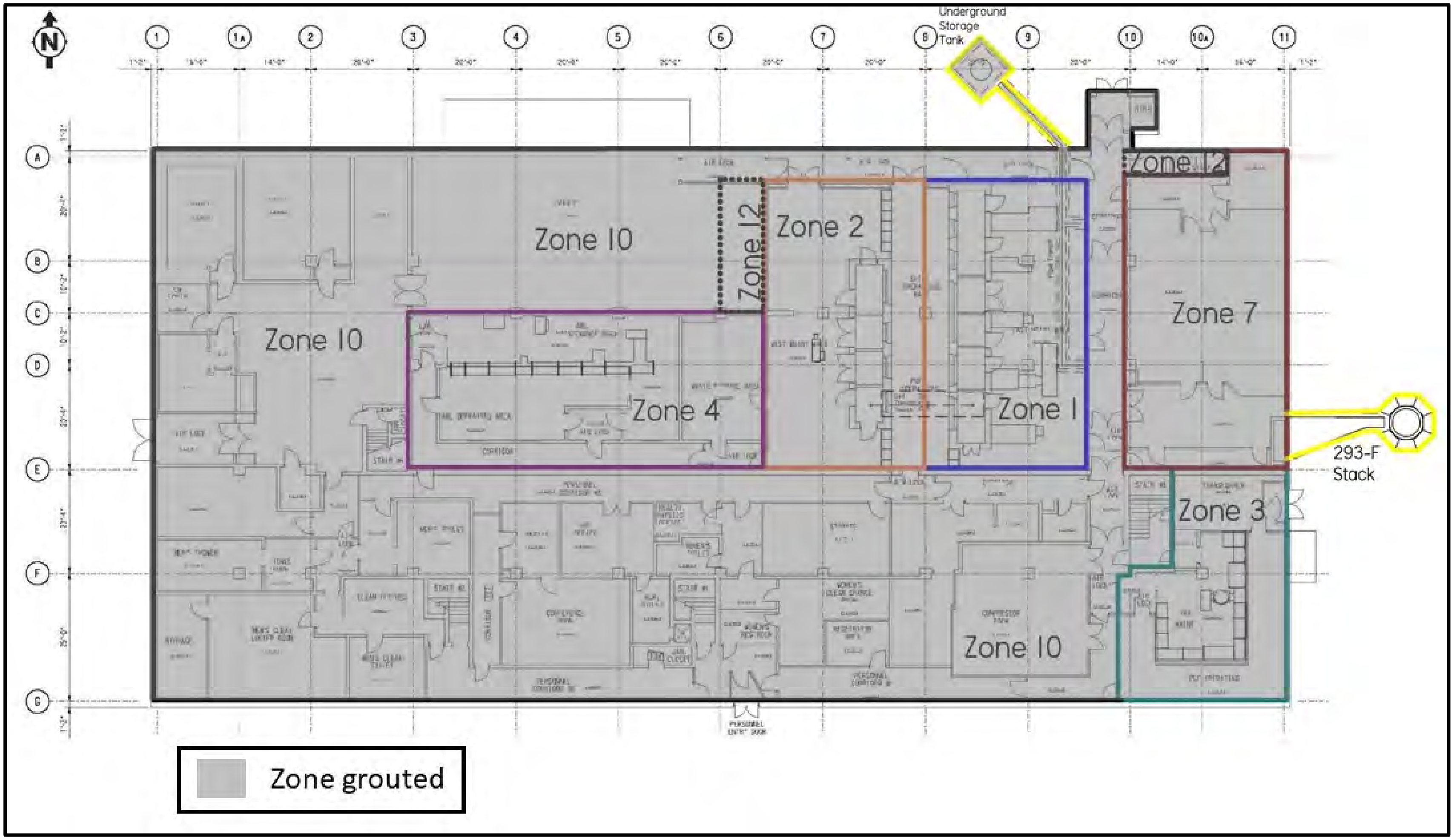


Figure 15. Alternative A-3 – ISD of Entire Building 235-F /Engineered Roof (First Level View)

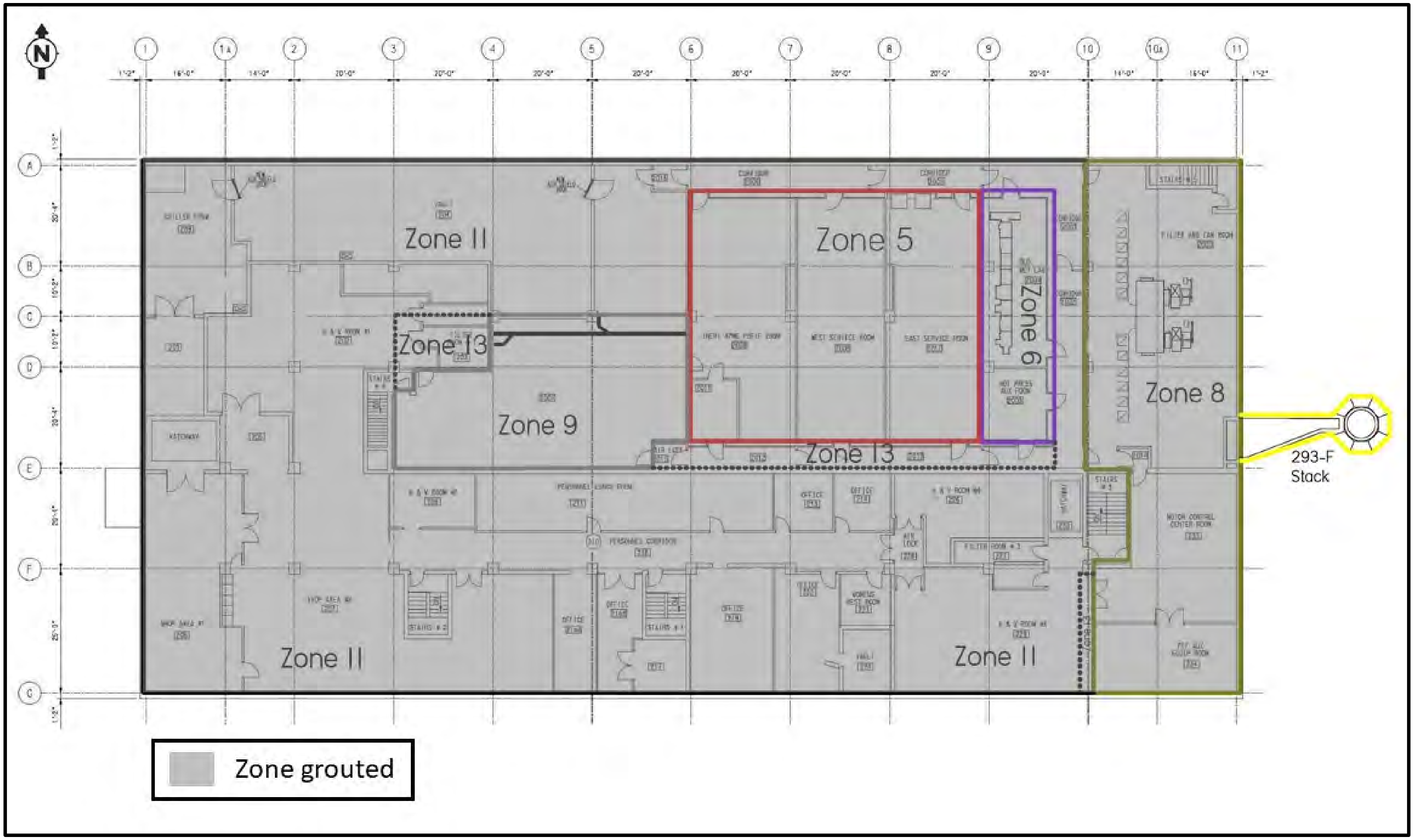


Figure 16. Alternative A-3 – ISD of Entire Building 235-F /Engineered Roof (Second Level View)

This page was intentionally left blank.

Table 1. Building 235-F Total Radiological Holdup

Location	Pu-238 grams (g) [1]	Percent of Pu-238	Np-237 (g) [1]	Percent of Np-237
PuFF Facility Cell 1	[redacted]	57%	-	-
PuFF Facility Cell 2	[redacted]	17%	-	-
PuFF Facility Cell 3	[redacted]	1%	-	-
PuFF Facility Cell 4	[redacted]	7%	-	-
PuFF Facility Cell 5	[redacted]	2%	-	-
PuFF Facility **Cell 1-5**	[redacted]	**85%**	-	-
PuFF Facility Cells 6-9	[redacted]	0%	-	-
PuFF Facility Exhaust Ductwork	[redacted]	3%	-	-
PuFF Facility Inert Atmosphere Line	[redacted]	4%	-	-
PEF	[redacted]	3%	[redacted]	0%
ABL	[redacted]	**1%**	[redacted]	**87%**
ABL E1 Exhaust	[redacted]	0%	[redacted]	12%
OML	[redacted]	4%	[redacted]	0%
Rm 2000 E1 Exhaust	[redacted]	0%	[redacted]	1%
Rest of Building[2]	[redacted]	**14%**	[redacted]	**13%**
Entire Building[3]	[redacted]	**-**	[redacted]	-

[1] Inventory from *Characterization Report for Building 235-F, F-Area Material Storage Building (FAMS)*, G-ESR-F-00097, Revision 0, May 2020, Table 12.
[2] Rest of Building refers to radiological holdup exclusive of PuFF Facility Cells 1-5 and ABL.
[3] Entire Building is the total building inventory for Pu-238 and Np-237 from all locations.

This page was intentionally left blank.

Table 2. Description of NTC Removal Action Alternatives for Building 235-F

Building 235-F Final Decommissioning End State	A-1. No Action	A-2. ISD of First and Second Level Process Areas / Engineered Roof	A-3. ISD of Entire Building 235-F / Engineered Roof	A-4. Complete Building 235-F Removal / Soil Cover
Removal Action End State Description	▪ Building 235-F, underground storage tank and pipe trench, and abandoned capped stack (293-F) remain as is following deactivation activities (i.e., building deactivation end state represents baseline conditions). ▪ Radiological holdup and contaminated equipment remain in place. ▪ Hazardous material (e.g., lead shielding, leaded glass, lead based paints, PCB-based paints, and non-friable ACM) remains in place. ________________________ ▪ Building O&M required after deactivation. - Ventilation systems operate to maintain a vacuum within process enclosures using unconditioned air. - 294-2F Sand filter (294-2F) and exhaust stack (291-2F) remain in operation. - Assume roof collapse at 150 years, at which time additional response action will likely be necessary to mitigate exposure/spread of contaminants.	▪ Below grade transfer trenches grouted. ▪ Fixative applied on contaminated surfaces. ▪ Where possible, ventilation ducts relocated to floor prior to grouting or breached to allow grout to fill. ▪ Recombiners added to process enclosures to prevent hydrogen build up. ▪ Process enclosures sealed and formed prior to grouting. ▪ Zones 1-9 and 12-13 are formed and grouted. ▪ Structural supports added to non-grouted rooms to support reinforced concrete roof addition. ▪ Roof structures (e.g., concrete ventilation ducts) removed as necessary, size reduced and disposed of. ▪ All exterior penetrations and doors are sealed. ▪ Engineered roof installed. ▪ Underground storage tank grouted/capped, and pipe trench grouted. ▪ Abandoned capped stack (293-F) permanently sealed.	▪ Below grade transfer trenches grouted. ▪ Fixative applied on contaminated surfaces. ▪ Where possible, ventilation ducts relocated to floor prior to grouting or breached to allow grout to fill. ▪ Recombiners added to process enclosures to prevent hydrogen build up. ▪ Process enclosures sealed and formed prior to grouting. ▪ All zones on first level are grouted (after actions are taken to ensure unimpeded flow) with near simultaneous lifts throughout. ▪ All zones on second level are grouted (after actions are taken to ensure unimpeded flow) with near simultaneous lifts throughout. ▪ Roof structures (e.g. concrete ventilation ducts) removed as necessary, size reduced and disposed of. ▪ All exterior penetrations and doors are sealed. ▪ Engineered roof installed. ▪ Underground storage tank grouted/capped, and pipe trench grouted. ▪ Abandoned capped stack (293-F) permanently sealed.	▪ Below grade transfer trenches grouted. ▪ Fixative applied on contaminated surfaces. ▪ Radiological holdup from building removed, packaged, and transported off-site to a TRU waste disposal facility. ▪ Radiologically contaminated equipment/ventilation ducts removed, packaged, and transported off-site to a TRU waste disposal facility. ▪ Hazardous material and low-level radionuclide material removed and transported to appropriate disposal facility. ▪ Building demolished to slab and rubble disposed of as LLW. ▪ Abandoned capped stack (293-F) removed. ▪ Underground storage tank grouted. ▪ Soil cover constructed over slab, tank, and former abandoned capped stack (293-F) area.

This page was intentionally left blank.

Table 3. Comparative Analysis of the NTC Removal Action Alternatives for Building 235-F

	A-1. **No Action**	**A-2.** **ISD of First and Second Level Process Areas / Engineered Roof**	**A-3.** **ISD of Entire Building 235-F / Engineered Roof**	**A-4.** **Complete Building 235-F Removal / Soil Cover**
Effectiveness	**Low** **Pros:** • Short-term effectiveness to the SRS worker is adequate while S&M activities and ventilation of enclosures continue. **Cons:** • Does not achieve RAO to prevent exposure to hypothetical future industrial worker. Unmitigated risk (2.3E+9) is extremely high for hypothetical future industrial worker scenario. • Does not achieve RAO to prevent migration of radionuclide contamination to groundwater that result in exceedances of MCLs. • Risk to co-located SRS workers impacts SRS missions. • Radiological holdup and hazardous substances remain in place with vulnerability to release increasing over time as the facility deteriorates.	**High** **Pros:** • Achieves RAO by preventing hypothetical future industrial worker exposure to contaminants. • Achieves RAO to prevent migration of contaminants to groundwater above MCLs at the POA. • Short-term effectiveness with respect to decommissioning worker achieved by minimizing entry into the cells/process enclosures. • Formwork mitigates grout movement and reduces the potential to spread contamination via flowing grout. **Cons:** • Increased decommissioning worker exposure during placement of formwork reduces short-term effectiveness. • Hazardous substances and PTSM remain entombed on site. • No unrestricted use.	**High** **Pros:** • Achieves RAO by preventing hypothetical future industrial worker exposure to contaminants. • Achieves RAO to prevent migration of contaminants to groundwater above MCLs at the POA. • Short-term effectiveness with respect to decommissioning worker achieved by minimizing entry into the cells/process enclosures. • Potential reduction in required formwork reduces decommissioning worker exposure during implementation. **Cons:** • Grouting rooms simultaneously increases potential for the spread of contamination within Building 235-F. • Hazardous substances and PTSM remain entombed on site. • No unrestricted use.	**Moderate** **Pros:** • Achieves RAO by preventing hypothetical future industrial worker exposure to contaminants. • Achieves RAO to prevent migration of contaminants to groundwater above MCLs at the POA. • A soil cover will prevent access and release of below grade contamination. **Cons**: • Short-term effectiveness is low due to the risk to the decommissioning worker and the potential to release airborne contamination during removal/demolition. • ACM removal increases risk of exposure to decommissioning workers. • Some contamination will remain entombed in grout in the below grade portions located under the removed cells. • No unrestricted use.
Implementability	**Moderate** **Pros:** Technically and administratively feasible. **Cons:** • Due to the amount of radiological holdup and Nuclear Safety concerns S&M is required for Building 235-F, as well as operation of the E5 fan exhaust ventilation system and sand filter (294-2F) and exhaust stack (291-2F). • Eventually, conditions within the facility will deteriorate making S&M activities hazardous and requiring response action to mitigate exposure/spread of contamination.	**High** **Pros:** • Technically and administratively feasible as exemplified by other grouting projects at SRS. • Material resources are widely available. • Funding resources could be stretched over multiple years, allowing implementation in phases with the highest contaminated areas to be grouted first. **Cons:** • Includes significant amount of formwork to isolate process areas. • Requires structural supports in non-grouted rooms to support roof loading. • Roof demolition and construction present industrial hazards.	**Moderate** **Pros:** • Technically and administratively feasible as exemplified by other grouting projects at SRS. • Material resources are widely available. • Funding resources could be stretched over multiple years, allowing implementation in phases with the first level to be grouted in one year and second level in another. • Structural supports not required for roof loading. **Cons:** • Roof demolition and construction present industrial hazards. • Anchoring plan required. • Managing slick lines during simultaneous lifts is more difficult than Alternative A-2. • Commercial concrete plant production and truck availability/traffic logistics may not be able to meet the demands for simultaneous lifts. • Phased implementation is more restrictive than Alternative A-2.	**Low** **Pros:** None. **Cons:** • Significant technical challenges exist associated with deactivating and removing radiological holdup/process enclosures. • Robotic resources may be required due to significant worker exposure to radionuclides and the significant risk for accidental exposure. • Permitting will be required for the removal of ACM. • Minimal SRS experience with removal of high activity/low particle size contamination may result in airborne release. • Project completion is required as expeditiously as possible, requiring full funding resources at beginning of removal action implementation since phased implementation is not practicable. • Waste handling will require nuclear material packaging and transportation to offsite disposal for TRU waste. • Building demolition and construction present industrial hazards. • Above and below grade interferences with soil cover footprint.
Cost	**$ 971M** (S&M for 150 years.)	**$ 92M**	**$ 100M**	**$ 201M**

This page was intentionally left blank.

APPENDIX A

Human Health Risk Assessment for Building 235-F

This page was intentionally left blank.

A.1 Introduction

A streamlined human health risk assessment (HHRA) for Building 235-F, F-Area Material Storage (FAMS) Building is presented in this appendix. The purpose is to evaluate the potential for adverse human health effects associated with exposure to the primary process contaminants in Building 235-F. The assessment estimates the risk potential in the absence of a response action and provides a basis for determining whether an action is necessary. This evaluation follows the technical approach that is recommended in United States Environmental Protection Agency (USEPA) guidance documents with respect to risk assessment for response actions at Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) sites.

This HHRA supports the evaluation of decommissioning removal action alternatives for Building 235-F including in-situ decommissioning (ISD) and source removal action alternatives. No current or projected future development of the Building 235-F complex is planned, and future land use is expected to remain industrial.

Extensive assays of Building 235-F have been performed that indicate significant radiological contamination (oxides of plutonium-238 [Pu-238] and neptunium-237 [Np-237])remains within the facility. It should be recognized that other radiological contaminants are also present within the facility due to process impurities and radioactive decay (i.e., daughter products). In addition, hazardous materials including asbestos, lead used for shielding, and polychlorinated biphenyls (PCBs) in paint and joint compound are expected to remain in the building following deactivation activities. Because risk from exposure to the radioactive impurities/decay products and hazardous materials is well bounded by the risk from exposure to Pu-238 and Np-237 in the process areas, this streamlined risk evaluation focuses primarily on the Pu-238 and Np-237, the risk drivers, since they provide sufficient information to justify a response action under CERCLA.

The contaminant migration analysis (fate and transport modeling) in Appendix B considers the building inventories of radionuclides (including daughter products and impurities), lead, and PCBs to evaluate contaminant migration through the vadose zone. Although a viable route of human exposure to hazardous materials (i.e., PCBs in paint and lead in building components) is unlikely, a qualitative risk discussion of hazardous constituents is provided to demonstrate the negligible risk contribution from the hazardous inventories when compared to the primary risk drivers.

A.2 Primary Risk Drivers: Pu-238 and Np-237

External exposure to plutonium poses very little health risk, since plutonium isotopes emit alpha radiation and almost no beta or gamma radiation. In contrast, internal exposure to plutonium, via the inhalation pathway, poses an extreme health hazard. It generally stays in the body for decades, exposing organs and tissues to radiation and increasing the risk of cancer. Depending on its particle size and how well the particular chemical form dissolves, plutonium can remain in the lungs when inhaled. Pu-238 decays quickly, releasing energy as alpha particles, and has an 87.7-year half-life during which it decays to uranium-234 (U-234), which has a 244,000-year half-life.

Neptunium is a health hazard if it is taken into the body, but there is also an external risk associated with the gamma rays emitted by Np-237 and its short-lived decay product, protactinium-233 (Pa-233). The primary means of exposure to Np-237 at Building 235-F is inhalation of contaminated airborne dust. Neptunium is taken up in the body much more readily if inhaled rather than ingested. The major health concern is cancer resulting from the ionizing radiation emitted by neptunium isotopes deposited on bone surface and in the liver. Np-237 decays very slowly and has a 2,140,000-year half-life.

A.3 Data

This HHRA uses data from Table 12, "Building 235-F Total Holdup" in the *Characterization Report for Building 235-F, F-Area Material Storage Building (FAMS*) (SRNS 2020) (i.e., Table 1 of this RSER/EE/CA report). The inventories (grams [g]) of Pu-238 and Np-237 are provided for various portions of the facility. Three inventory divisions are evaluated in this assessment: 1) Entire Building 235-F; 2) Plutonium Fuel Form (PuFF) Process Cells 1-5 Total; and 3) Actinide Billet Line (ABL).

The PuFF Process Cells 1-5 were chosen for evaluation since they contain the majority [redacted] g, [85%]) of Pu-238 within the facility, and the ABL was chosen for evaluation since it contains the majority ([redacted] g, [87%]) of Np-237 within the facility. The Entire Building 235-F contains [redacted] g of Pu-238 and [redacted] g of Np-237 and includes the inventory from PuFF Process Cells 1-5, PuFF Process Cells 6-9, ABL, the Plutonium Experiment Facility (PEF), Old Metallurgical Laboratory (OML), and various exhaust and ductwork components. The contribution of the other process areas and building components (excluding PuFF Process Cells 1-5 and ABL) was [redacted] g (14%) of Pu-238 and [redacted] g (13%) of Np-237 (SRNS 2020).

A conversion from the total number of grams (g) to curies (Ci) is provided in Table A-1a (Entire Building 235-F), Table A-1b (PuFF Process Cells 1-5), and Table A-1c (ABL). Grams of each radionuclide were multiplied by their specific activity (Ci/g) to obtain the number of curies of each (Plexus 2020). Total curies were converted to picocuries (10^{12} pCi per Ci) for use in the risk calculations.

A.4 Receptors / Exposure Assumptions

A standard receptor scenario assuming default exposure assumptions was used in this evaluation. The *default indoor worker* receptor scenario is a standard USEPA scenario which addresses long-term-risks to workers who are exposed to building contaminants within an industrial setting. The exposure assumptions for this default scenario are 25 years, 250 days per year, and 8 hours per day.

This hypothetical indoor worker would be exposed to the ambient air in the building via two exposure routes. The first route is inhalation of air. The second exposure route is submersion. Submersion is external exposure from the contaminated air. This assessment conservatively assumes that no personal protective equipment (PPE) is used, and that all of the contamination remains suspended in the air within the building.

A.5 Sources of Risk-Based Threshold Values

The USEPA publishes building preliminary remediation goals (BPRGs) for radiological constituents that are risk-based radioactivities that can be used to evaluate potentially contaminated buildings. BPRGs combine current USEPA toxicity values with standard exposure factors that represent reasonable maximum exposure (RME) conditions to estimate contaminant concentrations that the agency considers protective of humans over a lifetime. The concentrations are based on direct exposure pathways for which generally accepted methods, models, and assumptions have been developed for specific conditions.

The *USEPA Building Preliminary Remediation Goals for Radionuclides (BPRGs)* website (USEPA 2018a) is the source of the BPRGs used in this assessment. USEPA's CERCLA guidance on addressing building contamination can be found at this website. The website was accessed on May 4, 2020; the equation used to derive the BPRGs for an indoor worker exposure to ambient air is provided in Table A-2. This equation includes a half-life decay function and is appropriate in situations where the contaminant in the air is not being replenished (e.g., contaminated settled dust from a previous release that is being resuspended). The BPRGs for an indoor worker scenario are obtained using the website calculator function and assuming all default parameters.

The indoor worker BPRGs for ambient air are presented in Table A-3 (Pu-238 = 1.69E-04 pCi/m^3; Np-237 = 2.79E-04 pCi/m^3). The BPRGs that are published in Table A-3 correspond to a risk = 1E-06.

A.6 Risk Calculations

Risk calculations for the indoor worker receptor scenario were performed per the following equation:

Indoor Worker Risk = (Concentration [pCi/m^3] / Ambient Air BPRG [pCi/m^3]) x 1E-06

Building dimensions for volume estimates used in the risk calculations were obtained from Attachment E of the characterization report (SRNS 2020). The entire 235-F facility volume was determined by summating from the first-floor volume of 294320.25 ft^3 (8334.22 m^3)(area 21801.5 ft^2 [2025.4 m^2] with a ceiling height of 13.5 ft [4.1 m]) and the second-floor volume of 308285.6 ft^3 (8729.68 m^3)(area of 22020.4 ft^2 [2045.7 m^2] with a ceiling height of 14 ft [4.3 m]). The ABL volume was the combined volumes of rooms 107A and 107D (total area of 1026.7 ft^2 [95.4 m^2] with a ceiling height of 13.5 ft [4.1 m]) for a combined volume of 13860.45 ft^3 [392.48 m^3]. The east maintenance area (room 1002) was considered the primary area for a release from PuFF Cells 1-5, resulting in a total volume of 19946.25 ft^3 (564.81 m^3) (area of 1477.5 ft^2 [137.3 mt^2] with 13.5 ft [4.1 m] ceilings).

The results of the risk calculations for the default indoor worker scenario are provided in Table A-4a (Entire Building 235-F), Table A-4b (PuFF Process Cells 1-5), and Table A-4c (ABL) and are summarized below:

Receptor Scenario	Radionuclide	Entire Building 235-F Risk Estimate	PuFF Cells 1-5 Risk Estimate	Actinide Billet Line Risk Estimate
Default Indoor Worker	Pu-238	2.3E+09	5.9E+10	6.9E+08
	Np-237	4.1E+04	--	1.6E+06
	Total Risk =	**2.3E+09**	**5.9E+10**	**6.9E+08**

A.7 Human Health Risk Discussion

USEPA guidance defines the target risk range for carcinogenic risk due to exposure to a known or suspected carcinogen between one excess cancer in an exposed population of ten thousand (1E-04) and one excess cancer in an exposed population of one million (1E-06). Risks within this range require risk management evaluation of response actions to determine if risks can be reduced below one excess cancer in one million (1E-06). Risks greater than 1E-04 indicate that a response action is generally warranted.

In addition, source materials are those materials that include or contain hazardous substances, pollutants, or contaminants that act as a reservoir for migration to groundwater, surface water, or air, or that act as a source for direct exposure. Principal threat waste is defined as those source materials that have a high toxicity or mobility and cannot be reliably contained or present a significant risk to human health or the environment (USEPA 1991). This source material is referred to as principal threat source material (PTSM) at SRS. They include liquids and other highly mobile materials such as those released from surface soil due to volatilization, suspension of particulates/fugitive dusts, leaching, or materials having high concentrations of toxic compounds. No “threshold level” of toxicity/risk has been established to equate to “principal threat.” However, the guidance does state that treatment alternatives for source materials should generally be evaluated where the combined toxicity and mobility pose a potential risk of 1E-03 (one excess cancer in one thousand) or greater.

The risk estimates for Building 235-F are extremely high and are difficult to explain in terms of probability or risk. For example, a risk of 1E+00 relates to one excess cancer in a population of one, or 100%. The risk estimates for the default indoor worker range from 6.9E+08 (ABL) to 5.9E+10 (PuFF Cells 1-5). Obviously, the contaminated dust/powder material within Building 235-F is considered PTSM because the estimated risk (e.g., based on highest risk from PuFF Cells 1-5) is 5.9E+14 times greater than a risk of 1E-04.

In an effort to provide some perspective on these risk numbers, an additional evaluation was performed using the *USEPA Dose Compliance Concentrations for Radionuclides in Buildings at Superfund Sites* website (USEPA 2018b). The recommended Building Dose Compliance Concentrations (BDCCs) on this website are dose levels for contaminated buildings to help implement the National Contingency Plan (NCP) and USEPA CERCLA guidance. The website was accessed on May 11, 2020. The BDCCs for the *default indoor worker* scenario are shown in Table A-5; the limits shown in the table correspond to a dose of 1 mrem/year. Dose calculations for the indoor worker receptor scenario were performed per the following equation:

$$\textit{Indoor Worker Dose (mrem/yr)} = \textit{Concentration } [pCi/m^3] \,/\, \textit{Ambient Air BDCC } [pCi/m^3]$$

The results of the dose calculations for the default indoor worker scenario are provided in Table A-6a (Entire Building 235-F), Table A-6b (PuFF Process Cells 1-5), and Table A-6c (ABL) and are summarized below:

Receptor Scenario	Radionuclide	Entire Building 235-F Dose Estimate *(mrem/yr)*	PuFF Cells 1-5 Dose Estimate *(mrem/yr)*	Actinide Billet Line Dose Estimate *(mrem/yr)*
Default Indoor Worker	Pu-238	7.9E+14	2.0E+16	2.4E+14
	Np-237	3.1E+09	--	1.2E+11
	Total Dose =	**7.9E+14**	**2.0E+16**	**2.4E+14**

Title 10, Part 20, of the *Code of Federal Regulations* (10 CFR Part 20) "Standards for Protection Against Radiation," establishes the dose limits for radiation workers. Although the limits vary, depending on the affected part of the body, the annual total effective dose equivalent for the whole body is 5,000 mrem (5 rem). The estimated total dose (e.g., based on highest dose from PuFF Cells 1-5) is 4.1E+12 times greater than the USEPA acceptable dose limit of 5 rem/yr.

Although these risk estimates are based only on the primary process constituents (Pu-238 and Np-237), they provide sufficient information to justify a response action under CERCLA. However, it should be recognized that the risk estimate would be higher (although negligibly) if other contaminants within the facility (i.e., process impurities, decay products and hazardous materials) were considered in the calculations. In addition, this risk evaluation pertains to an indoor worker scenario only and does not consider potential impacts to the public (off-site) from atmospheric releases.

Nonradiological hazardous materials including asbestos, lead and PCBs are present in the building. Direct contact with these building materials/components typically does not offer a complete direct exposure pathway for the purposes of calculating risk under an industrial worker scenario. The lead and PCB inventories were more appropriately considered in a contaminant migration analysis that models fate and transport to groundwater. However, a qualitative risk discussion is provided for asbestos, PCBs, and lead to demonstrate the negligible risk contribution from these hazardous inventories when compared to the primary risk drivers.

Asbestos: Building 235-F was thoroughly inspected for asbestos in 2006 using facility knowledge, visual inspection completed by certified asbestos inspectors, document review, and the collection of bulk samples. This inspection identified transite-containing material in the building. There is no intent to remove asbestos-containing material during deactivation unless it is determined to be friable or is disturbed by deactivation activities.

Asbestos is a health hazard and its use is now highly regulated by both the Occupational Safety and Health Administration and the USEPA. Breathing asbestos fibers can cause a buildup of scar-like tissue in the lungs called asbestosis and result in loss of lung function that may progress to disability and death. Asbestos also causes cancer of the lung and other diseases such as mesothelioma. The USEPA Regional Screening Levels (RSLs) website (USEPA 2020) does not provide a screening level for asbestos for the purposes of estimating risk.

Lead: Lead is present in gloveboxes, leaded glass on glovebox windows, lead washers, brass valves, lead-acid batteries such as in emergency lights, lead shielding, lead solder in sewer line joints, and lead counterweights. No lead disturbance or dust production is expected during decommissioning of the facility. Direct contact with these building materials/components does not offer a complete direct exposure pathway for the purposes of calculating risk under an industrial worker scenario. However, exposure to lead-based paint is a well-documented, recognized health hazard, particularly in a residential setting or a pregnant worker scenario.

USEPA has no consensus reference dose or cancer slope factor for inorganic lead, so it is not possible to calculate screening levels as done for other chemicals. USEPA considers lead to be a special case because of the difficulty in identifying the classic "threshold" needed to develop a reference dose. USEPA therefore evaluates lead exposure by using blood-lead modeling, such as the Integrated Exposure-Uptake Biokinetic Model. The USEPA Office of Solid Waste has also released a detailed directive on risk assessment and cleanup of residential soil lead. The directive recommends that soil lead levels less than 400 mg/kg are generally safe for residential use. An updated screening level for soil lead at commercial/industrial (i.e., non-residential) sites of 800 part per million (ppm) is based on a recent analysis of the combined phases of the National Health and Nutrition Examination Survey that choose a cleanup goal protective for all subpopulations. The USEPA RSLs website (USEPA 2020) identifies an industrial worker screening level of 800 mg/kg for soil media.

Appendix E of the Characterization Report identifies a lead concentration of 0.940 g/kg (940 mg/kg) in paint (SRNS 2020). For illustrative purposes, a conservative hazard quotient calculation (HQ) can be performed using the 800 mg/kg soil threshold for a non-residential setting:

HQ = 940 mg/kg lead in paint / 800 mg/kg screening level = 1.2.

This lead calculation demonstrates a negligible risk compared to the risk associated with the primary risk drivers, Pu-238 and Np-237.

PCBs: PCBs may be present in paint, joint compounds, insulation, capacitors, hydraulic oil, and light ballasts. PCBs have been demonstrated to cause a variety of adverse health effects. Data are suggestive but not conclusive concerning the carcinogenicity of PCBs in humans. Paint samples were taken with a highest concentration of PCBs of 3,900 mg/kg (SRNS 2020).

The USEPA RSL website (USEPA 2020) identifies a screening level of 0.942 mg/kg for soil media for an industrial worker scenario. For illustrative purposes, a conservative risk calculation can be performed as follows:

Risk = (3,900 mg/kg PCB / 0.942 mg/kg RSL) x 1E-06 = 4.1E-03

This PCB calculation demonstrates a negligible risk compared to the risk associated with the primary risk drivers, Pu-238 and Np-237.

A.8 References

Plexus, 2020. *Specific Activities,* Plexus Scientific Corporation, Nuclear Solutions Division. http://www.iem-inc.com/information/tools/specific-activities/ Website accessed May 4, 2020

SRNS, 2020. *Characterization Report for Building 235-F, F-Area Material Storage Building (FAMS)*, G-ESR-F-00097, Rev. 0, Savannah River Nuclear Solutions, Savannah River Site, Aiken, SC

USEPA, 1991. *A Guide to Principal Threat and Low-Level Threat Wastes*, United States Environmental Protection Agency, OSWER Superfund Publication 9380.3-06FS

USEPA, 2018a. *USEPA Building Preliminary Remediation Goals for Radionuclides* website, United States Environmental Protection Agency. https://epa-bprg.ornl.gov/ Website accessed May 4, 2020

USEPA, 2018b. *USEPA Dose Compliance Concentrations for Radionuclides in Buildings at Superfund Sites* website, United States Environmental Protection Agency. https://epa-bdcc.ornl.gov/ Website accessed May 11, 2020.

USEPA, 2020. *USEPA Regional Screening Levels website*, United Stated Environmental Protection Agency. https://www.epa.gov/risk/regional-screening-levels-rsls /

This page was intentionally left blank.

Table A-1a. Grams to Curies Conversion Entire Building 235-F

Radionuclide	Grams[1] *(g)*	Specific Activity[2] *(Ci/g)*	Curies[3] *(Ci)*	picoCuries[4] *(pCi)*
Primary Process Contaminants (Pu-238 & Np-237)				
Pu-238	[redacted]	1.7E+01	[redacted]	[redacted]
Np-237	[redacted]	6.9E-04	[redacted]	[redacted]

Table A-1b. Grams to Curies Conversion Puff Process Cells 1-5

Radionuclide	Grams[1] *(g)*	Specific Activity[2] *(Ci/g)*	Curies[3] *(Ci)*	picoCuries[4] *(pCi)*
Primary Process Contaminant (Pu-238)				
Pu-238	[redacted]	1.7E+01	[redacted]	[redacted]

Table A-1c. Grams to Curies Conversion Actinide Billet Line

Radionuclide	Grams[1] *(g)*	Specific Activity[2] *(Ci/g)*	Curies[3] *(Ci)*	picoCuries[4] *(pCi)*
Primary Process Contaminants (Pu-238 & Np-237)				
Pu-238	[redacted]	1.7E+01	[redacted]	[redacted]
Np-237	[redacted]	6.9E-04	[redacted]	[redacted]

1 - Grams from Table 12, "Building 235-F Holdup" in the Characterization Report for Building 235-F (SRNS 2020).
2 - Specific activity is the amount of radioactivity - or the decay rate - of a particular radionuclide per unit mass of the radionuclide, expressed as curies per gram (Plexus 2020).
3 - Curies = Grams x Specific Activity
4 - picoCuries = Curies x 1E12

Table A-2. Equation for Indoor Worker Exposure to Ambient Air

$$BPRG\ \left(\frac{pCi}{m^3}\right) = \frac{TR \times t\ (yr)\ \times\ \lambda\ \ \times (\frac{1}{yr})}{ET\ \left(\frac{8\ hr}{day}\right) \times \left(\frac{1\ day}{24\ hr}\right) \times EF\ \left(\frac{250\ days}{yr}\right) \times ED\ (25\ yr) \times F_{in} \times F_i \times (1 - e^{-\lambda t}) \times}$$

$$[\left(SF_i\ \left(\frac{risk}{pCi}\right) \times IFA\ \left(\frac{60\ m^3}{day}\right)\right) + \left(SF_{sub}\ \left(\frac{risk/yr}{pCi/m^3}\right) \times \left(\frac{1\ yr}{365\ days}\right) \times GSF\right)]$$

Where:

BPRG = Building Preliminary Remediation Goal for exposure to ambient air (pCi/m^3)

Target Risk (TR) = 1E-06

Indoor Worker time (t) = 25 years

Isotope specific decay constant (λ) = 0.693/half-life

Indoor Worker air exposure time (ET) = 8 hours/day

Indoor Worker exposure frequency (EF) = 250 days/year

Indoor Worker exposure duration (ED) = 25 years

Fraction time spent indoors (unitless)(F_{in}) = 1

Fraction of time spent in compartment (unitless)(F_i) = 1

Isotope specific inhalation slope factor – air (SF_i)(risk/pCi)

Indoor Worker inhalation rate (IFA) = 60 m^3/day

Isotope specific external exposure slope factor – submersion (SF_{sub}) (risk/yr per pCi/m^3)

Gamma shielding factor (unitless) (GSF) = 1 (assumes no shielding)

Table A-3. Default Indoor Worker BPRGs for Ambient Air

Default Indoor Worker Equation Inputs for Air

Variable	Value
TR (target cancer risk) unitless	0.000001
t_{iw} (time - indoor worker) yr	25
ED_{iw} (exposure duration - indoor worker) yr	25
ET_{iw} (exposure time - indoor worker) hr/day	8
EF_{iw} (exposure frequency - indoor worker) day/yr	250
F_{in} (fraction of time spent indoors) unitless	1
F_i (fraction of time spent in compartment) unitless	1
GSF_a (air gamma shielding factor) unitless	1
IRA_{iw} (inhalation rate - indoor worker) m^3	60

Default Indoor Worker BPRGs for Air

Radionuclide	Inhalation Slope Factor (risk/pCi)	External Exposure Slope Factor (Submersion) (risk/yr per pCi/m^3)	Lambda (1/yr)	Halflife (yr)	Inhalation BPRG TR=1E-06 (pCi/m^3)	External Exposure BPRG TR=1E-06 (pCi/m^3)	Total BPRG TR=1E-06 (pCi/m^3)
Np-237	2.87E-08	7.67E-11	3.23E-07	2.14E+06	2.79E-04	2.28E+03	**2.79E-04**
Pu-238	5.22E-08	2.56E-13	7.90E-03	8.77E+01	1.69E-04	7.55E+05	**1.69E-04**

Output generated 04May2020

Table A-4a. Risk Calculation for Default Indoor Worker — Entire Building 235-F

Radionuclide	Concentration[1] (pCi/m³)	Indoor Worker Ambient Air BPRG[2] (pCi/m³)	Indoor Worker Risk Estimate[3]
Pu-238	[redacted]	1.69E-04	2.29E+09
Np-237		2.79E-04	4.12E+04
Default Indoor Worker Entire Building Total Risk[4] =			**2.29E+09**

1 - Concentration (pCi/m3) = Total pCi/Volume (m3)
2 - Default Indoor Worker Ambient Air BPRG from Table A-3
3 - Risk Estimate = (Concentration/BPRG) x 1E-06
4 - Total Risk = Risk Estimate Pu-238 + Risk Estimate Np-237

Entire Building Dimensions

	Summed Area (ft2)	*Room Height (ft)*	*Volume (ft3)*	*Volume (m3)*
1st floor	21801.5	13.5	294320.25	8334
2nd floor	22020.4	14.0	308285.6	8730
Total			602605.85	17064

	*Total Ci**	*Total pCi*	***Concentration (pCi/m3)***	
Pu-238	[redacted]		[redacted]	**Pu-238**
Np-237				**Np-237**

*Total Curies from Table A-1a

Table A-4b. Risk Calculation for Default Indoor Worker — PuFF Process Cells 1-5

Radionuclide	Concentration[1] (pCi/m³)	Indoor Worker Ambient Air BPRG[2] (pCi/m³)	Indoor Worker Risk Estimate[3]
Pu-238	[redacted]	1.69E-04	5.91E+10
Default Indoor Worker PuFF Process Cells 1-5 Total Risk[4] =			**5.91E+10**

1 - Concentration (pCi/m3) = Total pCi/Volume (m3)
2 - Default Indoor Worker Ambient Air BPRG from Table A-3
3 - Risk Estimate = (Concentration/BPRG) x 1E-06
4 - Total Risk = Risk Estimate Pu-238 only

PuFF Process Cells 1-5 Dimensions

Footprint (ft2)	*Room Height (ft)*	*Volume (ft3)*	*Volume (m3)*
1477.5	13.5	19946.25	565

	*Total Ci**	*Total pCi*	***Concentration (pCi/m3)***	
Pu-238	[redacted]		[redacted]	**Pu-238**

*Total Curies from Table A-1b

Table A-4c. Risk Calculation for Default Indoor Worker — Actinide Billet Line

Radionuclide	Concentration[1] (pCi/m³)	Indoor Worker Ambient Air BPRG[2] (pCi/m³)	Indoor Worker Risk Estimate[3]
Pu-238	[redacted]	1.69E-04	6.92E+08
Np-237		2.79E-04	1.56E+06
Default Indoor Worker Actinide Billet Line Total Risk[4] =			**6.94E+08**

1 - Concentration (pCi/m3) = Total pCi/Volume (m3)
2 - Default Indoor Worker Ambient Air BPRG from Table A-3
3 - Risk Estimate = (Concentration/BPRG) x 1E-06
4 - Total Risk = Risk Estimate Pu-238 + Risk Estimate Np-237

Actinide Billet Line Dimensions

Footprint (ft2)	*Room Height (ft)*	*Volume (ft3)*	*Volume (m3)*
1026.7	13.5	13860.45	392

	*Total Ci**	*Total pCi*	***Concentration (pCi/m3)***	
Pu-238	[redacted]		[redacted]	**Pu-238**
Np-237				**Np-237**

*Total Curies from Table A-1c

Table A-5. Default Indoor Worker BDCCs for Ambient Air

Default Indoor Worker Equation for Air

Variable	Value
DL (dose limit) mrem/yr	1
t_{iw} (time - indoor worker) yr	1
EF_{iw} (exposure frequency) day/yr	250
ET_{iw} (exposure time - indoor worker) hr/day	8
IRA_{iw} (inhalation rate - indoor worker) m^3/day	60
F_i (fraction of time spent in compartment) unitless	1
F_{in} (fraction of time spent indoors) unitless	1
GSF_a (air gamma shielding factor) unitless	1

Default Indoor Worker DCCCs for Air

Radionuclide	Inhalation DCF (mrem/pCi)	External Exposure DCF (Submersion) (mrem/yr per pCi/m^3)	Lambda (1/yr)	Halflife (yr)	Inhalation BDCC DL=1 (pCi/m^3)	External Exposure BDCC DL=1 (pCi/m^3)	Total BDCC DL=1 (pCi/m^3)
Np-237	4.66E-02	1.61E+02	3.23E-07	2.14E+06	4.29E-03	2.73E-02	**3.71E-03**
Pu-238	4.07E-01	6.28E-01	7.90E-03	8.77E+01	4.93E-04	7.01E+00	**4.93E-04**

Output generated 11May2020

Table A-6a. Dose Calculation for Default Indoor Worker — Entire Building 235-F

Radionuclide	Concentration[1] (pCi/m³)	Indoor Worker Ambient Air BDCC[2] (pCi/m³)	Indoor Worker Dose Estimate[3] (mrem/yr)
Pu-238	[redacted]	4.93E-04	7.86E+14
Np-237	[redacted]	3.71E-03	3.10E+09
Default Indoor Worker Entire Building Total Dose[4] =			**7.86E+14**

1 - Concentration (pCi/m3) = Total pCi/Volume (m3)
2 - Default Indoor Worker Ambient Air BDCC from Table A-5
3 - Dose Estimate = Concentration/BDCC
4 - Total Dose = Dose Estimate Pu-238 + Dose Estimate Np-237

Entire Building Dimensions

	Summed Area (ft2)	*Room Height (ft)*	*Volume (ft3)*	*Volume (m3)*
1st floor	21801.5	13.5	294320.25	8334
2nd floor	22020.4	14.0	308285.6	8730
Total			602605.85	17064

	*Total Ci**	*Total pCi*	***Concentration (pCi/m3)***	
Pu-238	[redacted]		[redacted]	**Pu-238**
Np-237				**Np-237**

*Total Curies from Table A-1a

Table A-6b. Dose Calculation for Default Indoor Worker — PuFF Process Cells 1-5

Radionuclide	Concentration[1] (pCi/m³)	Indoor Worker Ambient Air BDCC[2] (pCi/m³)	Indoor Worker Dose Estimate[3] (mrem/yr)
Pu-238	[redacted]	4.93E-04	2.03E+16
Default Indoor Worker PuFF Process Cells 1-5 Total Dose[4] =			**2.03E+16**

1 - Concentration (pCi/m3) = Total pCi/Volume (m3)
2 - Default Indoor Worker Ambient Air BPRG from Table A-5
3 - Dose Estimate = (Concentration/BDCC)
4 - Total Dose = Dose Estimate Pu-238 only

PuFF Process Cells 1-5 Dimensions

Footprint (ft2)	*Room Height (ft)*	*Volume (ft3)*	*Volume (m3)*
1477.5	13.5	19946.25	565

	*Total Ci**	*Total pCi*	***Concentration (pCi/m3)***	
Pu-238	[redacted]		[redacted]	**Pu-238**

*Total Curies from Table A-1b

Table A-6c. Dose Calculation for Default Indoor Worker — Actinide Billet Line

Radionuclide	Concentration[1] (pCi/m³)	Indoor Worker Ambient Air BDCC[2] (pCi/m³)	Indoor Worker Dose Estimate[3] (mrem/yr)
Pu-238	[redacted]	4.93E-04	2.37E+14
Np-237	[redacted]	3.71E-03	1.17E+11
Default Indoor Worker Actinide Billet Line Total Dose[4] =			**2.37E+14**

1 - Concentration (pCi/m3) = Total pCi/Volume (m3)
2 - Default Indoor Worker Ambient Air BDCC from Table A-5
3 - Dose Estimate = Concentration/BDCC
4 - Total Dose = Dose Estimate Pu-238 + Dose Estimate Np-237

Actinide Billet Line Dimensions

Footprint (ft2)	*Room Height (ft)*	*Volume (ft3)*	*Volume (m3)*
1026.7	13.5	13860.45	392

	*Total Ci**	*Total pCi*	***Concentration (pCi/m3)***	
Pu-238	[redacted]		[redacted]	**Pu-238**
Np-237				**Np-237**

*Total Curies from Table A-1c

APPENDIX B

Fate and Transport Analysis for Building 235-F

This page was intentionally left blank.

B.1 Introduction

This appendix summarizes the evaluation used for determining the expected performance of the removal action alternatives with respect to the protection of groundwater from the contaminants within Building 235-F (SRNL 2020). The final evaluation was performed in the 1-dimensional (1-D) GoldSim model (Version 12.1, GTG 2018), which provides the ability to perform stochastic Monte Carlo simulations to account for parameter uncertainty over the long time periods modeled. The GoldSim model was supported by the 2-dimensional (2-D) and 3-dimensional (3-D) modeling in PORFLOW (Version 6.42.9, ARCi 2010). PORFLOW has advanced flow and transport parameters for the building and surrounding Upper Three Runs Aquifer (UTRA) but does not account for parameter uncertainty. Therefore, the fate and transport parameters established in PORFLOW were then incorporated into the GoldSim model to perform the final analysis.

B.2 Alternatives

The following fate and transport modeling scenarios, including a No Action scenario (baseline conditions) and three in-situ decommissioning (ISD) scenarios were evaluated in the groundwater model to support the removal action alternatives considered in the RSER/EE/CA.

1) **No Action Scenario**. Represents Building 235-F deactivation state in regard to facility condition and radiological holdup locations. Future deactivation activities are not expected to result in changes to the no action scenario since cementitious material (concrete floors and walls) and contaminant locations are to remain largely unaltered during deactivation.
2) **Grout First Building Level Scenario**. Second level contaminated equipment is moved from the second level to the first level process areas and first level process areas area grouted.
3) **Grout First and Second Level Scenario**. First and second level process areas are grouted, leaving all contamination in place.

4) **Grout First and Second Levels with Engineered Roof Scenario.** First and second level process areas are grouted, leaving all contamination in place. An engineered roof is placed over Building 235-F.

These fate and transport modeling scenarios are designed to support the removal action alternatives proposed in the RSER/EE/CA in the ultimate 1-D transport context. The No Action scenario assumes the building structure will be maintained for 150 years through preventive maintenance. The second scenario assumes that all radiological holdup is grouted in place on the first level and the facility roof is maintained for 150 years. The second level floor slab is then assumed to be intact for an additional 600 years. Only grouting the areas containing radiological holdup is required due to the 1-D nature of the model and grouting the non process areas on the first level provides no additional protectiveness. For the third scenario, the structure is assumed to be intact for 1,000 years with both the first and second level process areas being grouted. Due to the 1-D nature of the final GoldSim model, for any area with radiological holdup, both the first and second levels are assumed to be grouted even if no radiological holdup is present in the corresponding area above or below. The fourth scenario is the same as the third with the addition of an engineered sloped concrete reinforced roof slab with integral crystalline waterproofing. The engineered roof is assumed to last over 1,000 years. Again, due to the 1-D nature of the model, the grout and roof is required in/below/above any areas were radiological holdup is located.

Although the modeling results for all four scenarios are discussed in Appendix B, the second and third scenarios were not carried forward as proposed removal action alternatives in the RSER/EE/CA. The second scenario to grout only the first building level involved relocating the second level process equipment and ventilation ducts to the first level with no modifications to the roof. This modeling scenario was not carried forward as a proposed removal alternative due to the high risk to the decommissioning worker to move the contaminated equipment. The third scenario to grout the first and second level process areas was not considered for a removal action alternative in the RSER/EE/CA because the existing roof could not serve as a robust barrier for 750 years without the modifications or installation of an engineered roof.

B.3 Data Inputs

B.3.1. Model Parameters

B.3.1.1 Infiltration Rates

Infiltration passing through the Building 235-F is driven by the condition of the building over time (e.g., intact versus collapsed) and has a significant impact upon migration of contaminants from the building. As Building 235-F has a flat concrete roof similar to the P-Reactor Building roof, a comparable performance of the roof is expected if vegetative growth is not prevented (i.e., roof collapse in ~150 yrs). By grouting the entire building, the potential for roof collapse is essentially eliminated. It is expected that addition of a sloped concrete reinforced roof slab with integral crystalline waterproofing to Building 235-F increases the time of loss of roof integratory to greater than 1,000 years.

Infiltration rate estimates employed, as supported by the Hydrologic Evaluation of Landfill Performance (HELP) analyses, (Schroeder et al., 1994) were primarily based on earlier scoping efforts by Savannah River National Laboratory (SRNL 2012) and those used in the R- and P-Reactor decommissioning efforts (Council 2008 and 2009). HELP model results were employed to establish degradation impact on infiltration by estimating rates through various media simulating various levels of degraded concrete. Infiltration through Building 235-F was estimated under the following conditions:

- Intact building infiltration rate – 1.2 cm/yr. (0.49 in/yr.);
- Partially collapsed building infiltration rate (loss of roof integrity) – 31.5 cm/yr. (12.4 in/yr.); and
- Completely collapsed building infiltration rate (loss of roof and second level floor integrity) – 62.5 cm/yr. (24.6 in/yr.)

The timing (with distributions used in stochastic realizations) for when these infiltration rates were employed in the 1-D model are:

1) **No Action Scenario –** Intact until a mean value of 150 ± 50 years beyond the year 2025, the assumed date of decommissioning.

2) **Grout First Level Scenario –** Building 235-F roof collapse at a mean value of 150 ± 50 years beyond the year 2025. Remainder of the building, including the remaining second level floor slab, collapses at a mean value of 750 ± 250 years beyond the year 2025.
3) **Grout ~~Entire 235-F~~First and Second Level Scenario –** The entire building collapses at a mean value of 750 ± 250 years beyond the year 2025. The structure is completely grouted with little void space and collapse of the roof onto the second level floor cannot occur. The erosion of the cementitious material results in a gradual loss in integrity causing the collapse of both levels simultaneously.
4) **Grout ~~Entire 235-F~~First and Second Levels with Engineered Roof Scenario.** The fourth scenario is based on using the roof designs employed for R- and P-Reactor Buildings decommissioning (Council 2009 and 2008, respectively). These engineered roof designs provide sufficient slope to greatly reduce vegetative growth beyond the period of land use control. In addition, these designs included an integral crystalline waterproofing material additive to the concrete mixtures to help potential cracks self-heal. HELP model results were employed to estimate degradation impact to the engineered roof performance. A comparison of roof performance for the R- and P-Reactor Building roof designs and the resulting infiltration rates used for Building 235-F is provided in Figure B-1. These infiltration rates vary slightly from other scenarios over the first 750 years due to the increased knowledge about engineered roof performance from R- and P-Reactors.

A visualization of all infiltration rates used in the modeling is provided in Figure B-2.

B.3.1.2. Sorption Coefficients and Solubilities

The chemical sorption and its degradation aspects for use at SRS have been studied extensively and an official “Geochemical Data Package” for use in performance assessments was generated (SRNL 2016). All chemical sorption coefficients (i.e., K_d values) and apparent liquid-phase solubility limits used were taken from this 2016 Geochemical Data Package. For cementitious materials, the conceptual model employed is based on changing pH values. Aging of cement with infiltration (quantified in exchange volumes) was including in the modeling effort through the change in pH and material composition. These changes are represented in four stages with corresponding changes in sorption coefficients for each radionuclide to reflect differences in contaminant fate and transport (Figure B-3).

B.3.1.3 Vadose Zone and Aquifer Properties

Building 235-F is located in F-Area, part of the General Separation Area (GSA), at SRS. The GSA is a highly characterized region from a hydrogeologic perspective. Characterization data have been collected in this region spanning several decades. SRNL has developed detailed 3D aquifer flow models of the GSA dating back to the mid-1990s. The most recent update to the 3D flow model is based on the PORFLOW code where updated calibration data and correlation techniques were employed (SRNL 2018). Data from the recent updated 3D flow model (referred to as the GSA2018 flow model), including aquifer and soil characteristics, was used for all transport simulations presented within this report.

B.3.1.4. Dose and Activity Calculation

SRNL Dose Toolkit (SRNL 2015) was used to compute all dose related values from concentration/activities of modeled radionuclides. SRNL Dose Toolkits generates dose through a series of calculations. First, concentrations from either PORFLOW or GoldSim are inputted to the SRNL PreDose module. The PreDose module expands the short-chain radionuclide decay chain results in the PORFLOW and GoldSim files to produce full decay chain results using the assumption of secular equilibrium. The results for the PreDose Module are inputted in the FAREA Dose Tool which computes POA concentration and dose impacts of parent radionuclides. The output of the FAREA Dose Tool includes a summary table (Tables B-1 through B-4) and plots of peak concentrations.

B.3.2. Inventory/ Radiological Holdup

B.3.2.1. Radiological Inventory

The radiological inventory or holdup in Building 235-F is Pu-238 and Np-237 in the form of oxide powders (SRNS 2020). The radiological holdup is primary located with PuFF Cells 1-5 (Pu-238) and ABL gloveboxes (Np-237) with some additional inventory spread through the other process areas. As a conservative measure (leading to higher concentrations), the entire mass of Pu-238 was placed in PuFF Cells 1-5 where each cell had its individual inventory and the radiological holdup in the rest of the facility was added to the Cell 5 inventory. This arrangement provides an upper bound of predicted concentrations in the plume emanating from the facility due to the direction of groundwater flow beneath the facility (Figure B-4). Likewise, all Np-237 was placed within ABL with most of the inventory being assigned to the cells and the rest of the inventory for the facility distributed over the entire ABL footprint.

The oxide powder feedstocks (PuO_2 and NpO_2 which included some impurities) were introduced and processed in Building 235-F over a period of several years centered about the year 1981. In the PORFLOW and GoldSim modeling of the ABL and PuFF facilities, parent nuclides were set to their initial values starting in 1981 (i.e., 07/01/1981 or year 1981.5). These parent inventories were decayed in place for 44 years to simulate a decommissioning year of 2025. Preliminary analysis showed that because the impurities present within the oxide powders were negligible, they imparted minimal impact on groundwater concentrations relative to Pu-238 and Np-237 and were not considered further. All progeny of Pu-238 and Np-237 with a half-life over 1 year were explicitly modeled.

Fate and transport modeling was completed prior to the completion of the final assays of facility (SRNL 2020). As such, modeling results were scaled (see discussion in section B.3.7) to the final total Pu-238 and Np-237 quantities for Building 235-F reported in the *Characterization Report for Building 235-F, F-Area Material Storage Building (FAMS)* (SRNS 2020) (i.e., Table 1 of this RSER/EE/CA report). The relative location of the inventory placement was still largely accurate considering the placement of radiological holdup within the model only within the PuFF cells and ABL.

B.3.2.2 Elemental Lead

Elemental lead, used primarily for shielding purposes, is located in several places within the Building 235-F. The modeled lead inventory was 26,447 kilograms (kg) (58,184 58,305 pounds [lbs]). This inventory conservatively bounds the estimated 17,244 kgs (37,937 38,016 lbs) of lead documented in the Characterization Report (SRNS 2020). For the GoldSim model, the total inventory of elemental lead was placed within PuFF Cell 3.

B.3.2.3 Polychlorinated biphenyls (PCBs)

PCBs were assumed to be contained within the surface paint employed throughout Building 235-F. The estimated total PCBs inventory is 2,375 g2.38 kgs (5.25 lbs) in paint (SRNS 2020). PCBs were equally distributed into PuFF Cells 1-5.

B.4. Model Setup

B.4.1. Points of Assessment (POAs)

The five POAs evaluated in the GoldSim model include the following:

- **1-m POA –** groundwater one meter (3.3 ft) beyond the outside perimeter of Building 235-F;
- **100-m POA –** groundwater 100 meters (330 ft) beyond the outside perimeter of Building 235-F (Figure B-5);
- **360-m POA –** groundwater 360 meters (1,181 ft) beyond the outside perimeter of Building 235-F which corresponds to the boundary of F Area industrialized area (Figure B-5);
- **Seepline POA -** groundwater 683 meters (2,241 feet) beyond the outside perimeter of Building 235-F which represents water seeping into the closest groundwater fed surface water intersecting the plume path (Figure B-6)

- **Unnamed Upper Three Runs (UTR) Tributary POA –** surface water in the unnamed UTR tributary after dilution into upgradient surface water (assumed clean)

Though all POAs are considered, the 360-m POA was used as the problem defining POA based on discussions with the United States Department of Energy (USDOE), U.S. Environmental Protection Agency (USEPA), and South Carolina Department of Health and Environmental Control (SCDHEC) at the October 19, 2019 informational meeting (USDOE 2019). The POA at the 360-m (1,181-ft) is expected to be consistent with the POA for all F Area sources in support of the eventual F-Area Operable Unit (FAOU) Record of Decision (ROD), while an assessment period of 0-10,000 years (0-10K) allows for the longer travel time for the longer-lived radionuclides.

B.4.2. Performance Criteria

Predicted concentrations at each POA were compared to the maximum concentration levels (MCLs) for gross alpha (15 pCi/L), beta-gamma dose (4 mrem/yr), radium (Ra-226 + Ra-228, 5 pCi/L), uranium (30 µg/L), lead (15 µg/L), and PCBs (0.5 µg/L).

B.4.3. Timeframes

The timeframes considered were 0-1,000 years (0-1K years), 0-10,000 years (0-10K years), and 0-100,000 years (0-100K years) with maximum concentrations over each time frame reported. Time of the peak concentration in each timeframe is also documented. The start year for each timeframe is 2025, the assumed year for the start of decommissioning activities. Sensitivity analyses showed that delaying the decommissioning until the year 2045 would have very little impact on the results. The maximum concentrations for 0-10,000 year timeframe were used as the problem defining criteria based on discussions with the USDOE, USEPA, and SCDHEC at the October 19, 2019 informational meeting (USDOE 2019) and consistent with the timeframe for removal actions at the P- and R-Reactor buildings.

B.4.4. Deterministic Plume and Dispersion Modeling (PORFLOW)

Building 235-F, the vadose zone beneath, and the surrounding UTRA were first modeled in PORFLOW to establish the plume characteristics that govern transport.

PORFLOW was first used to establish the plume path for contaminants released from Building 235-F. Preliminary tracer analysis indicated that the source terms associated with Building 235-F are isolated from all other sources within that region 3-D stream-traces (with 10-yr travel time markers shown as solid red circles) for the following facilities were analyzed:

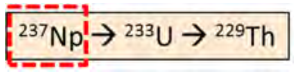
- Building 235-F facilities (i.e., ABL, PuFF Facility, and Sandfilter [294-2F]); and
- Other "hardened" facilities (i.e., Buildings 221-F, 772-F, 772-1F, 772-4F, 294F, and 294-1F).

The various 3D stream-traces fan out from left to right as a result of two major tributaries located on the northwest and northeast sides of F Area (Figure B-7). With this overall aquifer flow pattern, plume overlap (i.e., intermingling of neighboring contaminant plumes) is minimal for the various facilities of potential interest. Contaminants emanating from the Building 235-F facilities outcrop at seepage faces along the 288-F Ash Basin tributary and then migrate down this tributary to UTR. As such plumes emanating from the Building 235-F can be considered separate from any of the other facilities investigated.

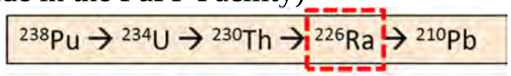
PORFLOW was used to establish the interaction between plumes emanating from the PuFF cells and the ABL. 3D steady-state tracer simulations were performed to determine the spatial interaction. These studies established that there is little plume overlap from each facility at the 1-m POA and there is up to a 40% interaction of plumes at the 100-m POA (Figure B-8).

In light of the limited spatial plume interaction potential at the 1-m POA, the potential impact of plume temporal interactions at the 100-m and beyond was investigated in PORFLOW. Temporal impacts are associated with half-life and sorption coefficient differences among parent nuclides and their progeny. The main two parent nuclides of interest within the Building 235-F are Np-237 within the ABL facility and Pu-238 within the PuFF. The abbreviated decay chains based on a 1-yr cutoff (i.e., short chains) are:

Np-237 - key parent nuclide in the ABL



Pu-238 (key parent nuclide in the PuFF Facility)



The dashed boxes indicate which short chain member(s) control the concentration and dose. For each facility the controlling member is:

ABL – Np-237 controls dose (Pa-233 in secular equilibrium)

PuFF Facility – Ra-226 controls dose (Po-218 and Po-214 in secular equilibrium)

To illustrate the temporal aspects at the 100-m POA, ABL and PuFF Facility PORFLOW 3D transport runs were made for the “Grout Entire 235-F” ISD scenario. The resulting concentrations of the two controlling radionuclides are plotted in Figure B-9. Np-237 has significantly different aquifer sorption coefficients versus the values for Ra-226. Therefore, the Np-237 peak occurs significantly sooner with a sharper shape than for Ra-226. Limited interaction between the plumes is expected at the 100-m POA and beyond due the difference in transport of the two dose and concentration controlling radionuclides. As such, due to both spatial and temporal effects, minimal overlap expected for the plumes emanating from ABL and PuFF Facility and these plumes were considered separate sources in further analysis.

B.4.5. Calibration

PORFLOW results were used to calibrate the GoldSim model because GoldSim is 1-D and does not solve porous-media flow equations. Several aspects associated with the fate and transport of contaminants from Building 235-F to the various POAs of interest are multidimensional in nature, and therefore can only be properly modeled in PORFLOW. However, PORFLOW does not provide the means to perform Monte Carlo simulations which account for model uncertainty through the stochastic variation of parameters. Therefore, the GoldSim model was calibrated to match PORFLOW deterministic results.

GoldSim transport simulations under a deterministic mode were compared to PORFLOW-based deterministic simulations where peak concentrations at the POAs of interest were assessed (i.e., timing of the peaks was also reviewed). Once a satisfactory match between the PORFLOW and GoldSim model results is obtained, GoldSim is used with confidence to perform Monte Carlo analyses running multiple simulations while sampling uncertainty distributions assigned to model variables. Results from the Monte Carlo simulations can then be used to determine a measure of uncertainty in the calculated contaminant concentrations and doses.

B.4.6. Probabilistic Modeling (GoldSim)

The physical shape, dimensions, and orientation of the Building 235-F PuFF Facility, compared to the ABL, are considerably different. Therefore, two separate GoldSim models were required, though both models have the same general structure and parameters such as material properties, infiltration rates, and radionuclide properties. Along with the morphological differences, substantial differences in mobility exist between the radionuclide contaminants (Np-237 vs Pu-238) contained within the two facilities. Numerically, this necessitates using smaller time stepping and finer spatial resolution to adequately capture peak concentrations for ABL. Conversely, radionuclide contaminants found in the PuFF Facility are relatively immobile, which tends to result in smoother concentration profiles that can be adequately modeled using larger time steps and coarser spatial resolution. Separating the two models based on these differences has an added benefit of reducing the computational runtime of Monte Carlo simulations for uncertainty analysis. Each model was ultimately designed to predict the radionuclide fate and transport for the three ISD model scenarios.

Both GoldSim models are 1-D representations of flow and radionuclide transport through the Building 235-F, the unsaturated vadose zone below the building, and along the aquifer to its outflow at the unnamed tributary to the UTR. A schematic diagram of the GoldSim PuFF Facility model of the vadose zone region, including the Building 235-F, is shown in Figure B-10.

The thicknesses and widths of each of these sections below the contaminant zone (i.e., PuFF cells or ABL) are consistent with drawings, known aquifer paraments, and aligned with PORFLOW results. The regions above the source zone are not explicitly modeled but are accounted for implicitly through infiltration and cementitious material breakdown. Infiltration through the building is handled using switch elements that trigger the intact, partial, or collapsed infiltration rate depending on both time and the ISD scenario. The degradation timeline and infiltration rates are consistent with the deterministic PORFLOW models.

Uncertainty within the 235-F GoldSim fate and transport model was represented using 54 stochastic elements. The model was designed as a stochastic model and can provide both deterministic and stochastic (probabilistic) results. During probabilistic simulation runs, GoldSim applies the Monte Carlo technique across the set of stochastic elements within the model. In the context of Building 235-F, the result of each case-specific probabilistic simulation run is a distribution of possible outcomes based on varied parameters. The results presented below (Tables B-1 and B-2) are the peak of means for 10,000 Monte Carlo realizations for each scenario, after scaling to account for changes in inventory (see section B.3.7)

B.4.7. Concentration Scaling with Inventory

The peak concentrations and doses at any given POA are a function of the assumed initial parent nuclide inventory. Both the vadose zone and aquifer transport equations are linear because the sorption isotherms are assumed to be linear. Within Building 235-F, both contaminant zones (i.e., ABL and PuFF Facility) contain non-linear isotherms due to solubility constraints placed on the key radionuclides when interacting with concrete and/or grout. However, the solubility constants only impact transport for a short timeframe and limited distance. Further, preliminary PORFLOW vadose zone and aquifer transport analyses were made where both Np-237 in the ABL facility and Pu-238 in the PuFF Facility were varied. The results from these sensitivity cases demonstrated that concentration and dose at all POA were essentially linear with respect to parent inventory values. As such, concentration dose estimates for any future updated assay values can be computed from the original model results using the formula:

$$C_i^{New} = \left[\frac{C_i^{New}}{C_i^{Old}}\right] C_i^{Old} \qquad (1)$$

This scaling allows for the simple conversion from the inventory value used in the original modeling effort (Hamm et al. 2020) and the updated values report in the Characterization Report (SRNS 2020).

B.5. Results

B.5.1. No Action Scenario

The No Action scenario was not considered to be protective of groundwater. The plume emanating from ABL is predicted to exceed the gross alpha MCL (15 pCi/L) at the 360-m POA within 10,000 years.

Exceedances are predicted to occur in the plume emanating from the PuFF cells (Table B-1). At the 1-m POA, Pu-238 in the PuFF cells results in large exceedances of gross alpha activity (614 pCi/L) and radium activity (204 pCi/L) before 10,000 years (Table B-2, Figure B-11). Beta-gamma dose (7 mrem/yr) also exceeds the MCL at the 1-m POA. Maximum concentrations of gross alpha (3474 pCi/L), beta-gamma dose (42 mrem/yr) and radium (1123 pCi/L) are predicted to occur around 25,000 years. At the 100-m POA, predicted gross alpha (38 pCi/L) and radium (13 pCi/L) also exceed MCLs within 10,000 years. Maximum concentrations of both gross alpha and radium are predicted to occur at 30,000 – 31,000 years at the 100-m POA. Groundwater concentrations at the 360-m and seepline POAs do not exceed MCLs until after 10,000 years with peak concentrations at 39,000 years and 50,000 years, respectively. Predicted concentrations in surface water in the unnamed UTR tributary are not significantly different than zero (<0.01).

Large exceedances of gross alpha activity (1,319 pCi/L) and beta-gamma dose (18 mrem/yr) are also predicted for the plume emanated from ABL with the exceedance of gross alpha occurring before 1,000 years due to the mobility of Np-237 (Table B-3). Maximum concentrations are predicted to occur at 1,340 years (Table B-4, Figure B-12) with beta-gamma dose also exceeding the MCL at that time. Exceedances in gross alpha are also predicted to occur at the 100-m, 360-m, and seepline POAs before 10,000 years with peak concentrations occurring at 1,410 years, 1,610 years, and 1,680 years, respectively. Predicted concentrations in surface water in the unnamed UTR tributary are not significantly different than zero (<0.02).

Elemental lead is not predicted to exceed its MCL for over 100,000 years in the No Action scenario at the 1-m POA; therefore, lead was not modeled further for the other scenarios. The only exceedance for PCBs (maximum concentration of 0.8 µg/L) was observed at the 1-m POA for the 100,000-year time frame under the No Action scenario. This maximum concentration is considered very conservative (higher than expected) since the location Building 235-F PCB inventory is limited to only the PuFF cells in the models when the PCB inventory is actually distributed throughout entire building. As such, further analysis was not warranted. All removal action scenarios considered in the RSER/EE/CA for the 0-10,000-year time frame are considered protective of groundwater and surface water in regard to PCBs and lead.

B.5.2. Grout First Level Scenario

The Grout First Level Scenario was not considered to be protective of groundwater. The plume emanating from ABL is predicted to exceed the gross alpha MCL (15 pCi/L) at the 360-m POA within 10,000 years.

Exceedances are predicted to occur in the plume emanating from the PuFF cells (Table B-1). At the 1-m POA, Pu-238 in the PuFF cells results in exceedances of gross alpha activity (89 pCi/L) and radium activity (30 pCi/L) before 10,000 years (Table B-2, Figure B-11). Maximum concentrations of gross alpha (3804 pCi/L), beta-gamma dose (46 mrem/yr), and radium (1,233 pCi/L) are predicted to occur around 32,000 years. Peak groundwater concentrations at the 100-m POA for gross alpha (415 pCi/L) and radium (135 pCi/L) do not exceed MCLs until after 10,000 years with a peak at 38,000 years. Peak groundwater concentrations at the 360-m POA for gross alpha (106 pCi/L) and radium (35 pCi/L) do not exceed MCLs until after 10,000 years with a peak at 47,000 years. Peak groundwater at the seepline POA for gross alpha (37 pCi/L) and radium (12 pCi/L) do not exceed MCLs until after 10,000 years with a peak at 60,000 years. Predicted concentrations in surface water in the unnamed UTR tributary are not significantly different than zero (<0.01).

Exceedances of gross alpha activity (368 pCi/L) and beta-gamma dose (5 mrem/yr) are also predicted before 10,000 years for the plume emanating from ABL at the 1-m POA (Table B-3). Maximum concentrations are predicted to occur at 5,700 years (Table B-4, Figure B-12). Exceedances in gross alpha are also predicted to occur at the 100-m POA (75 pCi/L), 360-m POA (21 pCi/L), and seepline POA (7 pCi/L) before 10,000 years with peak concentrations occurring at 5,800 years, 6,000 years and 6,100 years, respectively. Predicted concentrations in surface water in the unnamed UTR tributary are not significantly different than zero (< 0.01).

The grout the first building level scenario was not considered to be protective of groundwater within 10,000 years and not carried forward as a proposed removal action alternative in the RSER/EE/CA. As previously discussed, this scenario involved relocating the second level process equipment and ventilation ducts to the first level with no modifications to the roof which was considered a high risk to decommissioning workers.

B.5.3. Grout Entire Building 235-F Scenario

Grouting both the first and second levels of Building 235-F is considered protective of groundwater where the only exceedance at the 360-m POA occur after 10,000 years.

Exceedances are predicted to occur in the plume emanating from the PuFF cells (Table B-1). At all groundwater POAs, exceedances occur after 10,000 years. Peak groundwater concentrations at the 1-m POA for gross alpha (4237 pCi/L) and radium (1,377 pCi/L) do not exceed MCLs until after 10,000 years with a peak at 42,000 years. Peak groundwater concentrations at the 100-m POA for gross alpha (449 pCi/L) and radium (146 pCi/L) do not exceed MCLs until after 10,000 years with a peak at 47,000 years. Peak groundwater concentrations at the 360-m POA for gross alpha (116 pCi/L) and radium (38 pCi/L) do not exceed MCLs until after 10,000 years with a peak at 55,000 years. Peak groundwater concentrations at the seepline POA for gross alpha (39 pCi/L) and radium (13 pCi/L) do not exceed MCLs until after 10,000 years with a peak at 69,000 years. Predicted concentrations in surface water in the unnamed UTR tributary are not significantly different than zero (< 0.01).

Exceedances of gross alpha activity (49 pCi/L) are predicted before 10,000 years for the plume emanating from ABL at the 1-m POA (Table B-3). Exceedances in gross alpha are also predicted to occur at the 100-m (20 pCi/L) POA after 10,000 years with peak concentrations occurring at 12,800 years (Table B-4, Figure B-12,). No exceedances are predicted at the 360-m POA, or seepline POA. Predicted concentrations in surface water in the unnamed UTR tributary are not significantly different than zero (<0.01).

Although the grout the entire building scenario was considered to be protective of groundwater within 10,000 years, this scenario was not carried forward as a proposed removal action alternative in the RSER/EE/CA. As previously discussed, the existing roof could not serve as a robust barrier for 750 years without the modifications or installation of an engineered roof.

B.5.4. Grout Entire Building 235-F with Engineered Roof Scenario

Grouting both the first and second levels of Building 235-F and adding a robust engineered roof is considered protective of groundwater where the only exceedance at the 360-m POA occurs after 10,000 years.

Exceedances are predicted to occur in the plume emanating from the PuFF cells (Table B-1). At all groundwater POAs, exceedances occur after 10,000 years. Peak groundwater concentrations at the 1-m POA for gross alpha (4,463 pCi/L) and radium (1,456 pCi/L) do not exceed MCLs until after 10,000 years with a peak at 72,000 years. Peak groundwater concentrations at the 100-m POA for gross alpha (446 pCi/L) and radium (146 pCi/L) do not exceed MCLs until after 10,000 years with a peak at 76,000 years. Peak groundwater concentrations at the 360-m POA for gross alpha (114 pCi/L) and radium (37 pCi/L) do not exceed MCLs until after 10,000 years with a peak at 83,000 years. Peak groundwater concentrations at the seepline POA for gross alpha (36 pCi/L) and radium (12 pCi/L) do not exceed MCLs until after 10,000 years with a peak at 98,000 years. Beta-gamma dose is also predicted to exceed MCLs at the 1-m POA and 100-m POA. Predicted concentrations in surface water in the unnamed UTR tributary are not significantly different than zero (< 0.01).

The only predicted exceedance for the plume emanating from ABL is at the 1-m POA (Table B-3). An exceedance of gross alpha (55 pCi/L) is predicted to occur at the 1-m POA after 10,000 years with peak concentrations occurring at 22,000 years (Table B-4, Figure B-12). No exceedances are predicted at the 100-m POA, 360-m POA, or seepline POA. Predicted concentrations in surface water in the unnamed UTR tributary are not significantly different than zero (< 0.01).

This modeling scenario to grout both the first and second levels of Building 235-F and add a robust engineered roof supports both ISD removal action alternatives evaluated in the RSER/EE/CA. Differences between grouting the whole building (including the process areas) or grouting only the process areas would be undiscernible in the model results due to the relative percentage of contaminants in the process areas, the confined 2-D nature of the PORFLOW analysis, and the 1-D nature of the GoldSim model.

This page was intentionally left blank.

B.6 REFERENCES

ACRi, 2010. *PORFLOW Version 6.42.9 User's Manual*. Rev. 0, Analytical & Computational Research, Inc., Los Angeles, California, April 2018

Council, G., 2008. *Constituent Transport Analysis for the R Area Reactor Building (U)*, ERD-EN 2008-0060, Rev 0, GeoTrans, Inc., April 2009

Council, G., 2009. *Tier 2 Constituent Transport Analysis for the P-Area Reactor Building (U)*, ERD-EN-2008-0051, Rev-1, GeoTrans, Inc., September 2008

GTG, 2018. *GoldSim® User's Guide Version 12.1*. GoldSim® Technology Group. Issaquah, WA. June 2018

Schroeder, P., Lloyd, C. M., Zappi, P., and Aziz, N. 1994. *The Hydrologic Evaluation of Landfill Performance (HELP) Model User's Guide for Version 3*. EPA/600/R-94/168a. Office of Research and Development, United States Environmental Protection Agency (EPA), Cincinnati, Ohio. September 1994

SRNL, 2012. *Building 235-F GoldSim Fate and Transport Model*, SRNL-STI-2012-00504, Rev. 0, Savannah River National Laboratory, Savannah River Site, Aiken, SC 29808, September 14, 2012

SRNL, 2015. *Dose Calculation Methodology and Data for Solid Waste Performance Assessment and Composite Analysis at the Savannah River Site*, SRNL-STI-2015-00056, Rev. 0, Savannah River National Laboratory, Savannah River Site, Aiken, SC 29808, April 2015

SRNL, 2016. *Geochemical Data Package for Performance Assessment and Composite Analysis at the Savannah River Site – Supplemental Radionuclides*, SRNL-STI-2016-00267, Revision 0, Savannah River National Laboratory, Savannah River Site, Aiken, SC 29808, July 2016

SRNL, 2018. *Updated Groundwater Flow Simulations of the Savannah River Site General Separations Area*, SRNL-STI-2018-00643, Revision A, Savannah River National Laboratory, Savannah River Site, Aiken, SC 29808, November 2018.

SRNL, 2020. *Savannah River Site 235-F Facilities: Alternative End-State Analyses*, SRNL-TR-2019-00323, Rev. 1, Savannah River National Laboratory, Savannah River Site, Aiken, SC 29808, May 2021.

SRNS, 2020. *Characterization Report for Building 235-F, F-Area Material Storage Building (FAMS)*, G-ESR-F-00097, Rev. 0, May 2020, Savannah River Nuclear Solutions, LLC, Savannah River Site, Aiken, SC

USDOE, 2019. Letter, B. T. Hennessey (USDOE) to S. B. Fulmer (SCDHEC) and J. Richards (USEPA), *Building 235-F Regulatory Information Meeting (October 30, 2019)*, SEMS Number: 88, IACD-20-113, dated November 15, 2019

This page was intentionally left blank.

Table B-1. Peak Of The Mean Concentrations and Dose At Each POA for Various Timeframes for the Plume Emanating from Puff Facility (Pu-238)

Pu-238[1]		Gross Alpha (15 pCi/L)			Beta-Gamma (4 mrem/yr)			Radium (5 pCi/L)			Uranium (30 µg/L)		
		15	15	15	4	4	4	5	5	5	30	30	30

ISD Scenario	POA (m)	Alpha 0-1k	Alpha 0-10k	Alpha 0-100k	Beta-Gamma 0-1k	Beta-Gamma 0-10k	Beta-Gamma 0-100k	Radium 0-1k	Radium 0-10k	Radium 0-100k	Uranium 0-1k	Uranium 0-10k	Uranium 0-100k
No action	**1**	0	614	3474	0	7	42	0	204	1123	0	0	0
	100	0	38	388	0	0	5	0	13	126	0	0	0
	360	0	9	99	0	0	1	0	3	32	0	0	0
	Seepline (683)	0	1	36	0	0	0	0	0	12	0	0	0
	UTR Trib	0	0	0	0	0	0	0	0	0	0	0	0
Grout first level	**1**	0	89	3804	0	1	46	0	30	1233	0	0	0
	100	0	4	415	0	0	5	0	1	135	0	0	0
	360	0	1	106	0	0	1	0	0	35	0	0	0
	Seepline (683)	0	0	37	0	0	0	0	0	12	0	0	0
	UTR Trib	0	0	0	0	0	0	0	0	0	0	0	0
Grout entire 235-F	**1**	0	1	4237	0	0	51	0	0	1377	0	0	0
	100	0	0	449	0	0	5	0	0	146	0	0	0
	360	0	0	116	0	0	2	0	0	38	0	0	0
	Seepline (683)	0	0	39	0	0	0	0	0	13	0	0	0
	UTR Trib	0	0	0	0	0	0	0	0	0	0	0	0
Grout entire 235-F with engineered roof	**1**	0	0	4463	0	0	54	0	0	1456	0	0	0
	100	0	0	446	0	0	5	0	0	146	0	0	0
	360	0	0	114	0	0	2	0	0	37	0	0	0
	Seepline (683)	0	0	36	0	0	0	0	0	12	0	0	0
	UTR Trib	0	0	0	0	0	0	0	0	0	0	0	0

1- Exceedances are highlighted.

Table B-2. Peak Time (Years) for Each Performance Criteria for the Plume Emanating from the Puff Facility

ISD Scenario	POA (m)	Alpha Peak Time	Beta-Gamma Peak Time	Radium Peak Time	Uranium Peak Time
No action	**1**	25200	25010	24960	20010
	100	30810	30610	30560	25250
	360	39350	39660	38960	24650
	Seepline (683)	53460	53360	53250	46250
	UTR Trib	53460	53360	53250	46250
Grout first level	**1**	32660	32550	32500	26600
	100	38110	37960	37900	31810
	360	46850	47050	46550	31060
	Seepline (683)	60450	60350	60150	53050
	UTR Trib	60450	60350	60150	53050
Grout entire 235-F	**1**	41860	41760	41710	34250
	100	46850	46700	46610	39450
	360	54850	55360	54550	38850
	Seepline (683)	68550	68360	68260	60460
	UTR Trib	68550	68360	68260	60460
Grout entire 235-F with engineered roof	**1**	71760	71660	71650	59360
	100	76360	76250	76160	64660
	360	82760	83260	82550	61750
	Seepline (683)	97660	97650	97450	86560
	UTR Trib	97660	97650	97450	86560

Table B-3. Peak of the Mean Concentrations and Dose at Each POA for Various Timeframes for the Plume Emanating from ABL (Np-237)

Np-237[1]		Gross Alpha (15 pCi/L)			Beta-Gamma (4 mrem/yr)			Radium (5 pCi/L)			Uranium (30 µg/L)		
		15	15	15	4	4	4	5	5	5	30	30	30
ISD	**POA**	**Alpha**			**Beta-Gamma**			**Radium**			**Uranium**		
Scenario	**(m)**	**0-1k**	**0-10k**	**0-100k**	**0-1k**	**0-10k**	**0-100k**	**0-1k**	**0-10k**	**0-100k**	**0-1k**	**0-10k**	**0-100k**
No action	**1**	75	1319	1319	1	18	18	0	0	0	0	0	0
	100	6	265	265	0	4	4	0	0	0	0	0	0
	360	0	69	69	0	1	1	0	0	0	0	0	0
	Seepline (683)	0	21	21	0	0	0	0	0	0	0	0	0
	UTR Trib	0	0	0	0	0	0	0	0	0	0	0	0
Grout first level	**1**	0	368	368	0	5	5	0	0	0	0	0	0
	100	0	75	75	0	1	1	0	0	0	0	0	0
	360	0	21	21	0	0	0	0	0	0	0	0	0
	Seepline (683)	0	7	7	0	0	0	0	0	0	0	0	0
	UTR Trib	0	0	0	0	0	0	0	0	0	0	0	0
Grout entire 235-F	**1**	0	49	99	0	1	1	0	0	0	0	0	0
	100	0	10	20	0	0	0	0	0	0	0	0	0
	360	0	3	6	0	0	0	0	0	0	0	0	0
	Seepline (683)	0	1	2	0	0	0	0	0	0	0	0	0
	UTR Trib	0	0	0	0	0	0	0	0	0	0	0	0
Grout entire 235-F with engineered roof	**1**	0	2	55	0	0	1	0	0	0	0	0	0
	100	0	0	11	0	0	0	0	0	0	0	0	0
	360	0	0	3	0	0	0	0	0	0	0	0	0
	Seepline (683)	0	0	1	0	0	0	0	0	0	0	0	0
	UTR Trib	0	0	0	0	0	0	0	0	0	0	0	0

1- Exceedances are highlighted.

Table B-4. Peak Time (Years) for Each Performance Criteria for the Plume Emanating from the ABL

ISD Scenario	POA (m)	Alpha Peak Time	Beta-Gamma Peak Time	Radium Peak Time	Uranium Peak Time
No action	**1**	1340	1340	0	19110
	100	1410	1410	0	24390
	360	1610	1610	0	41260
	Seepline (683)	1680	1680	0	47760
	UTR Trib	1680	1680	0	47760
Grout first level	**1**	5720	5720	0	24090
	100	5790	5790	0	29510
	360	6000	6000	0	46710
	Seepline (683)	6070	6070	0	53560
	UTR Trib	6070	6070	0	53560
Grout entire 235-F	**1**	12740	12740	0	31860
	100	12800	12800	0	37310
	360	12980	12980	0	54710
	Seepline (683)	13030	13030	0	61460
	UTR Trib	13030	13030	0	61460
Grout entire 235-F with engineered roof	**1**	21760	21760	0	55150
	100	21830	21830	0	60750
	360	23070	23070	0	78610
	Seepline (683)	22170	22170	0	85460
	UTR Trib	22170	22170	0	85460

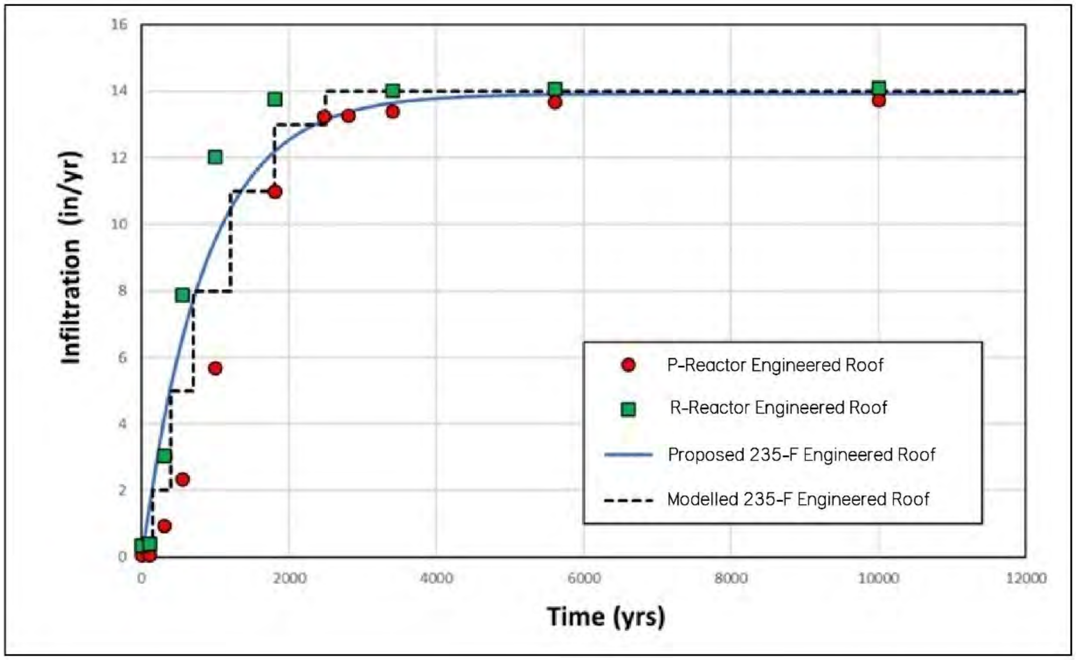


Figure B-1. Estimated Infiltration Rates for R- and P-Reactor Building Roof Designs Along with Values for a Fitted Roof Design Used in the Scenario "Grout Entire 235-F with Engineered Roof"

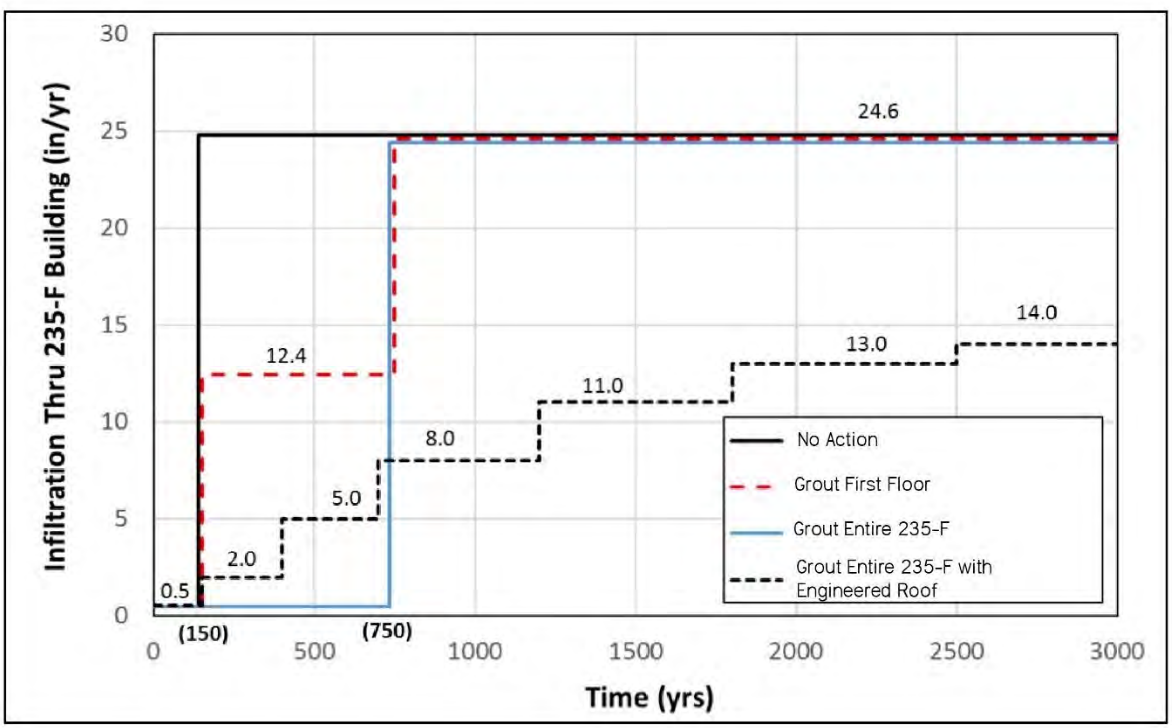


Figure B-2. Infiltration Rate Through Building 235-F Building for Each Modeling Scenario

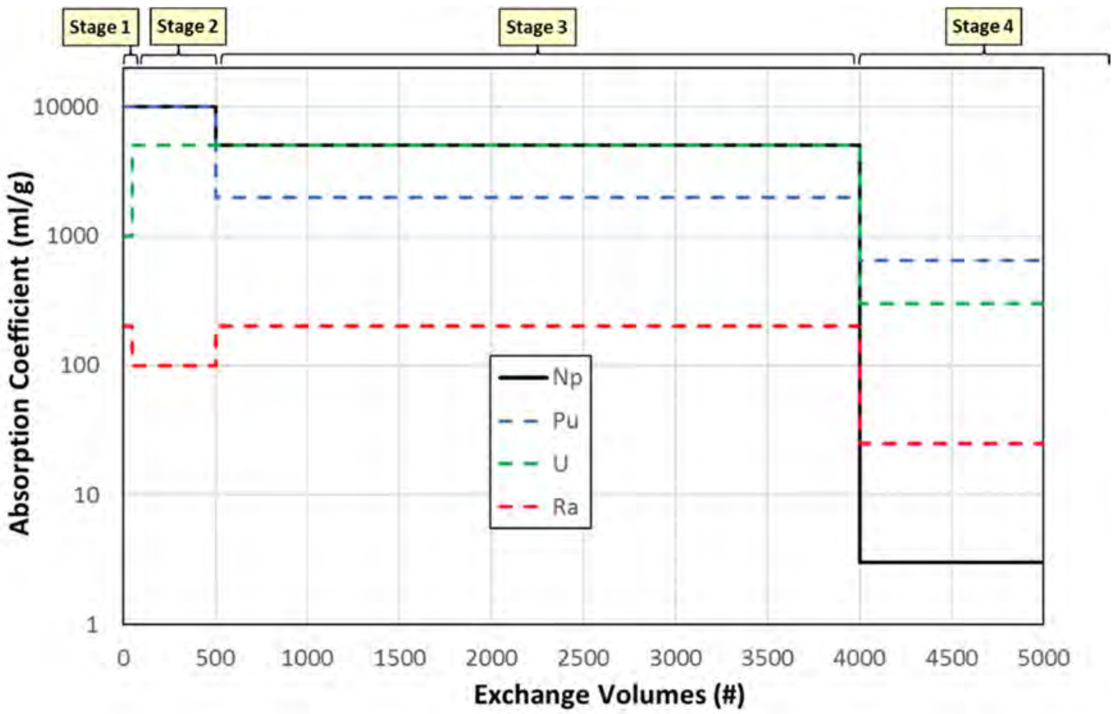


Figure B-3. Impact of Degradation on Cementitious Sorption Coefficient (K_d) Values

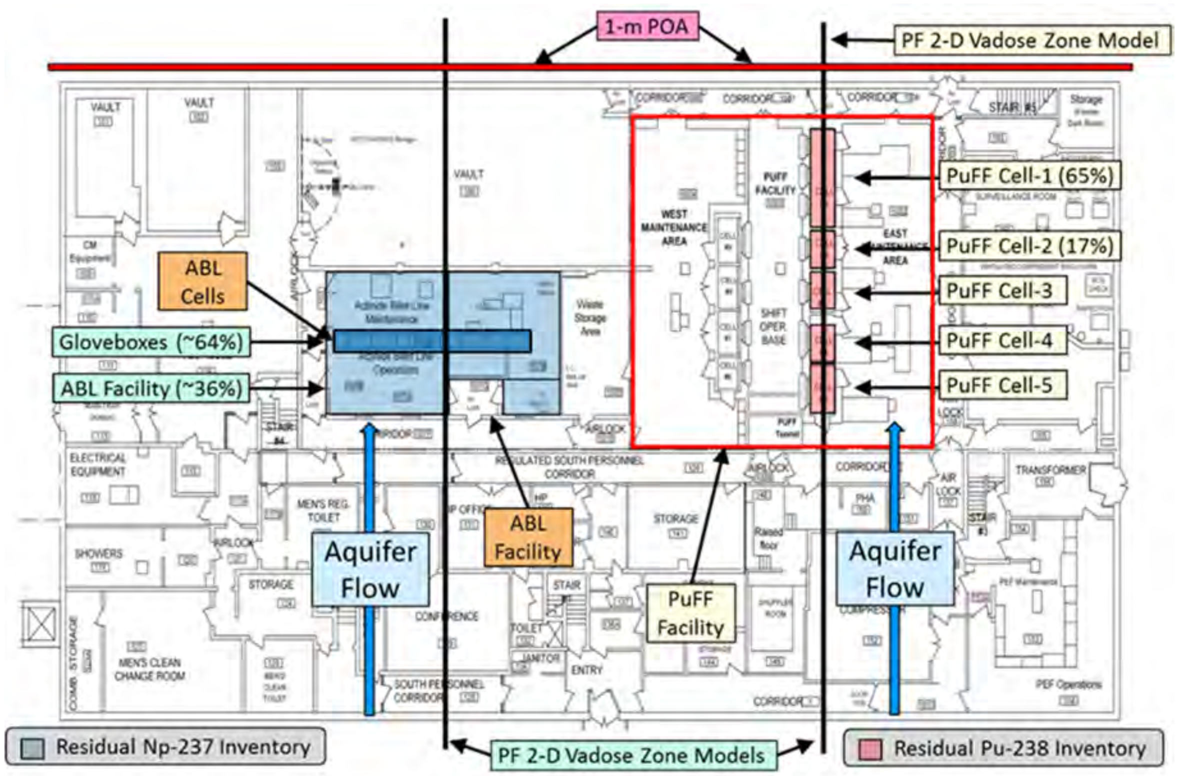


Figure B-4. Spatial Locations Where Residual Radiological Inventories Were Placed

Figure B-5. 100-M And 360-M POAs

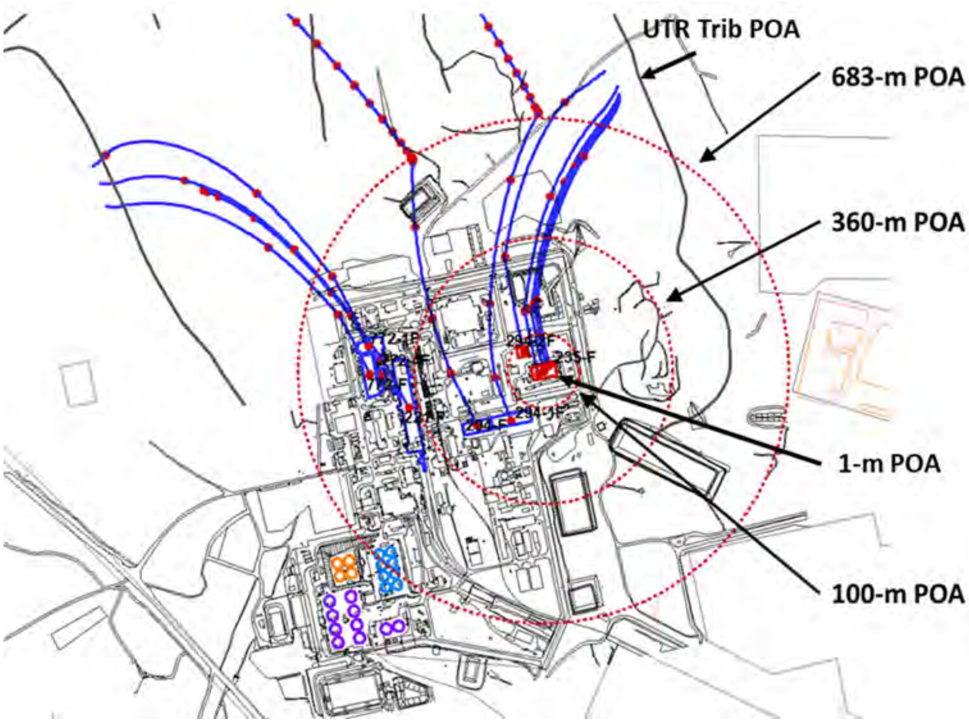


Figure B-6. Circles of Varying Radii to See Approximately Where POA Boundaries Reside

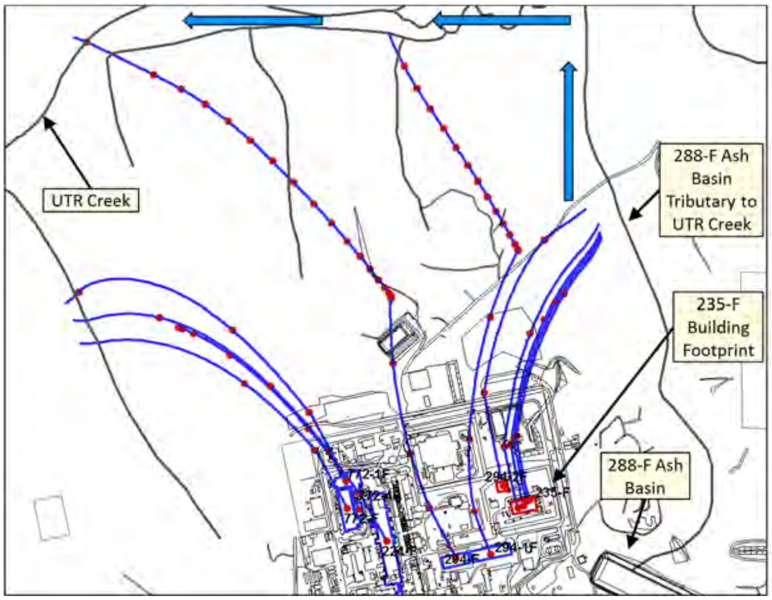


Figure B-7. 3-D Stream-Traces Emanating from Key F-Area Facilities

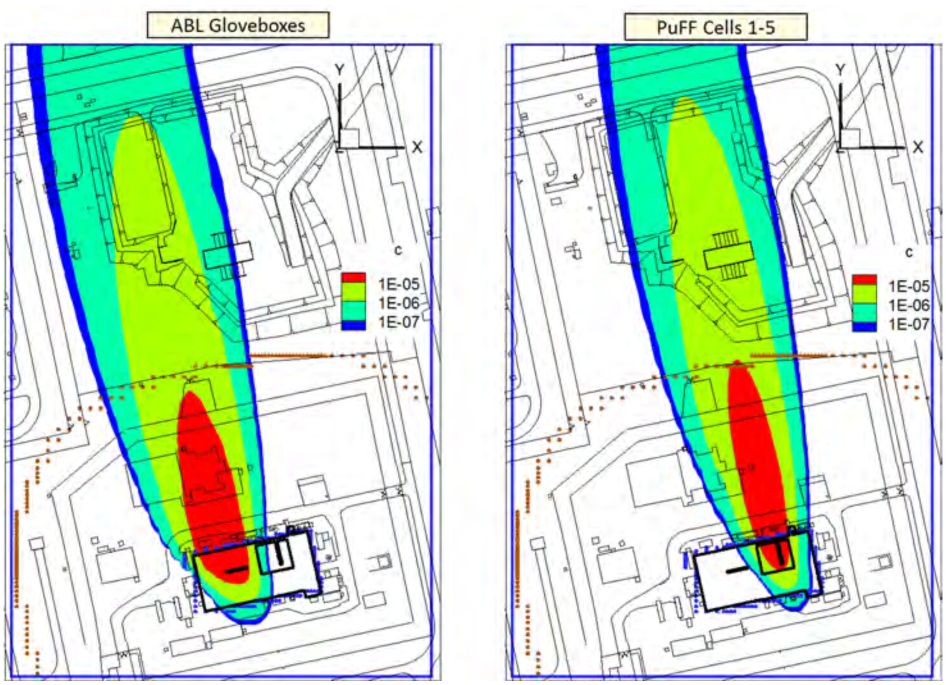


Figure B-8. PORFLOW-Based Steady-State Aquifer Tracer Plume Contours from Glovebox Source Terms Originating Within the ABL and Puff Facilities, Respectively

Blue Dots: 1-M POA. Orange Dots: 100-M POA

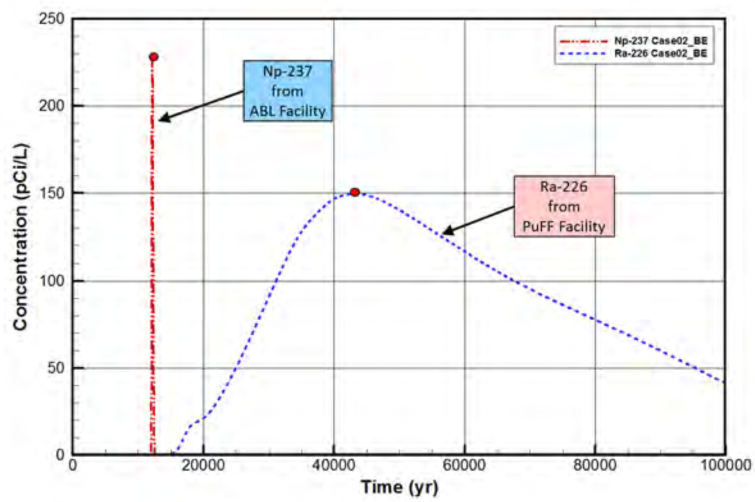


Figure B-9. PORFLOW-Based Predicted Concentrations at the 100-M POA Illustrating the Temporal Plume Separation in the Two Key Dose Contributors (Np-237 and Ra-226) Resulting from ABL and Puff, Respectively

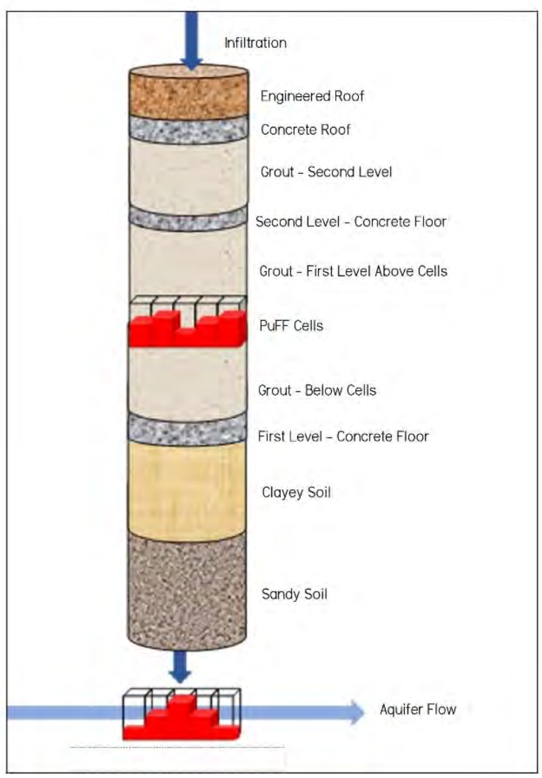


Figure B-10. Schematic Diagram (Not to Scale) of GoldSim Puff Facility and Vadose Zone Model

Note: Layers Above Puff Cells are Modeled Implicitly

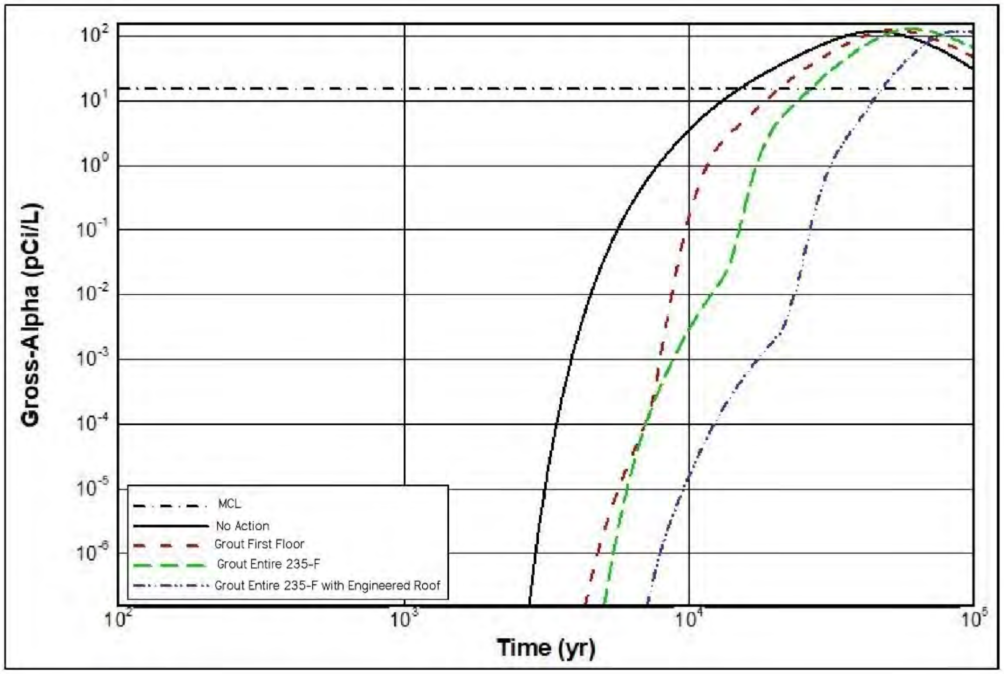


Figure B-11. Peak Timing for the Plume Emanating from the Puff Facility
Note: Concentrations Not Scaled Peaks and Figure Should Only Be Used for Timing

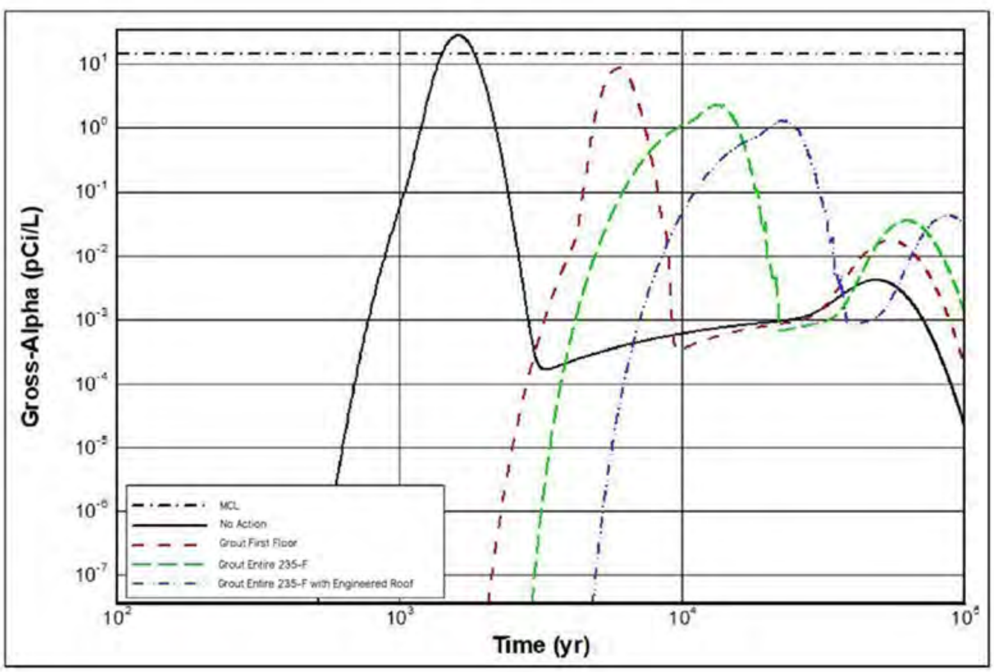


Figure B-12. Peak Timing for the Plume Emanating from the ABL
Note: Concentrations Not Scaled Peaks and Figure Should Only Be Used for Timing

APPENDIX C

Potential ARARs and TBC Criteria For Building 235-F

This page was intentionally left blank.

Table C-1. Potential ARARs and TBC Criteria for Building 235-F

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Activities potentially causing asbestos emissions	Discharge no visible emissions to the outside air during the collection, processing (including incineration), packaging and transporting of any asbestos-containing material generated by the source, or use one of the emission control and waste treatment methods specified in paragraphs (a)(1) through (4) of this section.	Owner or operator of any source covered under the provisions of § 61.145 *Standard for demolition and renovation* **– applicable (for Alternative A-4), or relevant and appropriate (for Alternatives A-2, A-3)**	40 CFR § 61.150(a)		✓	✓	✓
Emission control methods	Adequately wet asbestos-containing waste material as follows: • Mix control device asbestos waste to form a slurry; adequately wet other asbestos-containing waste material; and • Discharge no visible emissions to the outside air from collection, mixing, wetting, and handling operations, or use the methods specified by § 61.152 to clean emissions containing particulate asbestos material before they escape to, or are vented to, the outside air; and • After wetting, seal all asbestos-containing waste material in leak-tight containers while wet; or, for materials that will not fit into containers without additional breaking, put materials into leak-tight wrapping; and • Label the containers or wrapped materials specified in paragraph (a)(1)(iii) of this section using warning labels specified by Occupational Safety and Health Standards of the Department of Labor, Occupational Safety and Health Administration (OSHA) under 29 CFR 1910.1001(j)(4) or 1926.1101(k)(8). The labels shall be printed in letters of sufficient size and contrast so as to be readily visible and legible. • For asbestos-containing waste material to be transported off the facility site, label containers or wrapped materials with the name of the waste generator and the location at which the waste was generated.	Owner or operator of any source covered under the provisions of § 61.145 *Standard for demolition and renovation* **– applicable (for Alternative A-4), or relevant and appropriate (for Alternatives A-2, A-3)**	40 CFR § 61.150(a)(1)(i) – (v)		✓	✓	✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Emission control for asbestos-containing waste after demolition	Adequately wet the asbestos-containing waste material at all times after demolition and keep wet during handling and loading for transport to a disposal site. Asbestos-containing waste materials covered by this paragraph do not have to be sealed in leak-tight containers or wrapping but may be transported and disposed of in bulk.	Facilities demolished where RACM (as defined in 40 CFR § 61.141), is not removed prior to demolition according to §61.145(c)(1)(i)-(iv) or for facilities demolished according to § 61.145(c)(9) **– applicable**	40 CFR § 61.150(a)(3)				✓
Disposal of asbestos-containing waste material	All asbestos-containing waste material shall be deposited as soon as practicable by the waste generator at: • A waste disposal site operated in accordance with the provisions of § 61.154, or • An EPA-approved site that converts RACM and asbestos-containing waste material into nonasbestos (asbestos-free) material according to the provisions of § 61.155. • The requirements of paragraph (b) of this section do not apply to Category I nonfriable ACM that is not RACM.	Owner or operator of any source covered under the provisions of § 61.145 *Standard for demolition and renovation* **– applicable (for Alternative A-4), or relevant and appropriate (for Alternatives A-2, A-3)**	40 CFR § 61.150(b)(1)-(3)		✓	✓	✓
Pre-transport of asbestos-containing waste material	Mark vehicles used to transport asbestos-containing waste material during the loading and unloading of waste so that the signs are visible. The markings must conform to the requirements of §§ 61.149(d)(1)(i), (ii), and (iii).	Owner or operator of any source covered under the provisions of § 61.145 *Standard for demolition and renovation* **– applicable (for Alternative A-4), or relevant and appropriate (for Alternatives A-2, A-3)**	40 CFR § 61.150(c)		✓	✓	✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Inspection of facility for asbestos	Prior to the commencement of the demolition or renovation, thoroughly inspect the affected facility or part of the facility where the demolition or renovation operation will occur for the presence of asbestos, including Category I and Category II nonfriable ACM. The requirements of paragraphs (b) and (c) of § 61.145 apply to each owner or operator of a demolition or renovation activity, including the removal of RACM. NOTE: The *Notification requirements* of paragraph (b) of § 61.145 are considered "administrative' and therefore not identified as ARARs. However, some of the information included in the notice, for example a description of work to be performed and methods to be employed, work practices and engineering controls used to comply with the requirements of Subpart M, including asbestos removal and waste-handling emission control procedures should be included in the CERCLA decision document (e.g., ROD, Action Memorandum) and/or a subsequent Remedial Action or Removal Action Work Plan.	Demolition or renovation of a facility which may cause a disturbance of friable asbestos material and exceed the thresholds in 40 CFR 61.145(a)(1) **– applicable (for Alternative A-4), or relevant and appropriate (for Alternatives A-2, A-3)**	40 CFR § 61.145(a)		✓	✓	✓
RACM Thresholds	In a facility being demolished, all the requirements of paragraphs (b) and (c) of § 61.145 apply, except as provided in paragraph (a) of § 61.145, if the combined amount of RACM is (i) At least 80 linear meters (260 linear feet) on pipes or at least 15 square meters (160 square feet) on other facility components, or (ii) At least 1 cubic meter (35 cubic feet) of facility components where the length or area could not be measured previously. NOTE: The *Notification requirements* of paragraph (b) of § 61.145 are considered "administrative' and therefore not identified as ARARs.	Demolition of a facility which may cause a disturbance of friable asbestos material **– applicable**	40 CFR § 61.145(a)(1)				✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Procedures for asbestos emission control	Remove all RACM from a facility being demolished or renovated before any activity begins that would break up, dislodge, or similarly disturb the material or preclude access to the material for subsequent removal. RACM need not be removed before demolition if: (i) It is Category I nonfriable ACM that is not in poor condition and is not friable. (ii) It is on a facility component that is encased in concrete or other similarly hard material and is adequately wet whenever exposed during demolition; or (iii) It was not accessible for testing and was, therefore, not discovered until after demolition began and, as a result of the demolition, the material cannot be safely removed. If not removed for safety reasons, the exposed RACM and any asbestos-contaminated debris must be treated as asbestos-containing waste material and adequately wet at all times until disposed of. (iv) They are Category II nonfriable ACM and the probability is low that the materials will become crumbled, pulverized, or reduced to powder during demolition.	Demolition or renovation of a facility which may cause a disturbance of friable asbestos material and exceed the thresholds in 40 CFR 61.145(a)(1) **– applicable (for Alternative A-4), or relevant and appropriate (for Alternatives A-2, A-3)**	40 CFR § 61.145(c)(1)(i)-(iv)		✓	✓	✓
	When a facility component that contains, is covered with, or is coated with RACM is being taken out of the facility as a unit or in sections: (i) Adequately wet all RACM exposed during cutting or disjoining operations; and (ii) Carefully lower each unit or section to the floor and to ground level, not dropping, throwing, sliding, or otherwise damaging or disturbing the RACM.	Demolition or renovation of a facility which may cause a disturbance of friable asbestos material and exceed the thresholds in 40 CFR 61.145(a)(1) - **applicable (for Alternative A-4), or relevant and appropriate (for Alternatives A-2, A-3)**	40 CFR § 61.145(c)(2)		✓	✓	✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Procedures for asbestos emission control *con't*	When RACM is stripped from a facility component while it remains in place in the facility, adequately wet the RACM during the stripping operation.		40 CFR § 61.145(c)(3)		✓	✓	✓
	Component shall be stripped or contained in leak-tight wrapping, except as described in § 61.145(c)(5). If stripped, either: (i) Adequately wet the RACM during stripping; or (ii) Use a local exhaust ventilation and collection system designed and operated to capture the particulate asbestos material produced by the stripping. The system must exhibit no visible emissions to the outside air or be designed and operated in accordance with the requirements in § 61.152.	A facility component covered with, coated with RACM (as defined in 40 CFR § 61.141), taken out of the facility as a unit or in sections pursuant to 40 CFR § 61.145(c)(2) **– applicable (for Alternative A-4), or relevant and appropriate (for Alternatives A-2, A-3)**	40 CFR § 61.145(c)(4)(i) and (ii)		✓	✓	✓
	The RACM is not required to be stripped if the following requirements are met: (i) The component is removed, transported, stored, disposed of, or reused without disturbing or damaging the RACM. (ii) The component is encased in a leak-tight wrapping. (iii) The leak-tight wrapping is labeled according to § 61.149(d)(1)(i), (ii), and (iii) during all loading and unloading operations and during storage.	Large facility components such as reactor vessels, large tanks, and steam generators, but not beams containing RACM (as defined in 40 CFR § 61.141) **– applicable (for Alternative A-4), or relevant and appropriate (for Alternatives A-2, A-3)**	40 CFR § 61.145(c)(5)(i)-(iii)		✓	✓	✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Requirements for RACM (i.e., removed or stripped)	For all RACM, including material that has been removed or stripped: (i) Adequately wet the material and ensure that it remains wet until collected and contained or treated in preparation for disposal in accordance with § 61.150; and (ii) Carefully lower the material to the ground and floor, not dropping, throwing, sliding, or otherwise damaging or disturbing the material. (iii) Transport the material to the ground via leak-tight chutes or containers if it has been removed or stripped more than 50 feet above ground level and was not removed as units or in sections. (iv) RACM contained in leak-tight wrapping that has been removed in accordance with paragraphs (c)(4) and (c)(3)(i)(B)(*3*) of § 61.145 need not be wetted.	Generation of RACM (as defined in 40 CFR § 61.141), from demolition or renovation of a facility – **applicable (for Alternative A-4), or relevant and appropriate (for Alternatives A-2, A-3)**	40 CFR § 61.145(c)(6)(i)-(iv)		✓	✓	✓
Removal of RACM in freezing temperatures	The owner or operator need not comply with paragraph § 61.145(c)(2)(i) and the wetting provisions of § 61.145(c)(3). Shall remove facility components containing, coated with, or covered with RACM as units or in sections to the maximum extent possible. NOTE: Under § 61.145(c)(7)(iii), must record the temperature in the area containing the facility components at the beginning, middle and end of each workday and keep daily temperature records available for inspection. Recordkeeping requirements are generally considered “administrative’ and therefore not identified as ARARs.	Removal of RACM (as defined in40 CFR § 61.141), when the temperature at the point of wetting is below 0˚C (32˚F) **– applicable (for Alternative A-4), or relevant and appropriate (for Alternatives A-2, A-3)**	of § 61.145(c)(7)(i)-(ii)		✓	✓	✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Warning signs for disposal site	Display warning signs at all entrances and at intervals of 100m (328 feet) or less along the property line of the site or along the perimeter of the sections of the site where asbestos-containing waste material was deposited.	Closure of an area that received asbestos- containing waste materials that does not include a natural barrier to adequately deter access by the general public **– relevant and appropriate**	40 CFR § 61.151(b)(1)		✓	✓	✓
	The warning signs must: (i) Be posted in such a manner and location that a person can easily read the legend; and (ii) Conform to the requirements for (20”x14”) upright format signs specified in 29 CFR 1910.145(d)(4) and this paragraph; and (iii) Display the legend as prescribed in § 61.151(b)(1)(iii) located in the lower panel with letter sizes and styles of visibility at least equal to those specified in § 61.151(b)(1)(iii).	Closure of an area that received asbestos- containing waste materials that does not include a natural barrier to adequately deter access by the general public **– relevant and appropriate**	40 CFR § 61.151(b)(1)(i)-(iii)		✓	✓	✓
Management of PCB waste (e.g., contaminated PPE, equipment, wastewater)	Any person storing or disposing of PCB waste must do so in accordance with 40 CFR 761, Subpart D.	Generation of waste containing PCBs at concentrations ≥ 50 ppm **– applicable**	40 CFR 761.50(a)				✓
Temporary storage of PCB waste (e.g., PPE, rags) in a container(s)	Container(s) shall be marked as illustrated in 40 CFR 761.45(a).	Storage of PCBs and PCB Items at concentrations ≥ 50 ppm for disposal **– applicable**	40 CFR 761.40(a)(1)				✓
	Storage area must be properly marked as required by 40 CFR 761.40(a)(10).		40 CFR 761.65(c)(3)				✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
	Container(s) shall be in accordance with requirements set forth in DOT HMR at 49 CFR 171-180.		40 CFR 761.65(c)(6)				✓
Storage of PCB waste and/or PCB/radioactive waste in non-RCRA regulated unit	Storage facility must have or be: • Adequate roof and walls to prevent rainwater from reaching stored PCBs and PCB items;	Storage of PCBs and PCB Items at concentrations ≥ 50 ppm for disposal – **applicable**	40 CFR 761.65(b)(1) 40 CFR 761.65(b)(1)(i)				✓
	• Adequate floor that has continuous curbing with a minimum 6-inch high curb. Floor and curb must provide a containment volume equal to at least two times the internal volume of the largest PCB article or container or 25% of the internal volume of all articles or containers stored there, whichever is greater. *Note*: 6 inch minimum curbing not required for area storing PCB/radioactive waste;	Storage of PCB/radioactive waste as defined in 40 CFR 761.3 – **applicable**	40 CFR 761.65(b)(1)(ii)				✓
	• No drain valves, floor drains, expansion joints, sewer lines, or other openings that would permit liquids to flow from curbed area;		40 CFR 761.65(b)(1)(iii)				✓
	• Floors and curbing constructed of Portland cement, concrete, or a continuous, smooth, non-porous surface that prevents or minimizes penetration of PCBs; and		40 CFR 761.65(b)(1)(iv)				✓
	• Not located at a site that is below the 100-year flood water elevation.		40 CFR 761.65(b)(1)(v)				✓
	• Storage area must be properly marked as required by 40 CFR 761.40(a)(10).		40 CFR 761.65(c)(3)				✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Storage of PCB waste and/or PCB/radioactive waste in a RCRA-regulated container storage area	Does not have to meet storage unit requirements in 40 CFR 761.65(b)(1) provided unit: • is permitted by EPA under RCRA §3004, or • qualifies for interim status under RCRA §3005; or • is permitted by an authorized state under RCRA §3006 and, • PCB spills cleaned up in accordance with Subpart G of 40 CFR 761.	Storage of PCBs and PCB Items designated for disposal **– applicable**	40 CFR 761.65(b)(2)(i)-(iv)				✓
Storage of PCB/radioactive waste in containers	For liquid wastes, containers must be nonleaking. For non-liquid wastes, containers must be designed to prevent buildup of liquids if such containers are stored in an area meeting the containment requirements of 40 CFR 761.65(b)(1)(ii); and	Storage of PCB/radioactive waste in containers other than those meeting DOT HMR performance standards **– applicable**	40 CFR 761.65(c)(6)(i)(A) 40 CFR 761.65(c)(6(i)(B)				✓
	For both liquid and non-liquid wastes, containers must meet all regulations and requirements pertaining to nuclear criticality safety.		40 CFR 761.65(c)(6)(i)(C)				✓
Storage of liquid PCBs in stationary containers (e.g., leachate in storage tank)	Storage containers can be larger than the containers specified in paragraph (c)(6) of 40 CFR 761.65 provided that: • The containers are designed, constructed, and operated in compliance OSHA standards, 29 CFR 1910.106 *Flammable and combustible liquids*. Before using these containers for storing PCBs, the design of the containers must be reviewed to determine the effect on the structural safety of the containers that will result from placing liquids with the specific gravity of PCBs into the containers. • Owner/operator shall prepare and implement a Spill Prevention Control and Countermeasure (SPCC) Plan as described in part 112 of this title. *NOTE*: Substantive requirements of an SPCC Plan will be contained in the CERCLA Work Plan.	Storage of liquid PCB in *stationary containers* other than those meeting DOT HMR performance standards at 49 CFR parts 171 through 180 **– applicable**	40 CFR 761.65(c)(7)(i) and (ii)				✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Disposal of PCB bulk product waste (e.g., building demolition debris) in solid waste landfill	May dispose of in a facility permitted, licensed, or registered by a State as a municipal solid waste or non-municipal non-hazardous waste landfill. Includes Plastics (such as plastic insulation from wire or cable; radio, television and computer casings; vehicle parts; or furniture laminates); preformed or molded rubber parts and components; applied dried paints, varnishes, waxes or other similar coatings or sealants; caulking; Galbestos; non-liquid building demolition debris; or non-liquid PCB bulk product waste from the shredding of automobiles or household appliances from which PCB small capacitors have been removed (shredder fluff).	PCB bulk product waste listed in 40 CFR 761.62(b)(1)(i) including non-liquid building debris **–applicable**	40 CFR 761.62(b)(1)				✓
	May dispose of in a facility permitted, licensed, or registered by a State as a municipal solid waste or non-municipal non-hazardous waste landfill.	Other PCB bulk product waste, sampled in accordance with the protocols set out in subpart R of this part, that leaches PCBs at <10 µg/L of water measured using a procedure used to simulate leachate generation – **applicable**	40 CFR 761.62(b)(1)(ii)				✓
	May dispose of in a facility permitted, licensed, or registered by a State to manage as a municipal solid waste subject to 40 CFR 258 or non-municipal non-hazardous waste subject to 40 CFR 257.5 thru 257.30 if: • the PCB bulk product waste is segregated from organic liquids disposed of in the landfill; and • leachate is collected from the landfill and monitored for PCBs.	Other PCB bulk product waste not meeting conditions of 40 CFR 761.62(b)(1) (*e.g.*, paper/felt gaskets contaminated by liquid PCBs) **– applicable**	40 CFR 761.62(b)(2) 40 CFR 761.62(b)(2)(i) and (ii)				✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Disposal of PCB bulk product waste in an off-site solid waste landfill	Must provide written notice to the facility 15 days in advance of the first shipment from the same disposal waste stream.	Disposal of PCB bulk product waste regulated under 40 CFR 761.62(b)(1) at a facility without PCB approval – **applicable**	40 CFR 761.62(b)(4)(i)				✓
	The notice shall state that the PCB bulk product waste may include components containing PCBs at ≥ 50 ppm based on analysis of the waste in the shipment or general knowledge of the waste stream (or similar material) which is known to contain PCBs at those levels, and the waste is known or presumed to leach < 10 υg/L PCBs.						✓
	Must provide written notice to the facility 15 days in advance of the first shipment from the same disposal waste stream and with each shipment thereafter.	Disposal of PCB bulk product waste regulated under 40 CFR 761.62(b)(2) at a facility without PCB approval – **applicable**	40 CFR 761.62(b)(4)(ii)				✓
	The notice shall state that the PCB bulk product waste may include components containing PCBs at ≥50 ppm based on analysis of the waste in the shipment or general knowledge of the waste stream (or similar material) which is known to contain PCBs at those levels, and the waste is known or presumed to leach < 10 υg/L PCBs.						✓
Performance-based disposal of PCB bulk product waste	May dispose of by one of the following: • in an incinerator approved under 40 CFR 761.70; • in a chemical waste landfill approved under 40 CFR 761.75; • in a hazardous waste landfill permitted by EPA under §3004 of RCRA or by authorized state under §3006 of RCRA; • under alternate disposal approved under 40 CFR 761.60(e); • in accordance with decontamination provisions of 40 CFR 761.79; or • in accordance with thermal decontamination provisions of 40 CFR 761.79(c)(6) for metal surfaces in contact with PCBs.	Disposal of PCB bulk product waste as defined in 40 CFR 761.3 – **applicable**	40 CFR 761.62(a) 40 CFR 761.62(a)(1)-(6)		✓	✓	✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Risk-based sampling, storage and/or disposal of PCB bulk product waste	May sample or dispose of bulk product waste in a manner other than prescribed in 40 CFR 761.62(a) or (b), or store bulk product waste in a manner other than prescribed in 40 CFR 761.65, if receive approval in writing from EPA Regional Administrator and EPA finds that the method will not pose an unreasonable risk of injury to human health or the environment. Each application must contain information indicating that, based on technical, environmental or waste specific characteristics or considerations, the proposed sampling, disposal or storage methods will not pose an unreasonable risk of injury to human health or the environment. NOTE: Appropriate information required in an application can be provided in a CERCLA document (e.g., EE/CA, Action Memo, FS, PP, or ROD) that is approved or issued by EPA.	Sampling, storage and/or disposal of PCB bulk product waste (as defined in 40 CFR 761.3) **– applicable**	40 CFR 761.62(c)		✓	✓	✓
Transportation of PCB wastes off-site	Must comply with the manifesting provisions at 40 CFR 761.207 through 218.	Relinquishment of control over PCB wastes by transporting, or offering for transport **– applicable**	40 CFR 761.207(a)				✓
Transportation of hazardous materials	Shall be subject to and must comply with all applicable provisions of the HMTA and DOT HMR at 49 CFR 171-180.	Any person who, under contract with an department or agency of the federal government, transports "in commerce," or causes to be transported or shipped, a hazardous material **– applicable**	49 CFR 171.1(c)				✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Characterization of *solid* waste	Must determine if solid waste is a hazardous waste using the following method: Should first determine if waste is excluded from regulation under 40 CFR 261.4; and	Generation of solid waste as defined in 40 CFR 261.2 – **applicable**	40 CFR 262.11(a) SC R.61-79 262.11*(a)*		✓	✓	✓
	Must determine if waste is listed as hazardous waste under 40 CFR Part 261.	Generation of solid waste which is not excluded under 40 CFR 261.4(a) – **applicable**	40 CFR 262.11(b) SC R.61-79 262.11*(b)*		✓	✓	✓
	Must determine whether the waste is (characteristic waste) identified in subpart C of 40 CFR Part 261 by either: 1. Testing the waste according to the methods set forth in subpart C of 40 CFR part 261, or according to an equivalent method approved by the Administrator under 40 CFR 260.21; or 2. Applying knowledge of the hazard characteristic of the waste in light of the materials or the processes used.	Generation of solid waste which is not excluded under 40 CFR 261.4(a) – **applicable**	40 CFR 262.11(c) SC R.61-79 262.11*(c)*		✓	✓	✓
	Must refer to Parts 261, 262, 264, 265, 266, 268, and 273 of Chapter 40 for possible exclusions or restrictions pertaining to management of the specific waste.	Generation of solid waste which is determined to be *hazardous* waste – **applicable**	40 CFR 262.11(d) SC R.61-79 262.11*(d)*		✓	✓	✓
Determinations for management of *hazardous* waste	Must determine each USEPA Hazardous Waste Number (waste code) applicable to the waste in order to determine the applicable treatment standards under 40 CFR 268 *et seq.* *Note:* This determination may be made concurrently with the hazardous waste determination required in Sec. 262.11 of this chapter.	Generation of hazardous waste for storage, treatment or disposal – **applicable**	40 CFR 268.9(a) SCR.61-79 268.9*(a)*		✓	✓	✓
	Must determine the underlying hazardous constituents [as defined in 40 CFR 268.2(i)] in the characteristic waste.	Generation of RCRA characteristic hazardous waste (and is not D001 non-wastewaters treated by CMBST, RORGS, or POLYM of Section 268.42 Table 1) for storage, treatment or disposal – **applicable**	40 CFR 268.9(a) SCR.61-79 268.9*(a)*		✓	✓	✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
	Must determine if the hazardous waste meets the treatment standards in 40 CFR 268.40, 268.45, or 268.49 by testing in accordance with prescribed methods or use of generator knowledge of waste. *Note:* This determination can be made concurrently with the hazardous waste determination required in 40 CFR 262.11.	Generation of hazardous waste for storage, treatment or disposal – **applicable**	40 CFR 268.7(a) SC R.61-79 268.7 *(a)* (1)		✓	✓	✓
Off-Site Disposal of Solid Waste	Shall ultimately dispose of solid waste at facilities and/or sites permitted or registered by the Department for processing or disposal of that waste stream.	Generation of solid waste intended for off-site disposal – **relevant and appropriate**	SC R.61-107.5(D)(3)		✓	✓	✓
Off-site Disposal of RCRA-*Hazardous* Waste in a Land-Based Unit	May be land disposed if it meets the requirements in the table "Treatment Standards for Hazardous Waste" at 40 CFR 268.40 before land disposal.	Land disposal, as defined in 40 CFR 268.2, of restricted RCRA waste – **applicable**	40 CFR §268.40(a) SC R.61-79 268.40*(a)*		✓	✓	✓
	All underlying hazardous constituents [as defined in 40 CFR 268.2(i)] must meet the Universal Treatment Standards (UTSs), found in 40 CFR 268.48 Table UTS prior to land disposal.	Land disposal of restricted RCRA characteristic wastes (D001-D043) not managed in a wastewater treatment system regulated under the Clean Water Act (CWA), that is CWA equivalent, or that is injected into a Class I nonhazardous injection well – **applicable**	40 CFR §268.40(e) SC R.61-79 268.40*(e)*		✓	✓	✓
	Must be treated according to the alternative treatment standards of 40 CFR 268.49(c) <u>or</u> Must be treated according to the UTSs [specified in 40 CFR 268.48 Table UTS] applicable to the listed and/or characteristic waste contaminating the soil prior to land disposal.	Land disposal, as defined in 40 CFR 268.2, of restricted hazardous soils – **applicable**	40 CFR §268.49(b) SC R.61-79 268.49*(b)*		✓	✓	✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
	To determine whether a hazardous waste identified in this section exceeds the applicable treatment standards of 40 CFR 268.40, the initial generator must test a sample of the waste extract or the entire waste, depending on whether the treatment standards are expressed as concentration in the waste extract or waste, or the generator may use knowledge of the waste. If the waste contains constituents (including underlying hazardous constituents [UHCs] in the characteristic wastes) in excess of the applicable UTS levels in 40 CFR 268.48, the waste is prohibited from land disposal, and all requirements of Part 268 are applicable, except as otherwise specified.	Land disposal of RCRA toxicity characteristic wastes (D004-D011) that are newly identified – **applicable**	40 CFR 268.34(f) SC R.61-79 268.34*(f)*		✓	✓	✓
Disposal of RCRA-*hazardous waste debris* in a land-based unit (i.e., landfill)	Must be treated prior to land disposal as provided in 40 CFR 268.45(a)(1)-(5) unless EPA determines under 40 CFR 261.3(f)(2) that the debris no longer contaminated with hazardous waste or the debris is treated to the waste-specific treatment standard provided in 40 CFR 268.40 for the waste contaminating the debris.	Land disposal, as defined in 40 CFR 268.2, of restricted RCRA-hazardous debris - **applicable**	40 CFR 268.45(a)				✓

Table C-1. Potential ARARs and TBC Criteria for Building 235-F *(Continued/End)*

Action	Requirements	Prerequisite	Citation	Alt A-1	Alt A-2	Alt A-3	Alt A-4
Transuranic Waste	Ensure that all USDOE radioactive waste is managed in a manner that is protective of worker and public health and safety, and the environment *Definition of Transuranic Waste. Transuranic waste is radioactive waste containing more than 100 nanocuries (3700 becquerels) of alpha-emitting transuranic isotopes per gram of waste, with half-lives greater than 20 years, except for:* *(1) High-level radioactive waste;* *(2) Waste that the Secretary of Energy has determined, with the concurrence of the Administrator of the Environmental Protection Agency, does not need the degree of isolation required by the 40 CFR Part 191 disposal regulations; or* *(3) Waste that the Nuclear Regulatory Commission has approved for disposal on a case-by-case basis in accordance with 10 CFR Part 61.*	Manage Plutonium bearing waste as TRU waste - **TBC**	DOE Order 5820.2A DOE M 435.1-1				✓

DOT = U.S. Department of Transportation
HMR = Hazardous Materials Regulations
HMTA = Hazardous Materials Transportation Act
PPE = personal protective equipment
RACM = regulated asbestos-containing material
Subpart M = National Emission Standard for Asbestos located at 40 CFR 61.140 *et.seq*.

> = greater than
< = less than
≥ = greater than or equal to
≤ = less than or equal to

APPENDIX D

Detailed Cost Analysis
for
Removal Action Alternatives for Building 235-F

This page was intentionally left blank.

Table D-1. Building 235-F Cost Analysis, Alternative A-1 No Action

Item	Quantity	Units	Unit Cost	Total Cost
Direct Capital Costs				
None	0	ea	$0	$0
Total Direct Capital Cost		(sum of * items)		**$0**
Indirect Capital Costs				
N/A				
Total Indirect Capital Cost				**$0**
Total Estimated Capital Cost				**$0**
Direct O&M Costs				
Annual Costs (during implementation)	-0.3%	discount rate for costs > 30 years duration[1]		
	1			
Surveillance and Maintenance of Deactivated 235-F		yr	$3,500,000	$3,500,000
Subtotal - Annual Costs				$3,500,000
Present Worth Annual Costs (-0.3% Discount Rate)				**$3,510,532**
Annual Costs	150 years S&M			
Surveillance and Maintenance of Deactivated 235-F **(which also includes operation of the E5 fan exhaust system, the Sand Filter House [292-2F], associated air compressors, and nuclear facility program costs)**	1	ea	$3,500,000	$3,500,000
Subtotal - Annual Costs				$3,500,000
Present Worth Annual Costs (-0.3% Discount Rate)				**$666,267,626**
Total Present Worth Direct O&M Cost				**$669,778,158**
Indirect O&M Costs				
Project/Admin Management	0%	of direct O&M		$0
Health & Safety	0%	of direct O&M		$0
Overhead	30%	of direct O&M		$200,933,447
Contingency	15%	of direct O&M		$100,466,724
Total Estimated Present Worth Indirect O&M Cost				**$301,400,171**
Total Estimated Present Worth O&M Cost				**$971,178,329**
TOTAL ESTIMATED COST				**$971,178,329**

1. Interest rate from OMB Circular No A-94 (November 2020)

Table D-2. Building 235-F Cost Analysis, Alternative A-2 In-situ Decommissioning of First and Second Level Process Areas/Engineered Roof

Item	Quantity	Units	Unit Cost	Total Cost
Direct Capital Costs				
Baseline Prep				
Assay strategy	400	hrs	$95.65	$38,260
Asbestos Sampling/Field Inspection	320	hrs	$95.65	$30,608
Hazardous Waste Management Plan	450	hrs	$95.65	$43,043
Project Controls	900	hrs	$95.65	$86,085
Procedures	800	hrs	$95.65	$76,520
Structural Mechanics	450	hrs	$95.65	$43,043
Drills	300	hrs	$95.65	$28,695
Revise Authorization Basis	8,952	hrs	$105.68	$946,047
Authorization Basis Implementation	4,545	hrs	$107.85	$490,178
Operational Readiness Review	10,000	hrs	$95.65	$956,500
Decommissioning Project Plan	600	hrs	$95.65	$57,390
Direct Work by Site Forces				
General Conditions	1	lump sum (ls)		$ 2,016,162
Zone 1 First Level Process Area	1	ls		$ 2,022,515
Zone 2 First Level Process Area	1	ls		$ 1,701,657
Zone 3 First Level Process Area	1	ls		$ 1,102,227
Zone 4 First Level Process Area	1	ls		$ 1,404,093
Zone 7 First Level Non-Process Area	1	ls		$ 527,769
Zone 12 First Level Non-Process Area	1	ls		$ 152,100
Zone 10 First Level Non-Process Area	1	ls		$ 613,889
Zone 5 Second Level Process Area	1	ls		$ 1,185,426
Zone 6 Second Level Process Area	1	ls		$ 719,426
Zone 8 Second Level Process and Non- Process Are	1	ls		$ 1,024,553
Zone 9 Second Level Non-Process Area	1	ls		$ 403,573
Zone 13 Second Level Non-Process Area	1	ls		$ 611,810
Zone 11 Second Level Non-Process Area	1	ls		$ 587,565
Roof Penetrations and Duct	1	ls		$ 2,967,906
Underground Storage Tank and Pipe Trench	1	ls		$ 98,363
Abandoned capped stack (293-F)	1	ls		$ 52,536
Craft Support	1	ls		$ 2,905,794
SRS Construction Equipment	1	ls		$ 1,002,377
Casual & Scheduled Overtime / 2nd Shift Differential	1	ls		$ 417,657
Construction NonManual Support	1	ls		$ 7,223,549
HP / RCO Support Cost / Decon	1	ls		$ 3,541,136
Subcontracted Work				
Roof - Subcontractor Installation	1	ls		$ 7,880,044
Subtotal - Direct Capital Cost				$42,958,496
Total Direct Capital Cost		(sum of * items)		**$42,958,496**

Table D-2. Building 235-F Cost Analysis, Alternative A-2 In-situ Decommissioning of First and Second Level Process Areas/Engineered Roof *(Continued/End)*

Indirect Capital Costs				
Engineering & Design	10%	of direct capital		$4,295,850
Design Management	5%	of direct capital		$2,147,925
Health & Safety	10%	of direct capital		$4,295,850
Overhead & Fee	50.0%	of direct capital		$21,479,248
Contingency	30%	of direct capital		$12,887,549
Total Indirect Capital Cost				**$45,106,420**
Total Estimated Capital Cost				**$88,064,916**
Direct O&M Costs				
Annual Costs (during implementation)	-0.3%	for costs > 30 years duration[1]		
Access Controls	1	yr	$750	$750
Subtotal - Annual Costs				$750
Present Worth Annual Costs (-0.3% Discount Rate)				**$752**
Annual Costs (Land Use Controls)	150	years O&M		
Access Controls	1	ea	$750	$750
Subtotal - Annual Costs				$750
Present Worth Annual Costs (-0.3% Discount Rate)				**$69,029**
Five Year Costs	30			
Roof Inspections	1	ea	$5,000	$5,000
Roof Maintenance	1	ea	$40,000	$40,000
Remedy Review	1	ea	$20,000	$20,000
Subtotal - Five Year O&M Costs				$65,000
Present Worth Annual Costs (-0.3% Discount Rate)				**$2,049,799**
Present Worth Five Year Costs				
Total Present Worth Direct O&M Cost				**$2,119,580**
Indirect O&M Costs				
Project/Admin Management	10%	of direct O&M		$211,958
Health & Safety	5%	of direct O&M		$105,979
Overhead & Fee	50.0%	of direct O&M		$1,059,790
Contingency	30%	of direct O&M		$635,874
Total Estimated Present Worth Indirect O&M Cost				**$2,013,601**
Total Estimated Present Worth O&M Cost				**$4,133,182**
TOTAL ESTIMATED COST				**$92,198,098**

1. Interest rate from OMB Circular No A-94 (November 2020)

Table D-3. Building 235-F Cost Analysis, Alternative A-3 In-situ Decommissioning of Entire Building/Engineered Roof

Item	Quantity	Units	Unit Cost	Total Cost
Direct Capital Costs				
Baseline Prep				
Assay strategy	400	hrs	$95.65	$38,260
Asbestos Sampling/Field Inspection	320	hrs	$95.65	$30,608
Hazardous Waste Management Plan	450	hrs	$95.65	$43,043
Project Controls	900	hrs	$95.65	$86,085
Procedures	800	hrs	$95.65	$76,520
Structural Mechanics	450	hrs	$95.65	$43,043
Drills	300	hrs	$95.65	$28,695
Revise Authorization Basis	8,952	hrs	$105.68	$946,047
Authorization Basis Implementation	4,545	hrs	$107.85	$490,178
Operational Readiness Review	10,000	hrs	$95.65	$956,500
Decommissioning Project Plan	600	hrs	$95.65	$57,390
Direct Work by Site Forces				
General Conditions	1	lump sum (ls)		$ 2,016,162
Zone 1 First Level Process Area	1	ls		$ 2,022,515
Zone 2 First Level Process Area	1	ls		$ 1,701,657
Zone 3 First Level Process Area	1	ls		$ 1,102,227
Zone 4 First Level Process Area	1	ls		$ 1,404,093
Zone 7 First Level Non-Process Area	1	ls		$ 527,769
Zone 12 First Level Non-Process Area	1	ls		$ 152,100
Zone 10 First Level Non-Process Area	1	ls		$ 2,115,380
Zone 5 Second Level Process Area	1	ls		$ 1,185,426
Zone 6 Second Level Process Area	1	ls		$ 719,426
Zone 8 Second Level Process and Non- Process Areas	1	ls		$ 1,024,553
Zone 9 Second Level Non-Process Area	1	ls		$ 412,319
Zone 13 Second Level Non-Process Area	1	ls		$ 610,070
Zone 11 Second Level Non-Process Area	1	ls		$ 1,569,263
Roof Penetrations and Duct	1	ls		$ 2,967,906
Underground Storage Tank and Pipe Trench	1	ls		$ 98,363
Abandoned capped stack (293-F)	1	ls		$ 52,536
Craft Support	1	ls		$ 3,149,082
SRS Construction Equipment	1	ls		$ 1,131,671
Casual & Scheduled Overtime / 2nd Shift Differential	1	ls		$ 471,530
Construction NonManual Support	1	ls		$ 7,675,888
HP / RCO Support Cost / Decon	1	ls		$ 3,739,535
Subcontracted Work				
Roof - Subcontractor Installation	1	ls		$ 7,880,044
Subtotal - Direct Capital Cost				$46,525,885
Total Direct Capital Cost		(sum of * items)		**$46,525,885**
Indirect Capital Costs				
Engineering & Design, Design Mgt	10%	of direct capital		$4,652,588
Design Mangement	5%	of direct capital		$2,326,294
Health & Safety	10%	of direct capital		$4,652,588
Overhead & Fee	50.0%	of direct capital		$23,262,942
Contingency	30%	of direct capital		$13,957,765
Total Indirect Capital Cost				**$48,852,179**
Total Estimated Capital Cost				**$95,378,064**

Table D-3. Building 235-F Cost Analysis, Alternative A-3 In-situ Decommissioning of Entire Building/Engineered Roof *(Continued/End)*

Indirect Capital Costs				
Engineering & Design, Design Mgt	10%	of direct capital		$4,652,588
Design Mangement	5%	of direct capital		$2,326,294
Health & Safety	10%	of direct capital		$4,652,588
Overhead & Fee	50.0%	of direct capital		$23,262,942
Contingency	30%	of direct capital		$13,957,765
Total Indirect Capital Cost				**$48,852,179**
Total Estimated Capital Cost				**$95,378,064**
Direct O&M Costs				
Annual Costs (during implementation)	0.7%	discount rate for costs > 30 years duration[1]		
Access Controls	1	yr	$750	$750
Subtotal - Annual Costs				$750
Present Worth Annual Costs (-0.3% Discount Rate)				**$752**
Annual Costs (Land Use Controls)	150	years O&M		
Access Controls	1	ea	$750	$750
Subtotal - Annual Costs				$750
Present Worth Annual Costs (-0.3% Discount Rate)				**$69,029**
Five Year Costs	30			
Roof Inspections	1	ea	$5,000	$5,000
Roof Maintenance	1	ea	$40,000	$40,000
Remedy Review	1	ea	$20,000	$20,000
Subtotal - Five Year O&M Costs				$65,000
Present Worth Annual Costs (-0.3% Discount Rate)				**$2,049,799**
Present Worth Five Year Costs				
Total Present Worth Direct O&M Cost				**$2,119,580**
Indirect O&M Costs				
Project/Admin Management	10%	of direct O&M		$211,958
Health & Safety	5%	of direct O&M		$105,979
Overhead & Fee	50.0%	of direct O&M		$1,059,790
Contingency	30%	of direct O&M		$635,874
Total Estimated Present Worth Indirect O&M Cost				**$2,013,601**
Total Estimated Present Worth O&M Cost				**$4,133,182**
TOTAL ESTIMATED COST				**$99,511,246**

1. Interest rate from OMB Circular No A-94 (November 2020)

Table D-4. Building 235-F Cost Analysis, Alternative A-4 Complete Building 235-F Removal/Soil Cover

Item	Quantity	Units	Unit Cost	Total Cost
Direct Capital Costs				
Baseline Prep				
Assay strategy	400	hrs	$95.65	$38,260
Asbestos Sampling/Field Inspection	320	hrs	$95.65	$30,608
Hazardous Waste Management Plan	450	hrs	$95.65	$43,043
Project Controls	900	hrs	$95.65	$86,085
Procedures	800	hrs	$95.65	$76,520
Structural Mechanics	450	hrs	$95.65	$43,043
Drills	300	hrs	$95.65	$28,695
Revise Authorization Basis	8,952	hrs	$105.68	$946,047
Authorization Basis Implementation	4,545	hrs	$107.85	$490,178
Operational Readiness Review	10,000	hrs	$95.65	$956,500
Decommissioning Project Plan	600	hrs	$95.65	$57,390
Deactivation/Removal of Process Enclosures and Contaminated Ducts				
OML (zone 5 and 6)	1	ea	$2,659,320	$2,659,320
PEF (zone 3)	1	ea	$8,555,640	$8,555,640
PUFF (zones 1 and 2)	1	ea	$19,466,820	$19,466,820
ABL (zone 4)	1	ea	$2,659,320	$2,659,320
Contaminated duct work of Zone 5&6	1	ea	$2,659,320	$2,659,320
Zone 8	1	ea	$2,659,320	$2,659,320
Zone 9	1	ea	$1,329,660	$1,329,660
Decontaminate Remainder of building	80,000	sqft	$10	$800,000
Demo Building 235-F and 293-F Stack	1	ea	$9,940,000	$9,940,000
Waste Disposal Costs				
Waste Boxes	18	ea	$10,000	$180,000
Low level	3,807	cubic m	$300	$1,142,100
Sanitary	1,342.70	cubic m	$60	$80,562
TRU	1,061	cubic m	$2,415	$2,562,315
Mixed Low Level	29.6	cubic m	$20,500	$606,800
Hazardous Waste	5	cubic m	$4,300	$21,500
Soil Cover Over Slab	3	acres	$268,000	$804,000
Grout Tank	20	cy	$5,000	$100,000
Subtotal - Direct Capital Cost				$59,023,046
Mobilization/Demobilization	10%	of subtotal direct capital		$5,902,305
Site Preparation/Site Restoration	55%	of subtotal direct capital		$32,462,675
Total Direct Capital Cost		(sum of * items)		**$97,388,025**

Table D-4. Building 235-F Cost Analysis, Alternative A-4 Complete Building 235-F Removal/Soil Cover *(Continued/End)*

Indirect Capital Costs				
Engineering & Design	10%	of direct capital		$9,738,803
Project/Construction Management	5%	of direct capital		$4,869,401
Health & Safety	10%	of direct capital		$9,738,803
Overhead & Fee	50%	of direct capital		$48,694,013
Contingency	30%	of direct capital		$29,216,408
Total Indirect Capital Cost				**$102,257,427**
Total Estimated Capital Cost				**$199,645,452**
Direct O&M Costs				
Annual Costs (during implementation)	0.7%	for costs > 30 years duration[1]		
Access Controls	1	yr	$750	$750
Subtotal - Annual Costs				$750
Present Worth Annual Costs (-0.5% Discount Rate)				**$754**
Annual Costs (Land Use Controls)	150	years O&M		
Access Controls	1	ea	$750	$750
Subtotal - Annual Costs				$750
Present Worth Annual Costs (0.7% Discount Rate)				**$69,029**
Five Year Costs	30			
Soil Cover Maintenance /Mowing	1	ea	$1,500	$1,500
Inspections/Sign Maintenance / Repairs (1 per year)	1	ea	$1,500	$1,500
Remedy Review	1	ea	$20,000	$20,000
Subtotal - Five Year O&M Costs				$23,000
Present Worth Annual Costs (-0.3% Discount Rate)				**$725,314**
Total Present Worth Direct O&M Cost				**$795,096**
Indirect O&M Costs				
Project/Admin Management	10%	of direct O&M		$79,510
Health & Safety	5%	of direct O&M		$39,755
Overhead & Fee	53.5%	of direct O&M		$425,377
Contingency	30%	of direct O&M		$238,529
Total Estimated Present Worth Indirect O&M Cost				**$783,170**
Total Estimated Present Worth O&M Cost				**$1,578,266**
TOTAL ESTIMATED COST				**$201,223,718**

1. Interest rate from OMB Circular No A-94 (November 2020)

This page was intentionally left blank.