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Dose Reconstruction  
Project for NIOSH**

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03/05/2024	04	Revision to separate site profile into an Introduction and five TBDs. Incorporated the latest boilerplate in the TBD. Expanded the descriptions of some of the facilities to provide additional information to dose reconstructors; added absorption types in the tables that represent the possibilities for the various radionuclides; eliminated Attachment A and incorporated the discussion and the tables in the main body. Expanded the discussion in Section 2.3.20 on tritium facilities to include additional history and process description; added a discussion of organically bound tritium and metal tritides. Revision to incorporate comments and guidance associated with issuance of two SEC classes implemented with Petition SEC-0103. Includes expanded details on major SRS operating areas. Replaced Attachment A with new list of radionuclides. Updated Special Exposure Cohort information in Section 2.1.3. Activity fractions for the radionuclides of concern were removed from the tables throughout, descriptive text and photographs were added for visualization, and production campaigns for the 300-M area were clarified. U-233 added to Table 2-28. Thorium specified as Th-232 in Tables 2-41, 2-47, 2-48, and 2-49. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by John M. Byrne and authored by James M. Mahathy.

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**ACRONYMS AND ABBREVIATIONS**

ABL	Actinide Billet Line
ABRWH	Advisory Board on Radiation and Worker Health
ADS	Analytical Development Section
AEC	U.S. Atomic Energy Commission
ANL	Argonne National Laboratory
ARU	Acid Recovery Unit
AWE	Atomic Weapons Employer
cfm	cubic feet per minute
CHTS	Chemical and Hydrogen Technology Section
Ci	curie
CLAB	Central Laboratory Facility
CMX	Corrosion Mock-up Experiment
cpm	counts per minute
CTW	construction trade worker
d	day
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
DW	Distillation of Water (a.k.a. DuPont Water)
DWPF	Defense Waste Processing Facility
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EPA	U.S. Environmental Protection Agency
ETF	Effluent Treatment Facility
F	fast (absorption type)
FAMS	F-Area Materials Storage Building
ft	feet
g	gram
gal	gallon
GP	general purpose
gpm	gallons per minute
GS	Girdler-Sulfide
HANM	H-Area New Manufacturing
HAOM	H-Area Old Manufacturing
HEPA	high-efficiency particulate air
HFDP	High Flux Demonstration Program
HHW	high-heat waste
HPA	Health Physics Area
HPD	Health Physics Department
HQ	headquarters
hr	hour
HW	heavy water
HWCTR	Heavy Water Components Test Reactor
in.	inch
ITP	in-tank precipitation

kg	kilogram
L	liter
LANL	Los Alamos National Laboratory
lb	pound
LBNL	Lawrence Berkeley National Laboratory
LHW	low-heat waste
LiAl	lithium-aluminum (alloy)
LLNL	Lawrence Livermore National Laboratory
LMF	code for manufactured lithium-aluminum-natural uranium alloy
M	moderate (absorption type)
MCi	megacurie
MeV	megavolt-electron, 1 million electron-volts
mg	milligram
mi	mile
min	minute
mL	milliliter
mol	mole
MPPF	Multi-Purpose Processing Facility
MW	megawatt
n	neutron
NIOSH	National Institute for Occupational Safety and Health
NRDC	Natural Resources Defense Council
NSR	New Special Recovery
NTG	nuclear test gauge
ORAU	Oak Ridge Associated Universities
ORAUT	ORAU Team
ORNL	Oak Ridge National Laboratory
PDP	Process Development Pile
PEF	Plutonium Experimental Facility
pH	negative log of the hydrogen ion concentration, a measure of alkalinity or acidity
ppm	parts per million
PSF	Plutonium Storage Facility
PuAl	plutonium-aluminum (alloy)
PuFF	<sup>238</sup> PuO <sub>2</sub> Fuel Form Facility
PUREX	plutonium-uranium extraction
RBOF	Receiving Basin for Offsite Fuel
RR	railroad
RRF	Resin Regeneration Facility
RW	Rework Water Unit
s	second
S	slow (absorption type)
SEC	Special Exposure Cohort
SP	Standard Pile
SPFF	Special Products Fabrication Facility
SRDB Ref ID	Site Research Database Reference Identification (number)
SRNL	Savannah River National Laboratory



SRR	Savannah River Remediation
SRS	Savannah River Site
SRTC	Savannah River Technology Center
SWDF	Solid Waste Disposal Facility
Sylcor	Sylvania Electric Products/Sylvania-Corning Nuclear Corporation
t	ton
TBD	technical basis document
TFCT	Thorium Fuel Cycle Technology
THOREX	thorium extraction
TNX	Semi-works Laboratory
TRU	transuranic
UAI	uranium-aluminum (alloy)
U.S.C.	<i>United States Code</i>
WCF	Waste Certification Facility
WSRC	Westinghouse/Washington Savannah River Company
wt%	weight percent
yr	year
γ	gamma ray
μCi	microcurie
§	section or sections

## 2.1 INTRODUCTION

Technical basis documents (TBDs) and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historical background information and guidance to assist in the preparation of dose reconstructions at particular U.S. Department of Energy (DOE) or Atomic Weapons Employer (AWE) facilities or categories of DOE or AWE facilities. They will be revised in the event additional relevant information is obtained about the affected DOE or AWE facility(ies), such as changing scientific understanding of operations, processes, or procedures involving radioactive materials. These documents may be used to assist NIOSH staff in the evaluation of Special Exposure Cohort (SEC) petitions and the completion of individual dose reconstructions under Part B of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA).

In this document the word “facility” is used to refer to an area, building, or group of buildings that served a specific purpose at a DOE or AWE facility. It does not mean nor should it be equated to an “AWE facility” or a “DOE facility.” The term “AWE facility” is defined in EEOICPA to mean “a facility, owned by an atomic weapons employer, that is or was used to process or produce, for use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining or milling.” 42 *United States Code* (U.S.C.) § 7384I(5). On the other hand, a DOE facility is defined as “any building, structure, or premise, including the grounds upon which such building, structure, or premise is located—(A) in which operations are, or have been, conducted by, or on behalf of, the [DOE] (except for buildings, structures, premises, grounds, or operations ... pertaining to the Naval Nuclear Propulsion Program); and (B) with regard to which the [DOE] has or had—(i) a proprietary interest; or (ii) entered into a contract with an entity to provide management and operation, management and integration, environmental remediation services, construction, or maintenance services.” 42 U.S.C. § 7384I(12). The DOE determines whether a site meets the statutory definition of an AWE facility and the U.S. Department of Labor (DOL) determines if a site is a DOE facility and, if it is, designates it as such.

Under EEOICPA, a Part B cancer claim for benefits must be based on an energy employee’s eligible employment and occupational radiation exposure at a DOE or AWE facility during the facility’s designated time period and location (i.e., a “covered employee with cancer”). After DOL determines that a claim meets the eligibility requirements under Part B of EEOICPA, DOL transmits the claim to NIOSH for a dose reconstruction. EEOICPA provides, among other things, guidance on eligible employment and the types of radiation exposure to be included in an individual dose reconstruction. Under EEOICPA, eligible employment at a DOE facility includes individuals who are or were employed by DOE and its predecessor agencies, as well as their contractors and subcontractors at the facility. 42 U.S.C. § 7384I(11). Also under EEOICPA, the types of exposure to be included in dose reconstructions for DOE employees are those radiation exposures incurred in the performance of duty. As such, NIOSH includes all radiation exposures received as a condition of employment at DOE facilities in its dose reconstructions for covered employees, which may include radiation exposures related to the Naval Nuclear Propulsion Program at DOE facilities, if applicable. This is because NIOSH does not determine the fraction of total measured radiation exposure at a DOE facility that is contributed by the Naval Nuclear Propulsion Program at the DOE facility during a specified period of time for inclusion in dose reconstruction.

NIOSH does not consider the following types of exposure as those incurred in the performance of duty as a condition of employment at a DOE facility. Therefore these exposures are not included in dose reconstructions for covered employees [NIOSH 2010a]:

- Background radiation, including radiation from naturally occurring radon present in conventional structures, and
- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons.

### 2.1.1 Purpose

This TBD describes the facilities and processes at the Savannah River Site (SRS) that supported the U.S. nuclear weapons program.

### 2.1.2 Scope

SRS operations played an important role in the U.S. nuclear weapons program [DOE 1997]. Processes included nuclear fuel fabrication, reactor operation, radiochemical processing, uranium recycling, plutonium production, neutron source production, and waste management. This TBD describes the various processes and facilities, years of operation, and radionuclides that were used and generated in these facilities. Tables are provided showing the years of operation of the various facilities, the radionuclides of concern, and an indication if they were significant for either external or internal exposure potentials. Attachment A contains a list of radionuclides that have been present at SRS along with decay information. This information will assist the dose reconstructors in the evaluation of worker dose from these processes using the methods in OCAS-IG-001, *External Dose Reconstruction Implementation Guideline* [NIOSH 2007], and OCAS-IG-002, *Internal Dose Reconstruction Implementation Guideline* [NIOSH 2002].

### 2.1.3 Special Exposure Cohort

#### **January 1, 1953, through September 30, 1972**

The Secretary of the U.S. Department of Health and Human Services has added the following class of SRS employees to the SEC [Sebelius 2012, p. 3]:

*All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Savannah River Site from January 1, 1953, through September 30, 1972, for a number of work days aggregating at least 250 work days, occurring either solely under this employment or in combination with work days within the parameters established for one or more other classes of employees included in the Special Exposure Cohort.*

The Secretary based this designation on the findings of NIOSH's SEC evaluation report, which found it is not feasible to estimate internal exposures with sufficient accuracy for all externally monitored employees from January 1, 1953, through December 31, 1957, whose records have dosimetry codes A, G, CMX, or TNX. Further, NIOSH found that it lacked sufficient internal thorium monitoring data or other data or methods to support bounding internal thorium doses for SRS workers who may have worked with thorium in the 700-Area or the Corrosion Mock-up Experiment (CMX) and Semi-works Laboratory (TNX) areas from January 1, 1958, through September 30, 1972, whose records have dosimetry codes 5A, 5C, 6B through 6Z, 12D through 12H, or 12J through 12Z [NIOSH 2011].

#### **October 1, 1972, through December 31, 1990**

The Secretary has also added the following class [Becerra 2021, p. 3]:

*All construction trade employees of Department of Energy subcontractors [excluding employees of the following prime contractors who worked at the Savannah River Site in Aiken, South Carolina, during the specified time periods: E. I. du Pont de Nemours and Company, October 1, 1972, through March 31, 1989; and Westinghouse Savannah River Company, April 1, 1989, through December 31, 1990], who worked at the Savannah River Site from October 1, 1972, through December 31, 1990, for a number of work days aggregating at least 250 work days, occurring either solely under this employment or in combination with work days within the parameters established for one or more other classes of employees included in the Special Exposure Cohort.*

The Secretary based this designation on the recommendation of the Advisory Board on Radiation and Worker Health (ABRWH), which found that dose reconstruction for unmonitored subcontractor construction trade workers (CTWs) who should have been monitored via the permit-driven job-specific monitoring program is not feasible using the co-exposure models NIOSH developed due to the nature of radiological work assigned to transient subcontractor CTWs, the lack of assurance provided their bioassay monitoring, and identified gaps in the permit-driven job-specific monitoring program. The ABRWH concluded that the completeness and representation of subcontractors who were, or should have been, monitored has not been sufficiently established [Anderson 2021].

Although the ABRWH found that it is not possible to completely reconstruct radiation doses for either class, NIOSH intends to use any internal and external monitoring data that might become available for an individual claim and that can be interpreted using existing NIOSH dose reconstruction processes or procedures to conduct partial dose reconstructions for employees who do not qualify for inclusion in the SEC [Anderson 2021; NIOSH 2011].

## 2.2 SITE ACTIVITIES AND PROCESSES

This section briefly describes the facilities and processes, the radionuclides associated with these processes, and the times of operation during which the potential for exposure of the workers existed. Dose reconstructors would use these data only when monitoring data are unavailable and/or other methods might not be amenable for dose reconstruction.

The site borders the Savannah River and covers about 310 mi<sup>2</sup> in the South Carolina counties of Aiken, Allendale, and Barnwell. SRS is about 12 mi south of Aiken, South Carolina, and 15 mi southeast of Augusta, Georgia [SRNS 2015, p. 33].

SRS was originally organized as nine manufacturing areas, a central administration area, and two service building areas. Each area was designated by a number indicating the function and a letter indicating the location. A list of the general areas is provided in Table 2-1. Building numbers generally align with the areas: 200s in F- and H- Areas, 300s in M-Area, etc. Area G was not a physical location but includes building numbers in the 600s that might be found in any area. In addition, buildings with numbers in the 700s might be found in any area [DOE 2005, pp. 23, 132].

Table 2-1. Area designations.

Primary process or function	Area(s)
100 – Reactors	C, K, L, P, and R
200 – Separations	F, H
300 – Fuel and Target Fabrication	M
400 – Heavy Water Production and Rework	D
500 – Infrastructure, 600 – General	G
700 – Administration	A, B
Pilot Plants/Research and Development	T
Heavy Water Components Test Reactor	U
Waste Management	E, S, Z, and J
Central Shops	N

The relative locations of these areas are shown on the SRS site map in Figure 2-1. A detailed map of each area is given in Westinghouse/Washington Savannah River Company (WSRC) [2001a].

At SRS, radioisotopes were created by neutron irradiation of target materials in fuel rod assemblies in the five production reactors. In the 313-M building, target and fuel elements of natural, depleted, or slightly enriched uranium were bonded in aluminum cans. Finished fuel and target assemblies were transported to site reactors and subjected to an irradiation cycle where plutonium and tritium were

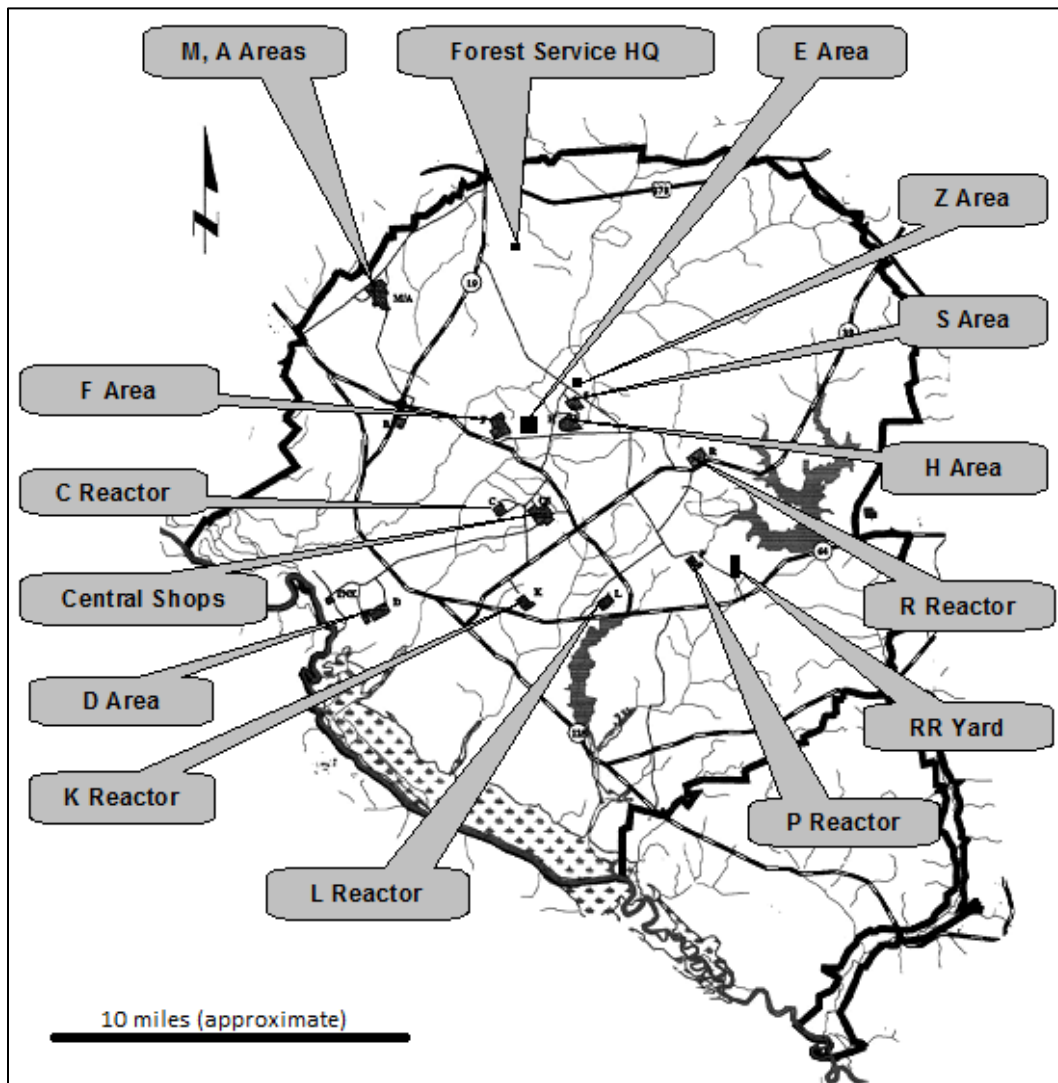


Figure 2-1. SRS site map [Rollins 2003, p. 12; SRNS 2015, p. 34]. HQ = headquarters; RR = railroad.

produced for nuclear weapons and small quantities of other radioisotopes were produced primarily for nonmilitary purposes. Fuel and target assemblies were remotely loaded and unloaded from the reactors from a crane control room. The exposed assemblies were handled underwater and stored underwater until their decay heat was low enough for shipment to the separation facilities [WSRC 2000a, pp. 14–17].

The fuel and targets were dissolved in nitric acid using remotely operated processes in the F- and H-Canyon buildings to allow chemical processing of the production radioisotopes, done primarily in F-Area. Uranium separation processes were indicated as “A” and plutonium as “B”; for example, the FB-Line is a plutonium separation line in F-Area. An overall process flow diagram is shown in Figure 2-2.

**Note: Information for dates of facilities known to be decontaminated or demolished are included in this document. For facilities not known to be decontaminated or demolished, assume they are currently contaminated.**

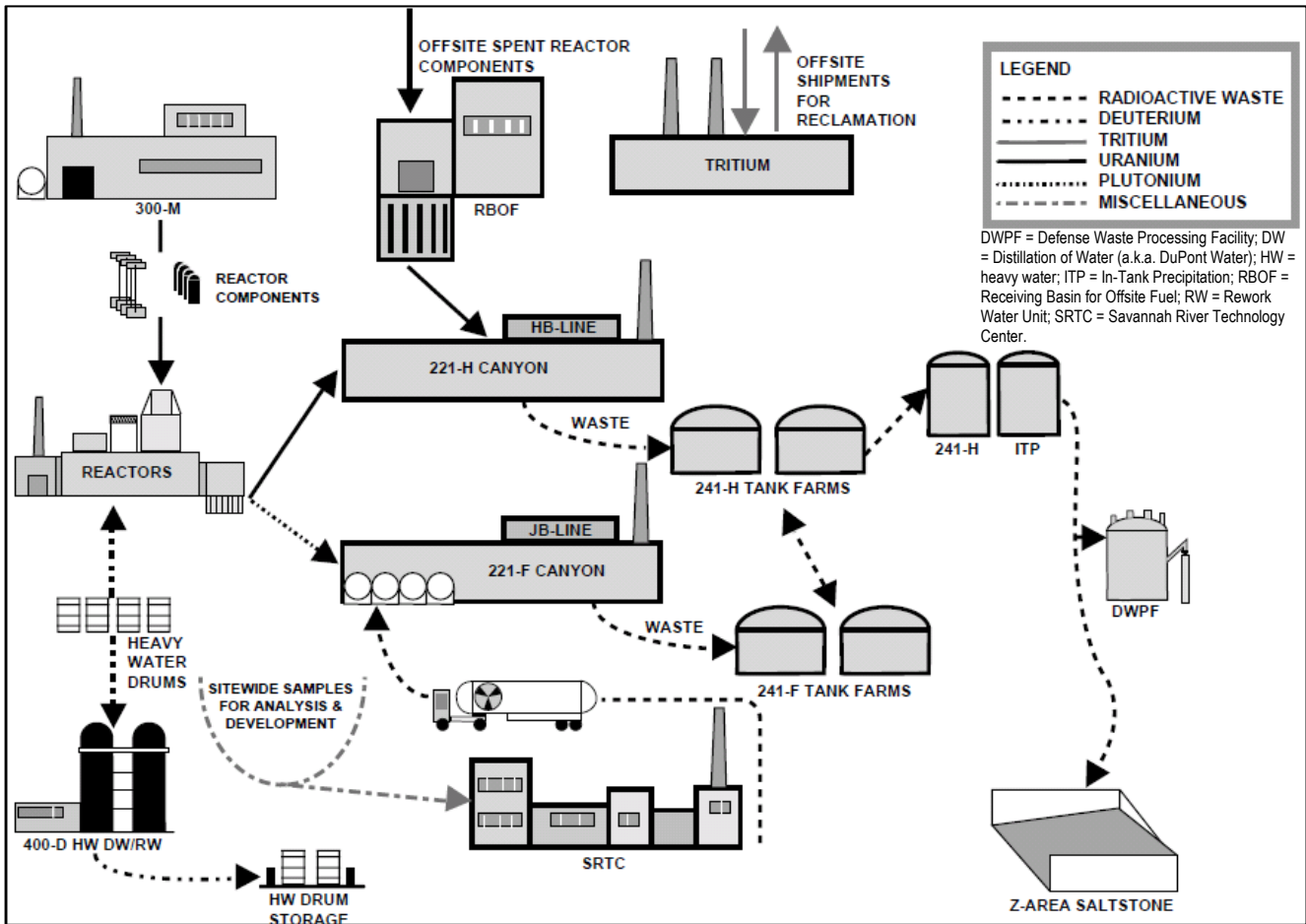


Figure 2-2. Process flow summary diagram [Till et al. 2001, p. 66].

**2.2.1 Reactors**

In the early 1950s, five heavy-water reactors (R, P, L, K, and C) were constructed at SRS. All five reactors were operating by 1955. Reactor shutdown began in 1964 with R Reactor. C Reactor was shut down in 1987; L Reactor was shut down in 1968, restarted in 1985, and shut down again in 1988. K and P Reactors operated with few interruptions until 1988. A restart of K Reactor began in 1991 but was not completed. Years of these reactor operations are summarized in Table 2-2 [Till et al. 2001, pp. 53, 67]. The location of each reactor is shown in Figure 2-1.

Table 2-2. Reactor years of operation.<sup>a</sup>

C	K	L	P	R
1955–1987	1954–1988	1954–1968 1985–1988	1954–1988	12/1953–1964

a. Source: Till et al. [2001, pp. 53, 67, 121].

R Reactor achieved operating status first, in December 1953, and was shut down permanently in 1964 when the demand for the weapons reactor products began to decrease [Reed 2010, p. 192].

P Reactor was started in February 1954 and shut down in August 1988. In February 1991, P Reactor was shut down permanently [Strack 2002, pp. 338, 519, 527].

L Reactor achieved operating status in August 1954 and was placed in cold standby in 1968 [Strack 2002, pp. 338, 435, 509]. It was restarted in 1985 after SRS made safety upgrades and run for

3 years [Strack 2002, pp. 338, 435, 508–511]. It was placed in warm standby in December 1991 to be put into operation as a backup to K Reactor, if necessary, but was subsequently shut down permanently.

C Reactor achieved operating status in March 1955 [Strack 2002, p. 338] and was shut down for maintenance in 1985 [WSRC 1997, p. 32]. It was placed in cold standby in 1987 when cracks were observed in the reactor vessel. It was subsequently placed in permanent shutdown status [DOE 1990, p. 113].

K Reactor achieved operating status in October 1954 [Strack 2002, p. 338] and was shut down in April 1988 [Strack 2002, p. 519]. Initial steps to restart K Reactor for tritium production began in December 1991. Successful power ascension testing was completed in July 1992. After ascension testing, the reactor was taken off line to allow for the tie-in of a new cooling tower. The tie-in was completed, and an operating permit was issued in December 1992. In 1993, the cooling tower was successfully tested, but the reactor was never restarted due to political pressures [Strack 2002, pp. 528–529]. K Reactor was then placed in cold standby, but the official status was changed to cold shutdown in 1996 [WSRC 1997, p. 32].

The SRS reactors differed substantially from the “graphite pile” reactors that had been previously built and operated at Hanford in that heavy water was used as both a neutron moderator and as the primary reactor coolant. Water from the Savannah River was used as the secondary coolant for all five of the reactors. The SRS reactor vessels were all made of stainless steel, about 16 ft in diameter, with precisely located holes at the top and corresponding pins at the bottom that held the reactor core lattice of fuel rods, target assemblies, control rods, safety rods, and instrument assemblies securely in place. Some of the deuterium in the heavy water was converted to radioactive tritium; other radioactive materials occasionally leaked from the fuel elements by way of defects in the cladding [Bebbington 1990, pp. 22–31; Reed 2010, pp. 143–150]. Accordingly, the operating crews in the reactor buildings were shielded from the high-level radiation in some of the reactor operating areas and protected against airborne radioactive materials. The workspaces in the reactor buildings that workers routinely occupied, such as the offices, control rooms, change rooms, and shops, had independent ventilation systems. The reactor process areas, including the fuel element assembly rooms and spent fuel storage basins, used once-through ventilation and were maintained at lower air pressures than the nonprocess building areas. The process area ventilation air was normally discharged through a series of air filters and up stacks approximately 200 ft tall [Reed 2010, pp. 14, 134–135, 169–172].

A sixth reactor, the Heavy Water Components Test Reactor (HWCTR), was constructed to test the concept of a heavy-water ( $D_2O$ ) moderated and cooled civilian reactor. SRS completed construction of HWCTR in 1962. The location of HWCTR was initially designated as Area U, but by the 1990s HWCTR was included as part of Area B. At first, HWCTR was under management of the D-Area Superintendent [DuPont 1962]. The reactor was placed in critical stage for the first time on March 3, 1962. SRS conducted a few zero-power tests through December 1964. The U.S. Atomic Energy Commission (AEC) decided to research organically cooled reactors at Idaho National Laboratory, which eliminated the need for HWCTR. It was closed in 1964 [Reed 2010, p. 55]. SRS removed all buildings and equipment. SRS grouted in situ the spent fuel pool and below-grade areas of the building [Freeman 2011].

A map of the layout of the 100-C Area (C Reactor) is shown in Figure 2-3; the portion of the map with buildings is magnified in Figure 2-4. The general building design for all reactor areas is shown in Figure 2-5. Each reactor building contained rooms and functions for assembly, processing, disassembly, purification, and personnel. The R Reactor is used as the example in discussing each functional room.

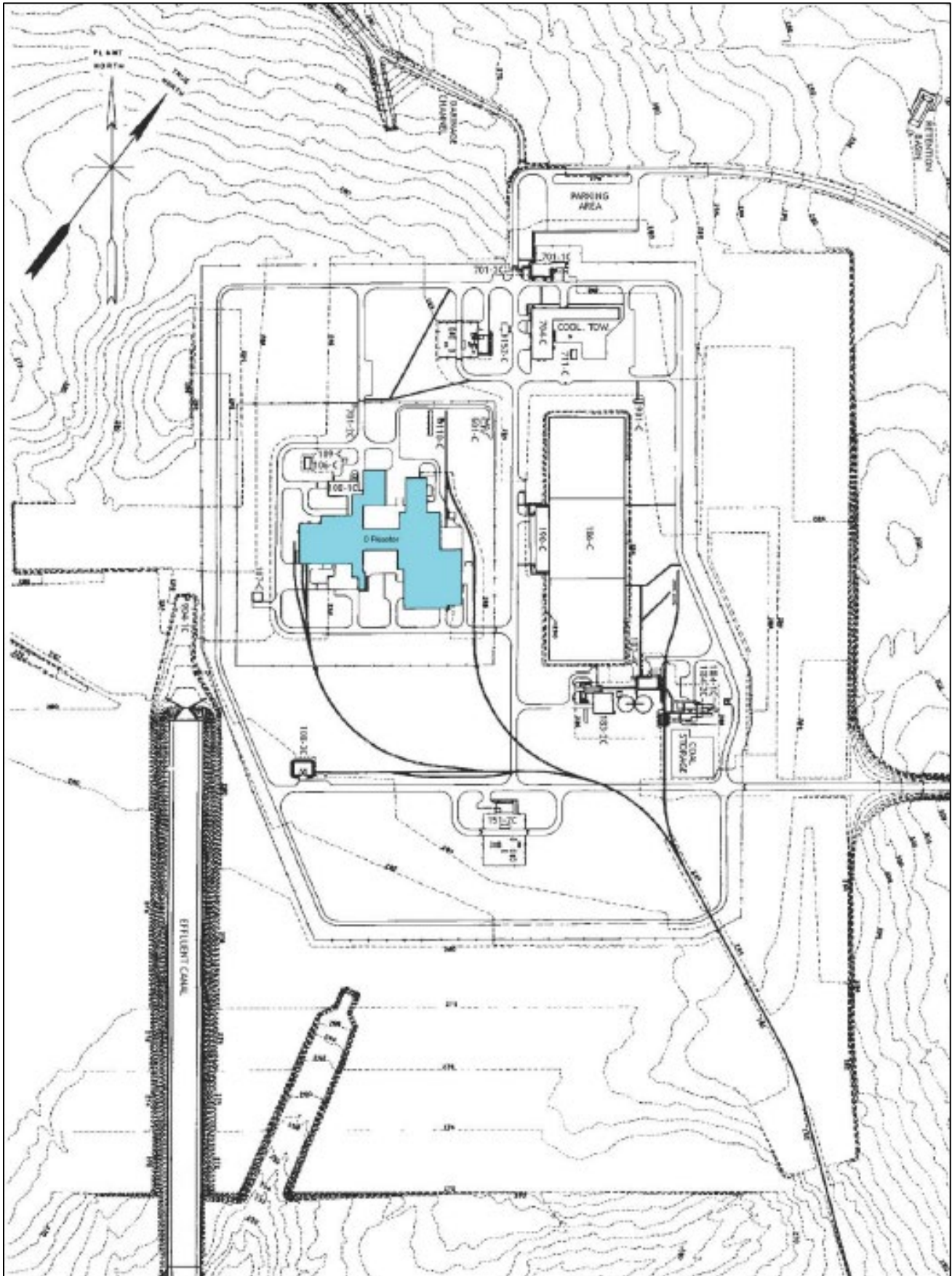


Figure 2-3. 100-C Area, general plan and building map [Reed 2010, p. 77].



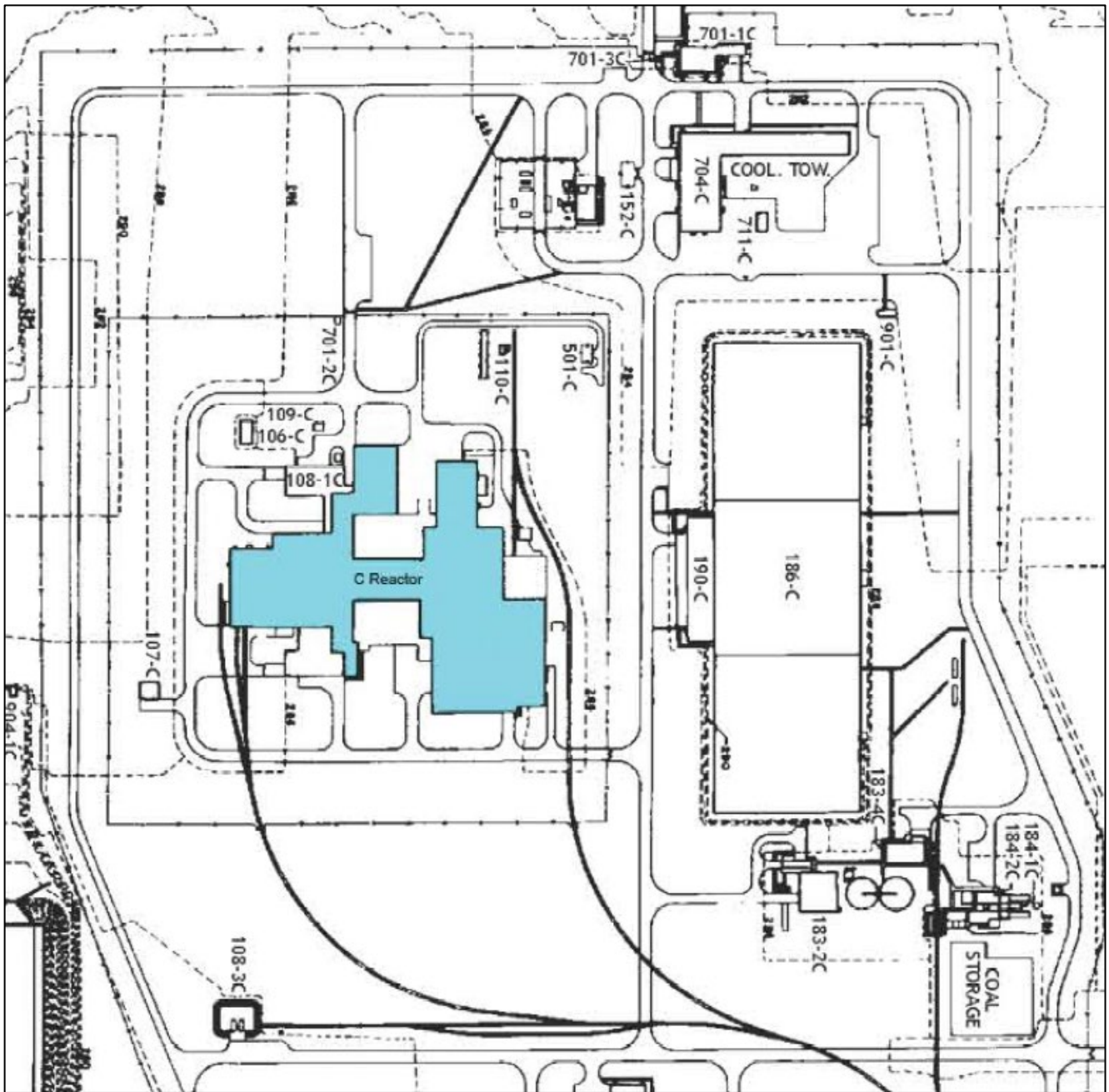


Figure 2-4. 100-C Area, closeup of building layout [Reed 2010, p. 77].

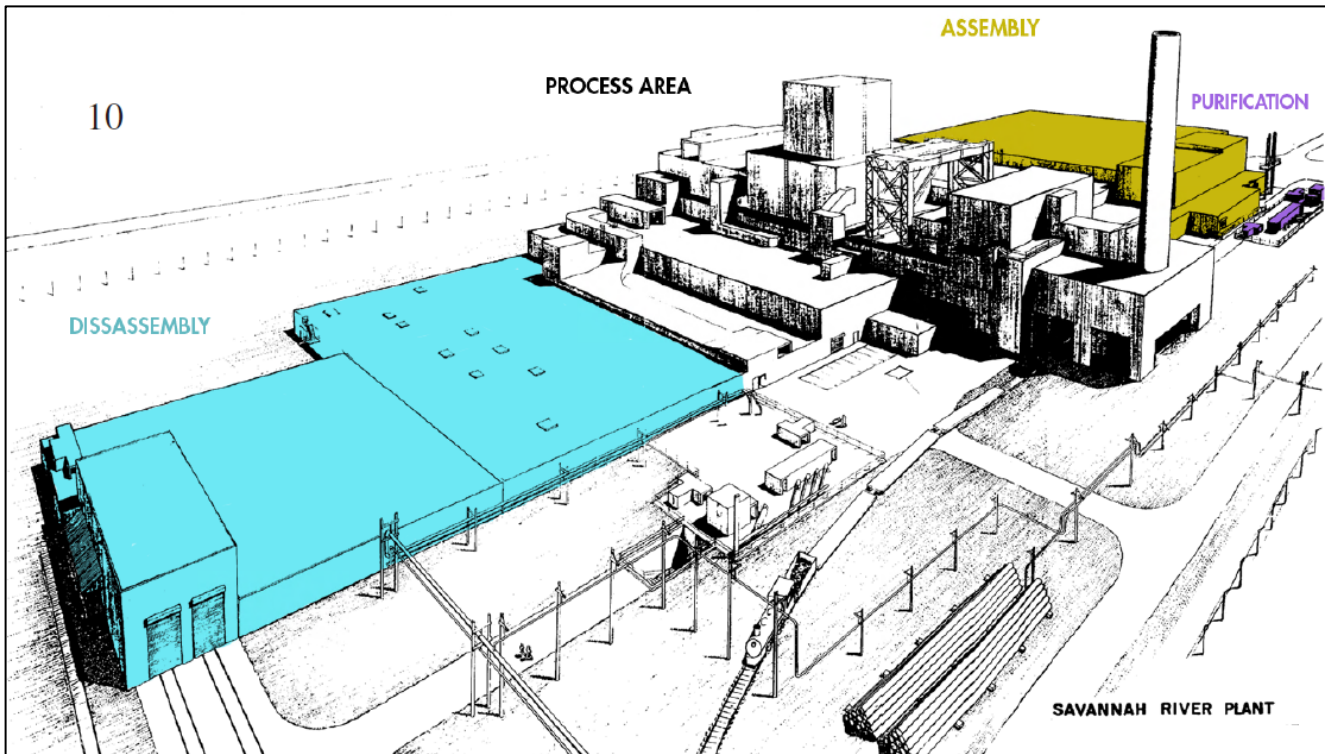


Figure 2-5. R Reactor primary building components [Reed 2010, p. 86].

### 2.2.1.1 Assembly Area

The Assembly Area was the location in each reactor building where fuel elements and target materials to be irradiated were received, cleaned, assembled, and stored before loading in the reactor. Though functionally alike, there were discernable differences in the size, orientation, and layout of the five assembly areas. The principal components of all the assembly areas included receiving, operations, component storage, and final storage. The purpose of the Assembly Area was to receive, clean, assemble, and test the various elements that were intended to be used in the reactor. Reed [2010] states assembly areas were maintained as clean as possible, and contamination was kept to a minimum. No irradiated materials were allowed into the Assembly Area. Assembled fuel elements were tested and stored in the Assembly Area.

The Assembly Area was a single-story area with a basement and with foundations of reinforced concrete with spread footings. Views of a primary assembly room are provided in Figure 2-6. An Assembly Area was divided into smaller areas for receiving of materials brought in including foils and elements from the 300 Area, working and storage area, and final storage room. A fan room was also located in R and P Reactor buildings. The fan room provided ventilation, moving outside air to the assembly area and moving air from the main assembly area into the final storage room [Reed 2010, p. 119]. The critical work of cleaning, loading, and testing fuel foils was performed in the working and storage area.

In the work and storage area, workers degreased and cleaned fuel components before assembling them into foils. Workers assembled them on the north side of the center bays after cleaning the fuel elements. Once cleaned, workers loaded aluminum-clad uranium slugs into the quaterfoils (a fuel tube containing four slugs). Finally, workers performed a flow test of the assembly to ensure that the water flow through each was correct and unimpeded. Once the assembled components passed testing, workers transferred them to the final storage room using a monorail system that picked up each component and positioned it in the storage hangers room [Reed 2010, pp. 119, 122]. The fuel and

# ASSEMBLY AREA VIEWS



*(Above Left) View of Assembly Area, Showing Fuel Element Water Testing Area, December 22, 1959. SRS Negative DPSPF 6401-48. (Above Right) Loading Slugs for Flow Testing in Assembly Area, December 22, 1959. SRS Negative DPSPF 6401-03. (Below) General View of Assembly Area, March 20, 1963. SRS Negative DPSPF 8930-06.*



Figure 2-6. Assembly Area view [Reed 2010, p. 124].

target assemblies contained uranium and, after 1961, recycled uranium [WSRC 2000a, p. 16]. The recycled uranium contained trace amounts of fission and activation products [WSRC 2000a, p. 15].

When scheduled for processing, assembled components were moved by means of the monorail system into a room known as the Gripper Pick Up Station. The monorail system carried the components through a tall, narrow baffle in the wall that separated the Assembly Area from the Process Area. The baffle served as a radiation buffer to block contamination that might enter the Assembly Area through the narrow opening. That opening was called the Presentation Point. About 40 individuals were employed in assembling, testing, labeling, and storing the vertical elements before irradiation [Reed 2010, pp. 118–129].

### **2.2.1.2 Process Area**

The Reactor Room was known as the Process Room. Everything in the reactor building was connected to that room. The Assembly Area was connected by the Presentation Point. The Deposit Point and Exit Conveyor connected the Process Room to the Disassembly Room [Reed 2010, p. 135].

The Reactor Room was a rectangular-shaped room with reinforced concrete walls around 7 ft thick. In R Reactor, the reactor tank was 41 ft from the pickup point at the south end of the room, while the tank was 30 ft from the edge of the discharge canal at the north end of the room. The ceiling height of the room is 43 ft above the floor except for the area over the reactor tank itself, which had a height of 61 ft. The only windows into the room were from the control room. The ceiling of the Reactor Room was 5 ft thick. In the part over the reactor, there were 427 openings through the ceiling to allow the control rod extensions to pass through and down to the reactor tank. Actuators that drove the control rods were installed above the ceiling and were in the tallest portion of the Process Area.

Two photos of the Reactor Room are shown in Figure 2-7.

Concrete haunches were constructed along both sides of the room and supported the rails of two 120-t cranes and the Charging (C) and Discharging (D) machines, which were used to load and unload fuel. The Charging machine carried fuel assemblies from the Presentation Point while the Discharge machine carried the irradiated fuel to the Deposit Point, the entry to the Disassembly Area. When not in use, the C and D machines and the 120-t crane were removed from the Reactor Room for cleaning and storage in the Crane Wash Area and Crane Maintenance Area, which were separated from the Reactor Room by steel doors designed to block radiation leakage from the Reactor Room [Reed 2010, p. 138].

The Reactor Room also contained the Reactor Tank Robot, which was also carried by the 120-t crane. The robot contained devices such as grapplers, lights, and periscopes, which could be put into the fuel assembly positions by the Charging machine. There was a second robot equipped with dual telescopic booms, mechanical tool holders, and a video camera. The Reactor Room contained a holding rack for the intermediate holding of elements either before or after going into the reactor tank. The holding rack was also used to hold elements that were about to be charged to the reactor tank but were found to be defective [Reed 2010, p. 141].

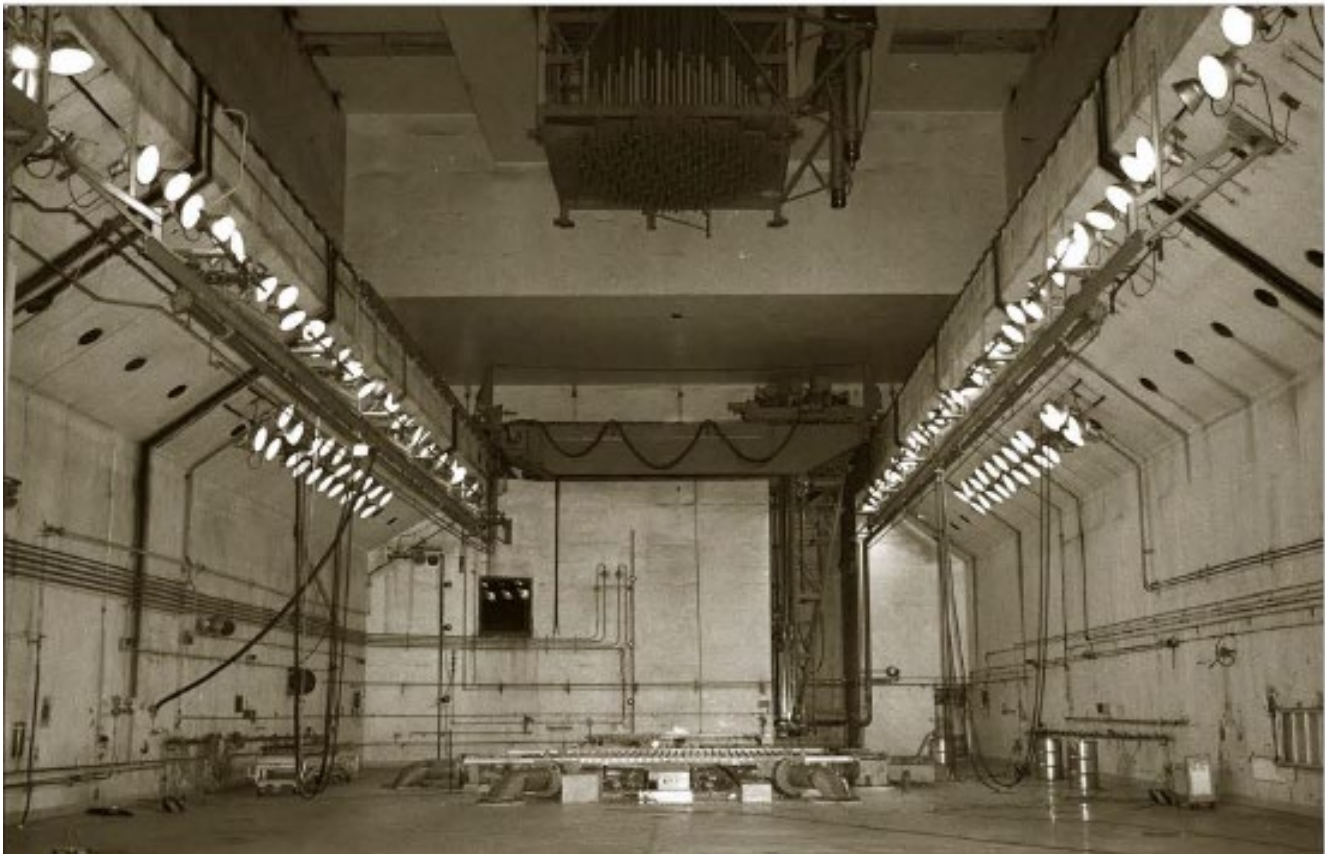


Figure 2-7. Two views of Reactor Room, R Reactor [Reed 2010, p. 136].

### **2.2.1.2.1 Actuator System**

The actuator was directly above the reactor and was essential for (1) the proper functioning of the reactor in control of the nuclear reaction when the reactor tank was in operation and (2) the need to shut down the reaction at the end of the operation. The actuator could be raised to several levels above the ceiling of the Reactor Room: 48, 66, 120, 130, and 149.6 ft above ground level. Control and safety rod actuators were housed at the 48-ft level as well as electromechanical means of raising the actuator structure off the reactor top during the charging and discharging operation. The 66-ft level contained a 1.5-t electric monorail hoist. Motors were housed in the 120-ft level and the 130-ft level. A 60-t hoist with a four-cable lift for the actuator system was above that. A 1.5-t electric monorail hoist was at the very top level [Reed 2010, pp. 162–169]. A diagram of the actuator system extended with rods in the reactor tank is shown in Figure 2-8.

A movable aluminum structure with vertical tubes and inner concentric members, eventually referred to as “the forest,” served as a guide for the entry of the control rods into the reactor tank [Reed 2010, p. 72].

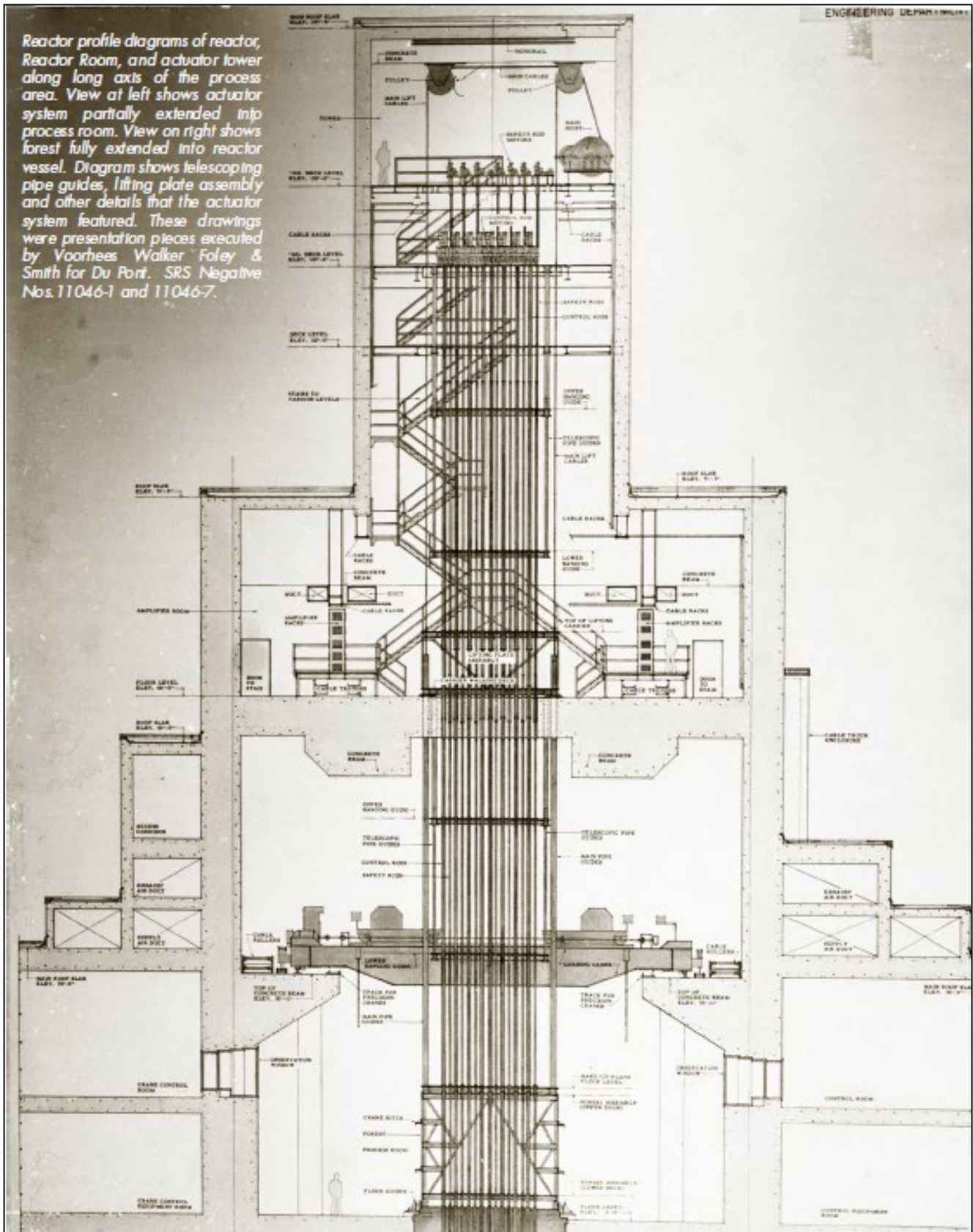


Figure 2-8. Actuator tower extended [Reed 2010, p. 164].

### 2.2.1.2.2 Reactor Tank

The reactor tank was in the center of the Reactor Room. A conceptual drawing of the tank is provided in Figure 2-9 showing a cutaway from the main floor and basement. The top of the tank was covered by a plenum measuring 17 ft in diameter and 1 ft thick. There were six intact pipes into the plenum so that water would enter the plenum, flow through the tank, and leave the tank through six large channels at the bottom of the tank.

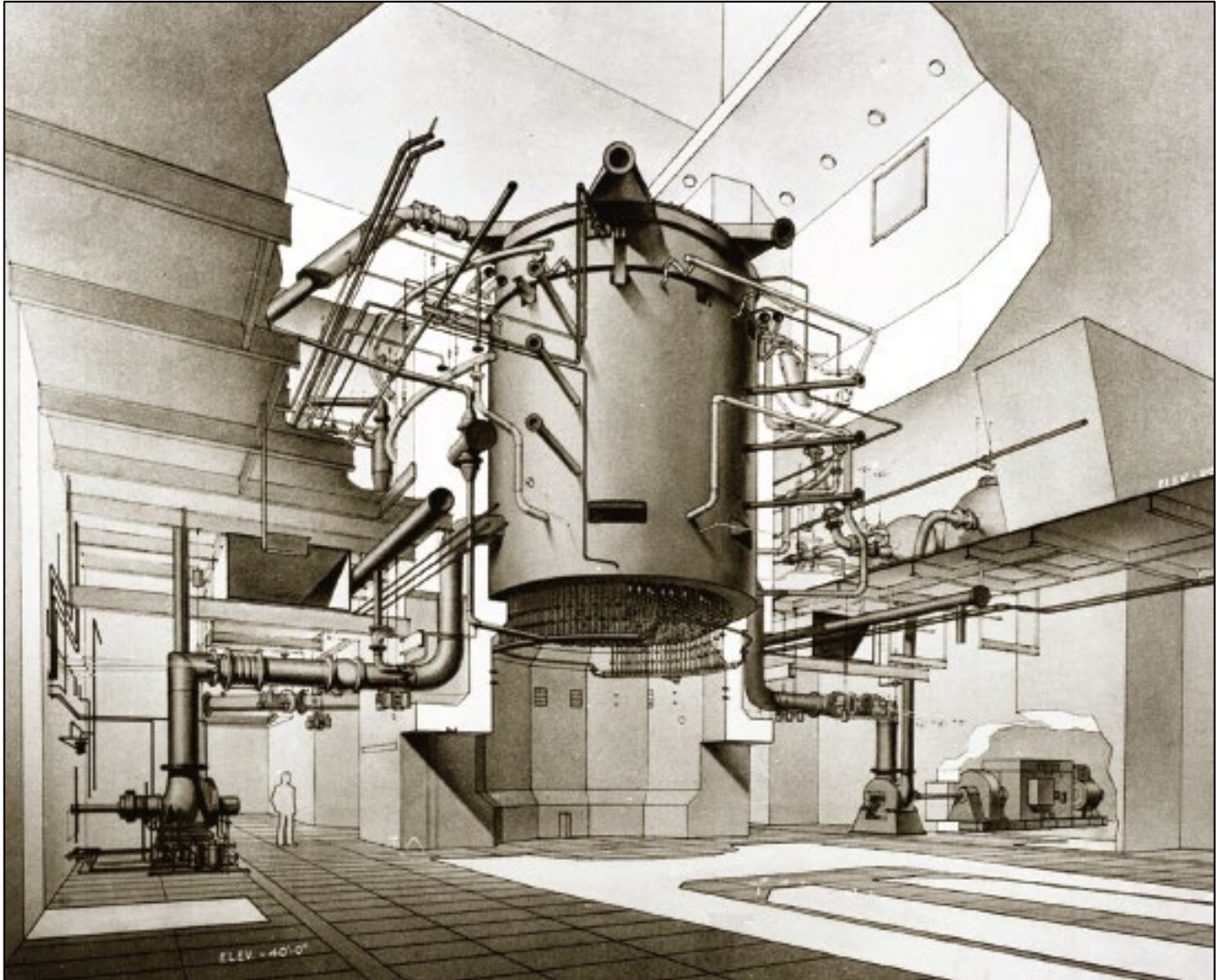


Figure 2-9. Cutaway drawing of reactor tank from -40-ft elevation [Reed 2010, p. 143].

There were 673 large diameter holes divided into 606 assemblies and 61 control rods, with the remaining 6 for gas release and another 165 small diameter holes for safety rods and instrument rods in each plenum. The reactor tank was encompassed by heavy shielding sufficient to withstand industrial and radiological stresses; thermal shielding around the reactor was based on anticipated power levels of about 700 MW. The thermal shield was designed to be a hollow annular tank filled with iron grid blocks or rings of small tube cut into small segments and packed into a bed. The annular tank was constructed so that workers could enter the annular tank sections to pack the metal. After packing, the annular tank sections were then filled with light water that circulated around the packed metal from top to bottom so that the reactor tank was surrounded by shielding material. The combination of iron and light water inside the thermal shields was considered optimal to absorb most of the radiation coming off the reactor tank [Reed 2010, pp. 143–150].



A top view of the plenum is shown in Figure 2-10.

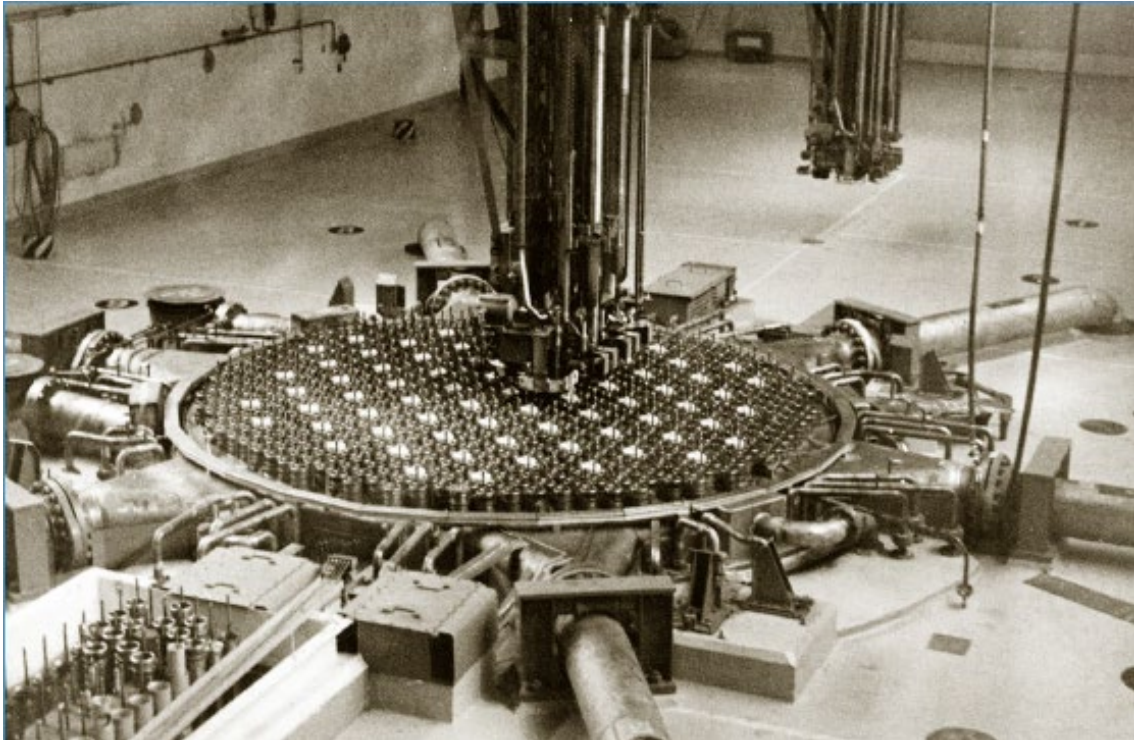


Figure 2-10. Reactor plenum [Reed 2010, p. 149].

#### **2.2.1.2.3 Water and Gas System for Reactor Tank**

The movement of the heavy water in the reactor tank was essential. SRS used a closed circular system that started at the reactor tank, the top of which was at 0 ft (floor level) with the base at the -20-ft level. Heavy water entered the tank through nozzles at the top, exited the tank through similar nozzles at the base, then traveled downward through pipes called suction lines to the pump, which pushed the water through the entire system. The pump and the generator were both at the -40-ft level. The pump then forced heavy water back up to the -20-ft level where it entered the heat exchangers. At that point, the hot heavy water came into close contact with the much cooler river water that was pumped from Building 190. River water used to cool heavy water entered the system only at the heat exchangers. River water, heated by heavy water, was discharged back to the outside and eventually to the river. Heavy water was then transferred from the heat exchangers to the 0-ft level and back into the reactor tank [Reed 2010, p. 151]. The heavy-water process is depicted in the drawing presented in Figure 2-11.

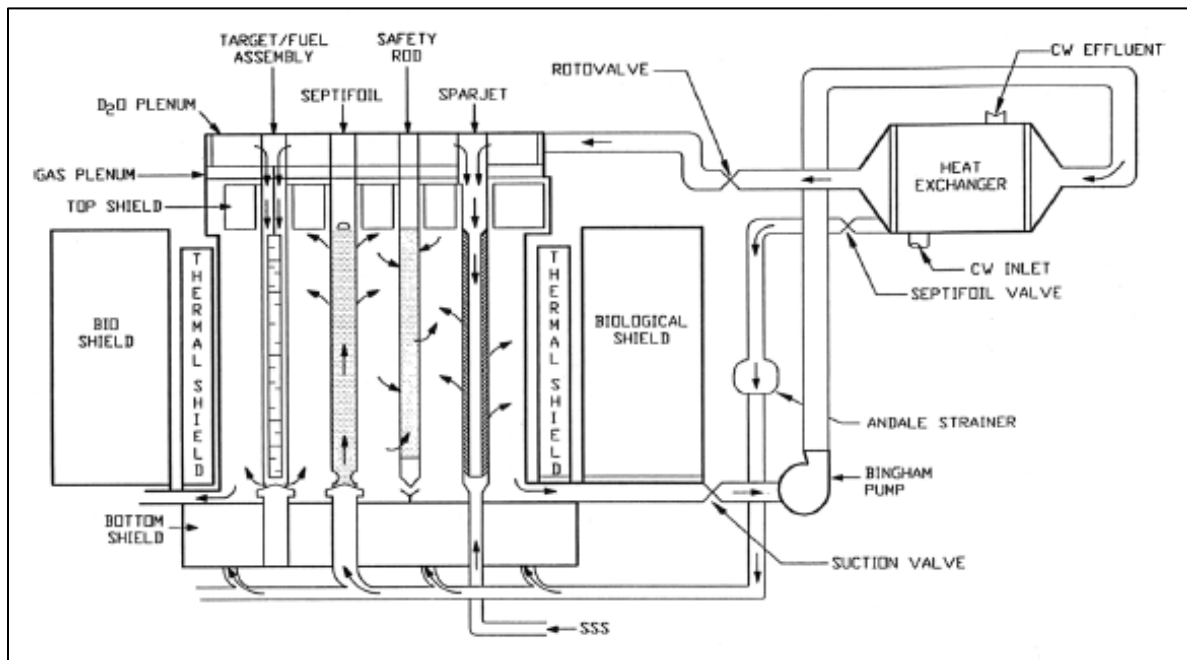


Figure 2-11. Diagram of process water flow in reactor system [Reed 2010, p. 152].

### 2.2.1.3 Disassembly Area

Irradiated reactor materials were processed in the Disassembly Area after being stored for a period to allow the excess radiation to decay. The Reactor Room and the Disassembly Area were connected by the Deposit Point. A machine and track device carried the irradiated material underwater from the Deposit Point to the Disassembly Area. Then the materials were disassembled to be further processed. Most of the disassembly tasks were performed underwater with remote-controlled equipment due to high radiation levels. Once the materials were disassembled, they were loaded into shipping casks for transfer to Separations.

Almost everything that could be inserted in the reactor tank was processed in the Disassembly Area. Fuel assemblies (quatrefoils) were the most numerous materials disassembled but others included instrument rods, safety rods, target control rods, nontarget type control rods, thimbles, and other types of fuel assemblies. While most work was performed underwater, there were components for the emergency handling of irradiated materials with spiral shields that provided shielding against the radiation in the tank [Reed 2010, pp. 174–176].

The building for the K Reactor underwent modifications and upgrades in the early 1990s and became the K-Area Complex. As of 2023, the facility provides for the handling and storage of plutonium and other special nuclear material [DOE, no date a]. In 1996, the underwater disassembly basin at L Reactor was reconfigured to handle spent nuclear fuel from offsite domestic and foreign sources. Since 2003, the L-Area Disassembly Basin has been the only receipt and storage facility at SRS [DOE, no date b].

### 2.2.1.4 Purification

The Purification Area was in the reactor building but not directly connected to the Reactor Room or Disassembly Area. The primary purpose of purification was to constantly check, clean, and remove foreign elements in the heavy-water moderator and the helium blanket gas in the plenum. Purification was done by a combination of filtration, deionization, and distillation with the water treated through an average flow of 30 gpm. This flow was diverted to the Purification Area after going through the heat

exchangers. Once purified, it was routed back to the main flow on the suction side of the circulating pumps. The moderator was also routed through the smaller blanket gas system to remove and collect oxygen and deuterium, which were returned to the heavy-water moderator [Reed 2010, pp. 180–183].

The purification work was carried out in remotely operated cells; each reactor had three cells labeled A to C. The first two reactors, R and P, contained four cells (A to D), and the later three had just two (E and F). The D cell served as a conduit and could detain any moderator that was deemed too impure to move to the other three cells. All other heavy water was pumped to cells A, B, and C through parallel systems with prefilters, deionizers, and afterfilters connected in series. In case of emergencies, there were two units on standby. The moderator was processed through a combination of filters and deionizers. There was a special filter only in B used to remove radioactive contamination in the gas used to dry equipment in A, B, or C. The last three reactors, L, K, and C, had Purification Areas that were smaller than those in R and P. The later reactors were only equipped with two cells, identified as E and F, neither of which were equipped with prefilters [Reed 2010, pp. 180–183]. A photograph of the front view of purification cells is shown in Figure 2-12.



Figure 2-12. View of Purification Area cells [Reed 2010, p. 183].

#### 2.2.1.5 Reactor Operating Problems and Incidents

At SRS, reactor operational problems were initially categorized as human error, failures of instruments or controls, or deterioration of equipment. In early years there were more human errors, while in later years most of the problems were due to the other causes. Any abnormal occurrences, even insignificant ones, were investigated as “reactor incidents”.

In 1985, SRS reported the 30 most significant incidents to date, which ranged from simple mechanical failures with small potential for harm to a few that had significant potential to cause damage to the reactor, the fuel, or both. Of the 10 most significant reactor incidents, 1 occurred from 1974 to 1985, 7

occurred from 1964 to 1973, and 2 occurred from 1954 to 1963. The incident SRS determined to have had the most serious potential to cause damage to the reactor or fuel occurred in January 1960 in L Reactor when the rate of power increase and the power level greatly exceeded prescribed limits. Reactor operators promptly inserted the control rods, which lowered reactor power and avoided potential damage. After that event SRS management took immediate action to prohibit rapid restarts of SRS reactors. The 10th-ranked incident happened in 1979 and involved the loss of offsite power. After 1985, only one event occurred that was comparable to the most significant of incidents in 1985 or before. That incident involved the startup of P Reactor after a long shutdown without discharge of fuel and target elements. An unusually large accumulation of  $^3\text{He}$  resulted from the decay of tritium in target elements. The effect of helium neutron absorption was overlooked in calculating the point at which the reactor would become critical as control rods were withdrawn. None of the 30 or any other reactor failures or incidents resulted in injuries to workers, significant adverse effects on the population, or damage to the environment [Bebbington 1990, pp. 85–93].

### 2.2.1.6 Production Campaigns

The original primary function of the SRS reactors was the production of plutonium and tritium for nuclear weapons. However, even in the early 1950s, the reactors were producing radioisotopes for nonweapon uses, which continued in the 1960s with the reduction of the military mission [Strack 2002, p. 438]. SRS irradiated depleted uranium,  $^{237}\text{Np}$ , and lithium targets to produce  $^{239}\text{Pu}$ ,  $^{238}\text{Pu}$ , and  $^3\text{H}$ , respectively. The targets were irradiated and then transferred to the Tritium Facilities, H-Area Separations ( $^{238}\text{Pu}$ ), or F-Area Separations ( $^{239}\text{Pu}$ ) for processing. SRS sent reactor fuel to H-Area Separations for processing. Between 1955 and 1967, SRS produced large quantities of special radionuclides in the reactors for various campaigns. For example, a series of campaigns produced 66 MCi of  $^{60}\text{Co}$  for Oak Ridge National Laboratory (ORNL), the U.S. Bureau of Mines, the U.S. Army Food Process Development Irradiator, Japan Atomic Energy Research Institution, Brookhaven National Laboratory, and Marine Products Development. SRS also produced 630 kg of  $^{233}\text{U}$  from irradiation of thorium for the thorium breeder reactor program [Bebbington 1990].

The target material in the C and K Reactors consisted primarily of  $^{242}\text{Pu}$ . In addition, three 1-in.-diameter thimbles contained  $^{243}\text{Am}$ ,  $^{244}\text{Cm}$ , and 150 other target materials of 66 elements for nine universities and laboratories [Boswell 2000]. Table 2-3 lists the target materials that were used to supply irradiated samples to Argonne National Laboratory (ANL), ORNL, Lawrence Livermore National Laboratory (LLNL), and Lawrence Berkeley National Laboratory (LBNL) [Crandall 1965, p. 27]. There were additional targets; for example, one tritium-producing core (Mark VI) was designed with a removable inner lithium target to irradiate the fuel to a very high exposure. The high exposure depleted 70% of the  $^{235}\text{U}$  and thus concentrated  $^{236}\text{U}$ , which was formed from every sixth neutron capture in  $^{235}\text{U}$ . Irradiation of  $^{236}\text{U}$  formed  $^{237}\text{Np}$ , which was the target material to produce  $^{238}\text{Pu}$  [Boswell 2000, pp. 110–118]. In addition, the irradiation of  $^{242}\text{Pu}$ , after a succession of neutron captures and beta decays, successively produced other target materials:  $^{242}\text{Pu}$ ,  $^{243}\text{Pu}$ ,  $^{243}\text{Am}$ ,  $^{244}\text{Am}$ ,  $^{244}\text{Cm}$ ,  $^{245}\text{Cm}$ ,  $^{246}\text{Cm}$ ,  $^{247}\text{Cm}$ ,  $^{248}\text{Cm}$ ,  $^{249}\text{Cm}$ ,  $^{249}\text{Bk}$ ,  $^{250}\text{Bk}$ ,  $^{250}\text{Cf}$ ,  $^{251}\text{Cf}$ , and  $^{252}\text{Cf}$  [Strack 2002, pp. 441, 443]. There were also special irradiations during both the High Flux Demonstration Program (HFDP) and the Curium I campaign to produce test quantities of  $^{46}\text{Sc}$ ,  $^{170}\text{Tm}$ ,  $^{171}\text{Tm}$ , and  $^{204}\text{Tl}$  [DuPont 1984a, p. 83].

The composition and relative abundance of the radionuclides in the reactor environment are not known. Radionuclide production in SRS reactors is plotted by time in Figure 2-13. For determining internal or external exposure in the reactor areas, the radionuclides of concern in the reactor areas are presented in Table 2-4.

Table 2-3. Examples of general samples irradiated during the HFDP.<sup>a</sup>

Sample	Laboratory	Sample	Laboratory	Sample	Laboratory
Si-30	ANL	Dy-158	ANL	Am-241	ANL, LLNL
S-33	ORNL	Tb-158	LBNL, LLNL	Cm-244	ANL, LLNL, LBNL
Ca-40	ORNL, ANL	Gd-160	ANL, LLNL, LBNL	Bk-249	ANL
Ca-48	ANL	Er-162	LBNL, LLNL	Cf-252	ANL, LLNL, LBNL
Fe-54	ORNL	Dy-164	ANL, LBNL	Es-254	LBNL
Fe-58	ORNL, ANL, LBNL	Ho-165	LBNL, LLNL	LiCl	LBNL
Ni-58	ORNL, ANL	Yb-168	ORNL	MgO+Cr-50	LLNL
Ni-62	ORNL	Tm-168	LBNL, LLNL	KPO <sub>3</sub>	LLNL
Nb-93	ORNL	Er-170	LBNL	Ar	LLNL
Zr-96	ANL	Yb-176	ANL, LBNL, LLNL	RbCl	ORNL
Ag-107	LBNL	Lu-176	LBNL	Hf	ANL
Ag-108	LBNL	Hf-177	LBNL	HFfO <sub>2</sub>	LLNL
Sn-118	ORNL, LBNL, ANL	Hf-179	LBNL	Os	LLNL
Sn-124	ANL	Hf-180	ANL, LLNL, LBNL	W	LLNL
Ba-130	LBNL	Ta-180	LBNL	Re	LLNL
Sm-144	ORNL, LBNL	Ta-181	LBNL	Be	ANL
Nd-146	ORNL	W-186	ANL, LLNL, LBNL	Be <sub>3</sub> N <sub>2</sub>	ORNL
Nd-150	ANL	Re-187	LLNL, LBNL	Xe	LLNL
Sm-150	LBNL, LLNL	Os-190	LBNL	TiO <sub>2</sub>	ANL
Sm-154	ORNL	Os-192	ANL, LLNL, LBNL		
Dy-156	LBNL, LLNL	Bi-209	LBNL		

a. Source: Crandall [1965, p. 27].

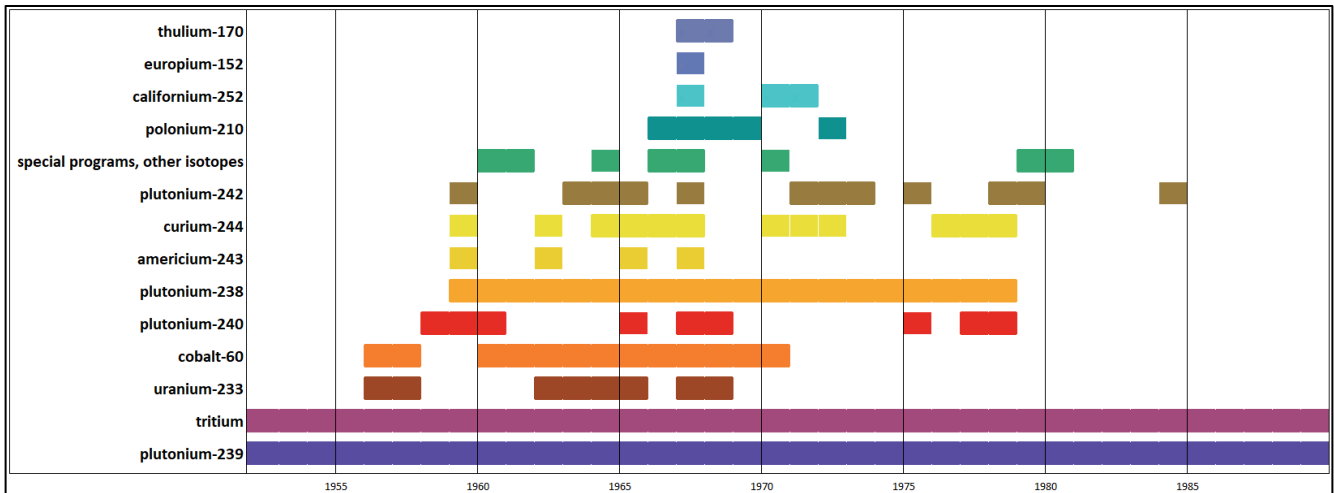


Figure 2-13. Radionuclide production [Strack 2002, p. 464].

Table 2-4. Radionuclides of concern for all reactors.<sup>a,b</sup>

Radionuclide	Radionuclide	Radionuclide	Radionuclide	Radionuclide
H-3 <sup>c</sup>	Sr-90	I-131	Ce-143	Np-239
Cr-51	Sr-91	I-133	Ce-144	Am-241
Mn-54	Sr-92	Cs-134	Eu-152	Cm-242
Co-58	Nb-95	Cs-137	Eu-154	Cm-244
Fe-59	Zr-95	Ba-140	U-234	
Co-60	Ru-103	La-140	Pu-238	
Zn-65	Ru-106	Ce-141	Pu-239	

a. Source: LaBone [1996, p. 4].

b. The composition and relative abundance are not known.

c. Tritium is assumed to produce most of the personnel exposure because of the large quantities present.

### 2.2.1.6.1 Curium

To prepare radioisotope thermoelectric generators for the space program, SRS produced 275 kg of  $^{238}\text{Pu}$  from irradiation of  $^{237}\text{Np}$ , 600 g of  $^{210}\text{Po}$ , 1,700 Ci of  $^{152}\text{Eu}$ , and 374 MCi of  $^{170}\text{Tm}$ . Other radionuclides produced during this period and in subsequent years included  $^{121}\text{Sn}$ ,  $^{139}\text{Ce}$ ,  $^{166}\text{Ho}$ , and  $^{41}\text{Ca}$  for Brookhaven National Laboratory, the University of Illinois, and Denmark [Bebbington 1990, pp. 96–98]. These sources were encapsulated before irradiation and shipped off site, so the potential for intake was minimal. The sources were inspected and loaded into casks underwater to reduce external exposure [Gray 2006, p. 2].

While SRS performed some work with transplutonium in the late 1950s and early 1960s, it implemented named programs in the years that followed the shutdown of R Reactor in 1964. Curium I was the first of these to produce  $^{244}\text{Cm}$  as a possible heat source for the thermoelectric generation of electricity for space exploration and as a target material for the anticipated production of californium. Californium was the goal of the AEC transplutonium programs. The Curium I program was conducted in C Reactor from May to December 1964. The Curium I program was initiated with the irradiation of  $^{239}\text{Pu}$  that was alloyed with aluminum and fabricated into aluminum-clad billets and then extruded into thin-walled fuel tubes [Strack 2002, pp. 366–368; Boswell 2000]. The entire progression from  $^{239}\text{Pu}$  to  $^{244}\text{Cm}$  was lengthy and complicated. After successive neutron captures and beta decays,  $^{239}\text{Pu}$  was transformed by stages of irradiation to become  $^{240}\text{Pu}$ ,  $^{241}\text{Pu}$ ,  $^{242}\text{Pu}$ , and eventually  $^{244}\text{Cm}$ . SRS split the entire progression into two programs due to differences required in the manufacturing techniques. Curium I took the progression from  $^{239}\text{Pu}$  to  $^{242}\text{Pu}$  [Strack 2002, pp. 440–441; Boswell 2000].

The Curium II program required a much higher neutron flux to push irradiated material to  $^{244}\text{Cm}$ , higher than had ever been attempted before. For that reason a separate high-flux program was interjected between Curium I and II; the HFDP was started in February 1965 and ended on February 9, 1966, at the conclusion of the 28th fuel cycle. That work was performed in C and K Reactors, with work also performed at the High Flux Isotope Reactor at ORNL. The target material was normally  $^{242}\text{Pu}$  using cadmium control rods. Fuel positions were greatly reduced, while the active length of  $^{235}\text{U}$  in the fuel assemblies was reduced by half. The neutron flux was demonstrably increased in neutrons per square centimeter per second. A high neutron flux record was established in December 1965 in C Reactor that has never been matched [Boswell 2000; Strack 2002, pp. 366–368]. A photo of Cherenkov radiation emanating from fuel assemblies irradiated during HFDP is shown in Figure 2-14.

The Curium II program was started in December 1965 with the goal to finish the production of  $^{244}\text{Cm}$  by transmuting the plutonium burnout residue from Curium I using a high-flux lattice. The program began with the separation and purification of materials from Curium I. Plutonium from Curium I was refabricated into new plutonium-aluminum fuel tubes, while the americium and curium produced in the program was set aside for later combination with the greater quantity of material that would come from the second program. The irradiation program for Curium II entailed a mixed lattice of five enriched uranium-aluminum fuel assemblies and one plutonium-aluminum assembly. The load was bombarded by a neutron flux in excess of  $10\text{ n/cm}^2/\text{s}$ . The irradiations were continued until  $^{242}\text{Pu}$  and  $^{243}\text{Am}$  approached a steady-state condition. The  $^{243}\text{Am}$  decayed to  $^{244}\text{Cm}$ . By the end of the Curium II program in May 1967, SRS produced a total of 5.9 kg of  $^{244}\text{Cm}$ , almost double the amount AEC had initially sought. Curium II also produced around 3 mg of  $^{252}\text{Cf}$  [Strack 2002, pp. 442–443].



Figure 2-14. Cherenkov radiation from irradiated fuel assembly [Strack 2002, p. 441].

#### **2.2.1.6.2 Californium**

SRS implemented the Californium I program in August 1969. This program was completed in November 1970 with irradiation work in K Reactor. The purpose of Californium I was to produce californium in sufficient quantities to make the isotope available to industry and medical facilities [Strack 2002, p. 443]. Before this, SRS had developed the concept that californium placed in platinum needles could be used for the treatment of deep tumors not easily accessible by other forms of treatment.

Residues from Curium II were used as targets to produce  $^{252}\text{Cf}$ . SRS designed a new high-flux lattice for the campaign. It consisted of 96 large-diameter assemblies of  $^{235}\text{U}$  and  $^{242}\text{Pu}$  and 104 target assemblies of  $^{237}\text{Np}$  for the simultaneous production of  $^{238}\text{Pu}$  [Strack 2002, p. 443]. The reactor core also contained six assemblies of  $^{243}\text{Am}$  and  $^{244}\text{Cm}$  target material. Each fuel cycle lasted 3.5 to 4 days and was followed by a 12- to 18-hour reactor shutdown to replace irradiated fuel tubes. SRS produced 2.1 g of  $^{252}\text{Cf}$  by the end of the program in November 1970, which was the largest amount of californium ever produced [Boswell 2000, p. 6].

Another californium production campaign was planned but never executed because the market for medical and other use of  $^{252}\text{Cf}$  did not materialize [Boswell 2000; Strack 2002, pp. 443–444].

#### **2.2.1.6.3 Plutonium-238**

The most consistently produced nonweapon product from SRS was  $^{238}\text{Pu}$ , which was used as a heat source that supplied many of the power generators for satellites and space exploration vehicles. SRS produced more  $^{238}\text{Pu}$  than any other facility in the United States [Strack 2002, p. 464].

SRS began production of  $^{238}\text{Pu}$  by the irradiation of  $^{237}\text{Np}$ . The challenge in making  $^{238}\text{Pu}$  is to minimize formation of  $^{236}\text{Pu}$  from absorption of fast neutrons. The solution to this problem was the design of reactor lattice configurations that maintained adequate separation of fast neutrons from neptunium target material. To accomplish this separation, neptunium targets were in the  $\text{D}_2\text{O}$  reflector beyond the fuel in high-flux cores. By 1978, SRS produced over 300 kg of  $^{238}\text{Pu}$ . SRS used an additional method for the formation of  $^{238}\text{Pu}$  that avoided the unwanted  $^{236}\text{Pu}$  isotope by producing

$^{238}\text{Pu}$  as a decay product of  $^{244}\text{Cm}$ . In 1970,  $^{238}\text{Pu}$  was produced and separated that contained less than 0.3 ppm of  $^{236}\text{Pu}$ . While production of  $^{237}\text{Np}$  was not a separate reactor campaign, the neptunium isotope was key to the production of  $^{236}\text{Pu}$  in this reaction:  $^{237}\text{Np} (n, \gamma) ^{238}\text{Np} \rightarrow ^{238}\text{Pu}$ . Refer to ORAUT-RPRT-0065, *An Evaluation of Neptunium Operations at Savannah River Site*, for information on neptunium fuel production [Oak Ridge Associated Universities (ORAU) Team (ORAUT) 2016].

#### **2.2.1.6.4 Polonium-210**

Starting in June 1966,  $^{210}\text{Po}$  was produced at SRS by the irradiation of high-purity bismuth slugs as part of the Curium II program. SRS produced 324,200 Ci through May 1967. The slugs were shipped to Mound in Ohio for recovery of the polonium [Gray 1967].

#### **2.2.1.6.5 Thorium-232 and Uranium-233**

Thorium was irradiated at SRS for production of  $^{233}\text{U}$ . In November 1954, 130 low-enriched uranium slugs and 156 thorium slugs were produced in the Assembly Area of Building 105-R for irradiation. Production was completed by December 1954 [DuPont 1952–1972, 1955–1962, 1961–1962]. Canned thorium slugs were received from Sylvania Electric Products/Sylvania-Corning Nuclear Corporation (Sylcor) for R-6 and R-7 full-core reactor loadings. In 1955, the slugs were irradiated through August; inventories of R-6 and R-7 slugs remained at the R and L Reactors. In February 1955, two quatrefoils containing striped LMF (code for an alloy of lithium, aluminum, and natural uranium) and thorium metal, and one quatrefoil containing one channel of all thorium slugs and three channels of aluminum spacers were assembled and put in L-3 for an irradiation test in the L Reactor. The quatrefoils containing striped LMF and thorium metal were discharged, their fuel column heights measured, and the slugs unloaded for visual inspection in Building 105-L. Inspection of the slugs indicated erosion or corrosion marks approximately 0.010-in. deep on the side of the thorium slug cans around the cap weld bead [DuPont 1955a, p. 40].

In October 1961, SRS asked Sylcor to produce 600 hollow thorium metal slugs for use in the Mark IV fuel [DuPont 1961a, p. 227]. Evaluations of thorium slugs that could be used in producing  $^{233}\text{U}$  continued through 1963 with some irradiations in L Reactor. Through this program, 18 kg of  $^{233}\text{U}$  with 3 to 5 ppm of  $^{232}\text{U}$  was produced from 20 t of thorium metal. The Sylcor core was redesigned to use thorium oxide rather than thorium metal as the target material. These redesigned assemblies were first stored in 1964 at the C, L, and P Reactors; irradiation commenced in 1965. Using the redesigned core, SRS produced 100 kg of  $^{233}\text{U}$  containing less than 5 to 6 ppm  $^{232}\text{U}$  from 56 t of thorium [DuPont 1984b, pp. 80–82].

In addition to the development of uranium fuel elements, SRS performed tests of thorium metal in 1964 to evaluate the performance of this material for a thorium/ $^{233}\text{U}$  converter cycle. Two thorium-uranium alloy tubes were fabricated and tested in the HWCTR to evaluate the performance of thorium tubes under conditions that would be encountered in a  $\text{D}_2\text{O}$ -moderated reactor operating on an equilibrium thorium/ $^{233}\text{U}$  fuel cycle [DuPont 1984a].

In 1965, SRS started development work on mixed lattices consisting of various target assemblies alternating with high-specific power, three-tube driver fuel assemblies. These lattices offered the capability of coproducing almost any variety of products that could have been desired. The first of the mixed lattices to go into the plant was the Mark XIIA-Mark 50A, which was intended to produce  $^{233}\text{U}$  along with lesser amounts of other radioisotopes [DuPont 1984a]. Lattice testing continued through June 1966 with irradiations performed in the L Reactor [DuPont 1952–1972, 1955–1962, 1961–1962].

SRS performed a campaign to produce a minimum of 150 kg  $^{233}\text{U}$  in K Reactor in June 1967. Operation of the four authorized charges continued into 1968 [DuPont 1984a]. The mixed-lattice charge consisted of 261 Mark XIIA fuel assemblies (three-tube drivers), 255  $\text{ThO}_2$  targets, and



84 depleted uranium blanket assemblies. Three thorium target stages (two heavy and one light) were irradiated during each driver cycle. Eight of the 12 authorized target runs were completed during 1967 and produced about 125 kg  $^{233}\text{U}$  with a  $^{232}\text{U}$  concentration of about 7.5 ppm [DuPont 1984b]. Thorium oxide was irradiated in C and K Reactors between 1966 and 1969 as part of the thorium extraction (THOREX) I and II campaigns. The THOREX II campaign was completed with the recovery of 80 kg of  $^{233}\text{U}$  from 35 t of irradiated  $\text{ThO}_2$ . A final THOREX campaign was conducted during 1969 with the recovery of 181 kg of  $^{233}\text{U}$  [DuPont 1984a]. Irradiation of thorium oxide was performed in the K Reactor. Some thorium materials were stored in the K Reactor through at least May 1971 [DuPont 1952–1972, 1955–1962, 1961–1962].

A summary of documented thorium operations at SRS reactors is shown in Table 2-5. A detailed description of thorium separations is given in Section 2.2.17.

Table 2-5. Thorium operations at reactors.

Period	Buildings and operations	Buildings and operations
07/1954–12/1957	105-R	Irradiation; nondestructive examination of irradiated thorium slugs and rods; thorium targets and fuel storage
07/1954–12/1957	105-K	Thorium storage
07/1954–12/1957	105-R	Irradiation; nondestructive examination of irradiated thorium slugs and rods; thorium targets and fuel storage
01/1960–12/1963	105-R, 105-L	Irradiation; nondestructive examination of irradiated thorium slugs and rods; thorium targets and fuel storage
01/1964–12/1964	HWCTR	Irradiation; nondestructive examination of irradiated thorium slugs and rods; thorium targets and fuel storage
01/1964–12/1966	105-L, 105-P, 105-K, 105-C	Irradiation; nondestructive examination of irradiated thorium slugs and rods; thorium targets and fuel storage
01/1967–12/1968	105-K	Irradiation; nondestructive examination of irradiated thorium slugs and rods; thorium targets and fuel storage
01/1967–12/1968	105-C	Thorium storage
01/1969–05/1971	105-K	Irradiation; nondestructive examination of irradiated thorium slugs and rods; thorium targets and fuel storage
01/1969–05/1971	105-L, 105-C	Thorium storage

#### 2.2.1.6.6 Cobalt-60

Cobalt-60 was produced by the irradiation of natural cobalt ( $^{59}\text{Co}$ ) in the SRS reactors for potential sources for cancer treatment, food sterilization, radiography, and as heat sources for thermoelectric generators. SRS produced small amounts as early as 1955. Beginning in 1965, the HFDP provided an opportunity to make large quantities of very high specific activity  $^{60}\text{Co}$  when  $^{59}\text{Co}$  was the preferred material for control and safety rods in high-flux reactor cores. The first production period (1955 to 1958) made  $^{60}\text{Co}$  with activity of 50 Ci/g. The second period (1959 to 1964) created  $^{60}\text{Co}$  with 100 Ci/g. The third program (1964 to 1967) generated material with 700 Ci/g. Total  $^{60}\text{Co}$  produced at SRS is estimated to be in the kilogram range [Strack 2002, pp. 365–366].

### 2.2.2 F-Area

F-Area is one of two designated areas SRS used to recover plutonium from reactor-irradiated uranium.

#### 2.2.2.1 221-F, F-Canyon

The primary building in the F-Area was a very large concrete building that processed radioactive materials from onsite reactors until the reactors were shut down. The F-Canyon went into operation in

November 1954. Fuel elements from the reactors were dissolved in acid to separate the uranium and plutonium from waste fission products by chemical extraction in solution. Both operations and maintenance were done remotely. [Reed et al. 2013, p. 57]. The F-Canyon was deactivated in 2006 (H-Canyon continues to operate) [SRNS 2015, p. 36]. Figure 2-15 shows a layout of the entire 200-F Area, and Figure 2-16 shows an architectural rendering of an aerial view of the 200-F Area with 221-F labeled as Canyon Building [Reed et al. 2013, pp. 70, 79].

The 221-F Canyon was a long rectangular building consisting of two parallel canyons that constituted the process area measuring 850 ft long by 122 ft wide and 66 ft high. The interior was organized in two ways. Horizontally, there were three long thin divisions: the Hot Canyon on the east side, the Warm Canyon on the west side, and the Central portion in the middle. Vertically, the Hot and Warm Canyons were set up for equipment to be located at the bottom. A small corridor for gang valves and air exhaust was at the bottom level. A concrete slab known as a "cell cover," a movable work platform (only on the Warm Canyon side), and a work crane for each canyon were above the equipment near the top of the building. The Central portion of the building was for personnel, feed tanks, and control. The Central portion was divided into four stories and contained feed tank galleries, sample aisles, and control rooms. A cross-section drawing of F-Canyon is shown in Figure 2-17. There were 18 sections overall in the F-Canyon building, numbered from south to north. The first section at the south end was 85 ft long to accommodate the special facilities for the beginning of the process, followed by 16 sections, each 43 ft long, followed by one final section that measured 45.5 ft long at the north end. Processing equipment was in the Hot and Warm Canyons from Sections 5 through 18. Each of those sections contained four cells or modules. Each module was set up to accommodate one piece of equipment [Reed et al. 2013, pp. 57, 88, 94, 98]. Figure 2-18 shows a drawing of a cutaway view of Section 14.

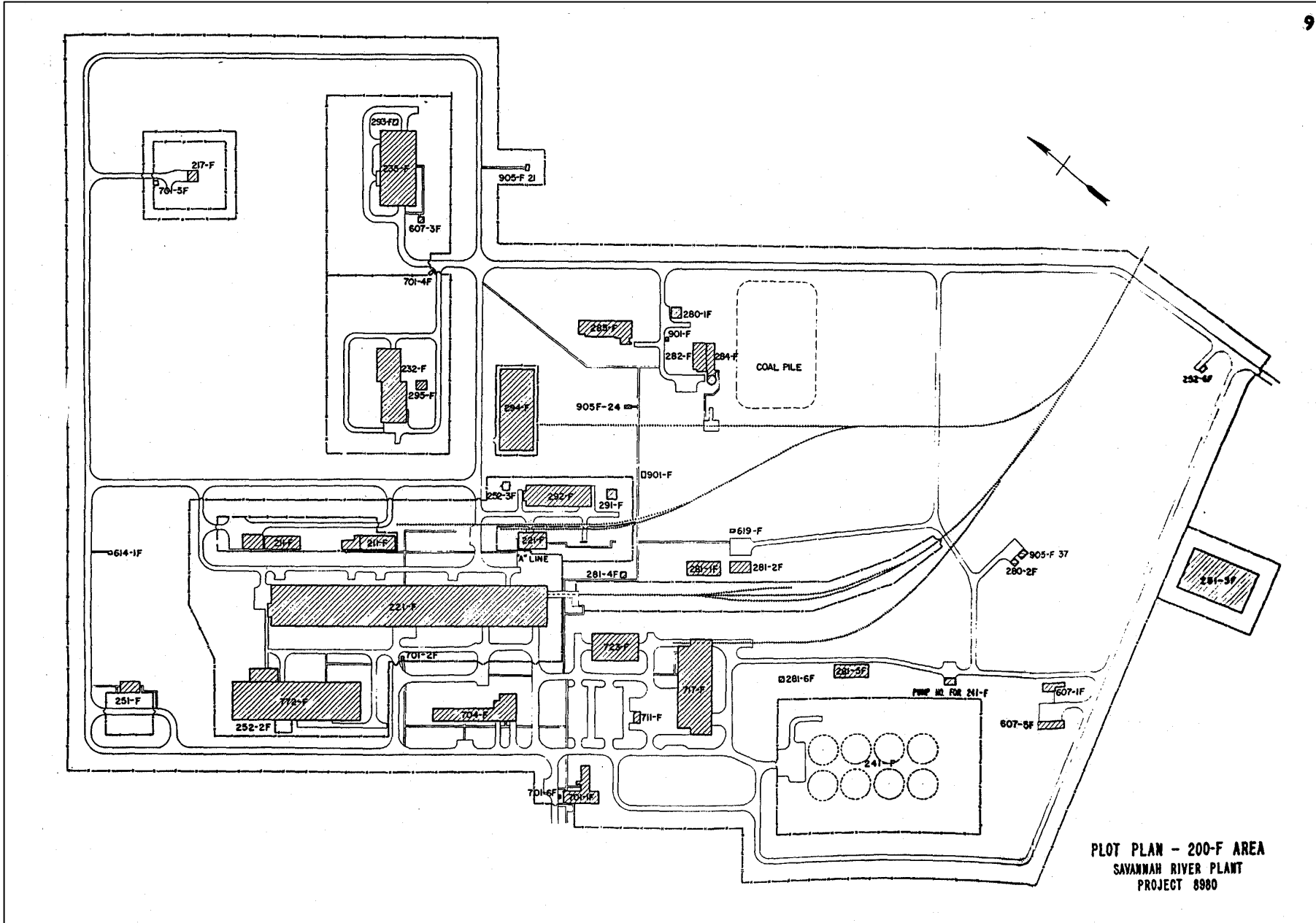


Figure 2-15. 200-F Area, January 1957 [DuPont 1957a, p. 13].

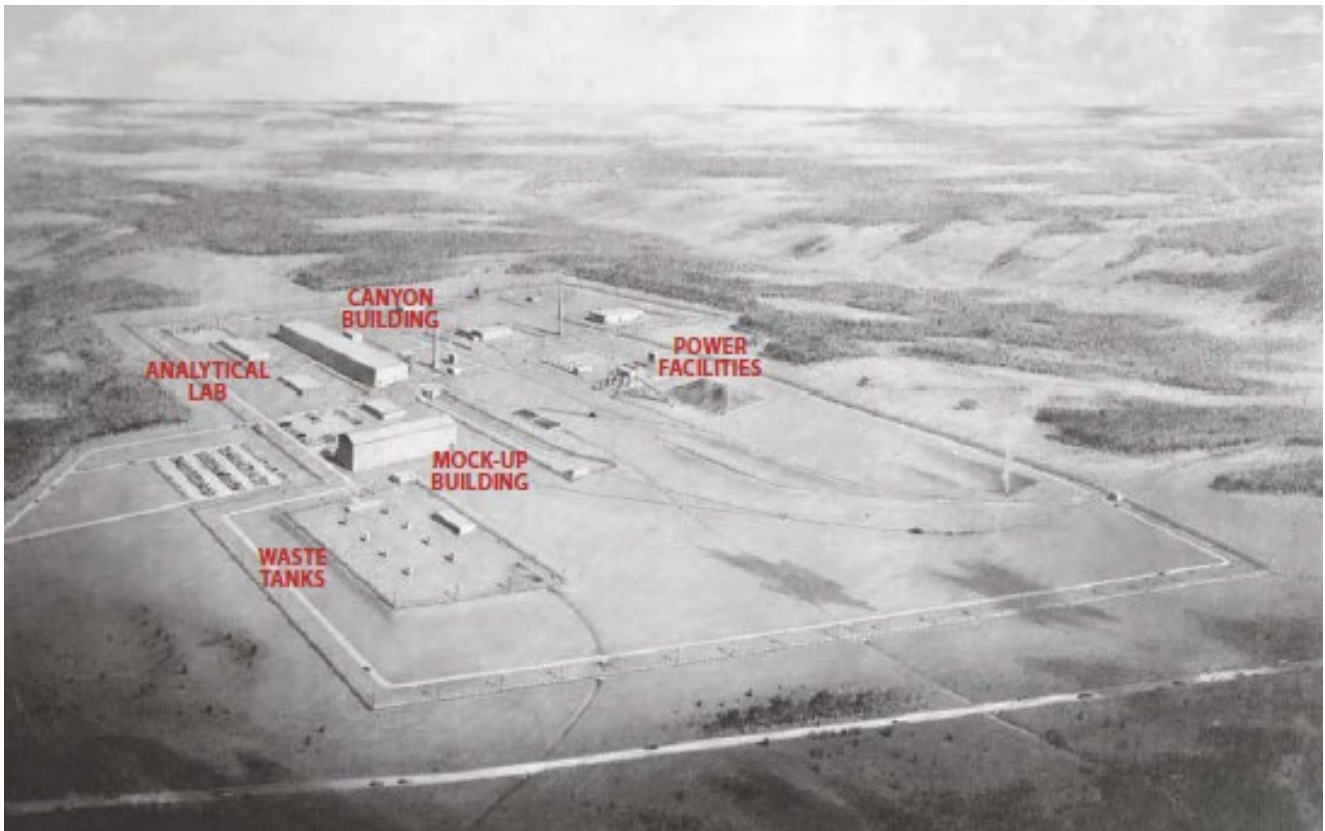


Figure 2-16. Architectural rendering of an aerial view of 200-F Area [Reed et al. 2013, p. 70].

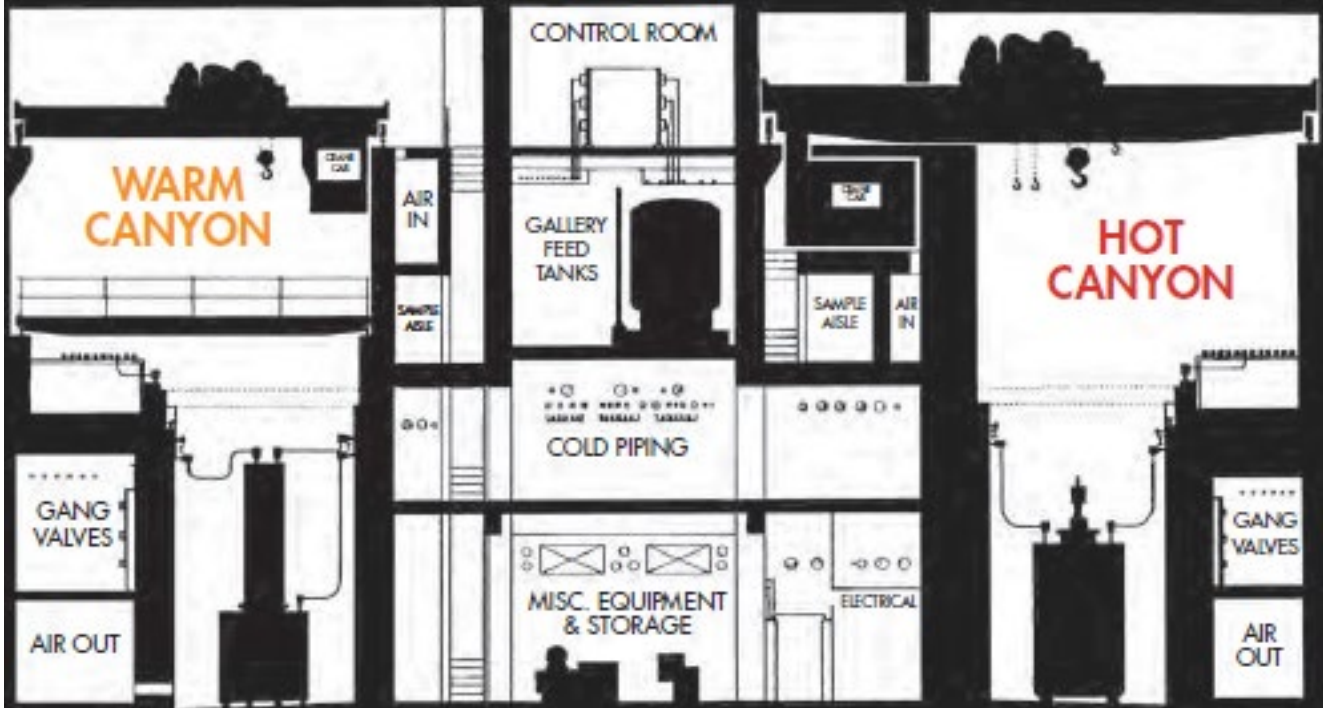


Figure 2-17. Canyon cross-section [Reed et al. 2013, p. 86].

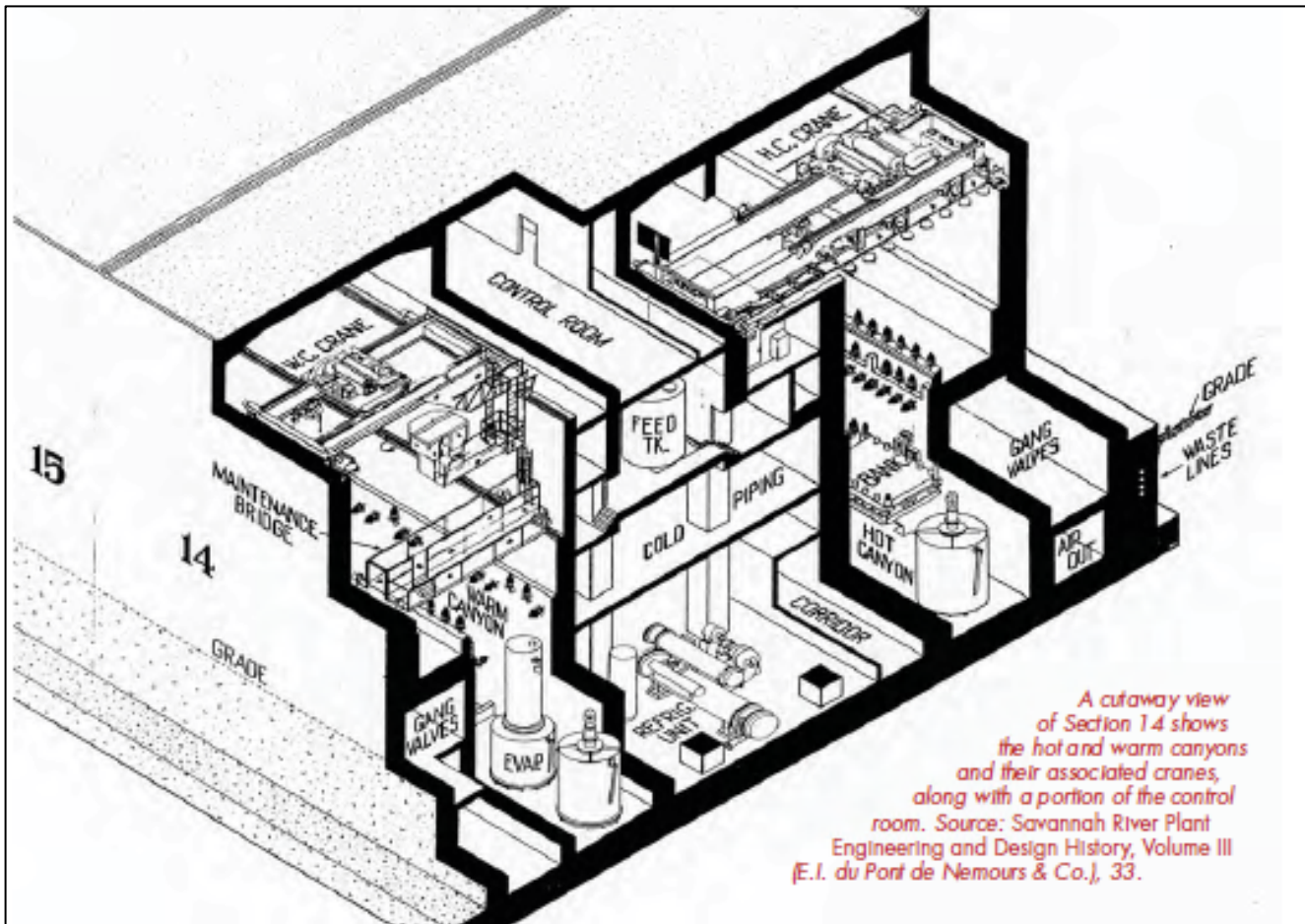


Figure 2-18. Section 14 cutaway view [Reed et al. 2013, p. 90].

### 2.2.2.2 Separation Process

The initial process for plutonium and uranium separation at Savannah River involved five major steps: dissolution, head-end treatment, first cycle, second uranium cycle, and second plutonium cycle. The last three steps made up the plutonium-uranium extraction (PUREX) process to recover  $^{239}\text{Pu}$  and  $^{238}\text{U}$  from reactor-irradiated targets. Recovered plutonium was transferred to the B-Line. Recovered uranium was transferred to the A-Line Facility. The entire PUREX process is diagrammed in Figure 2-19.

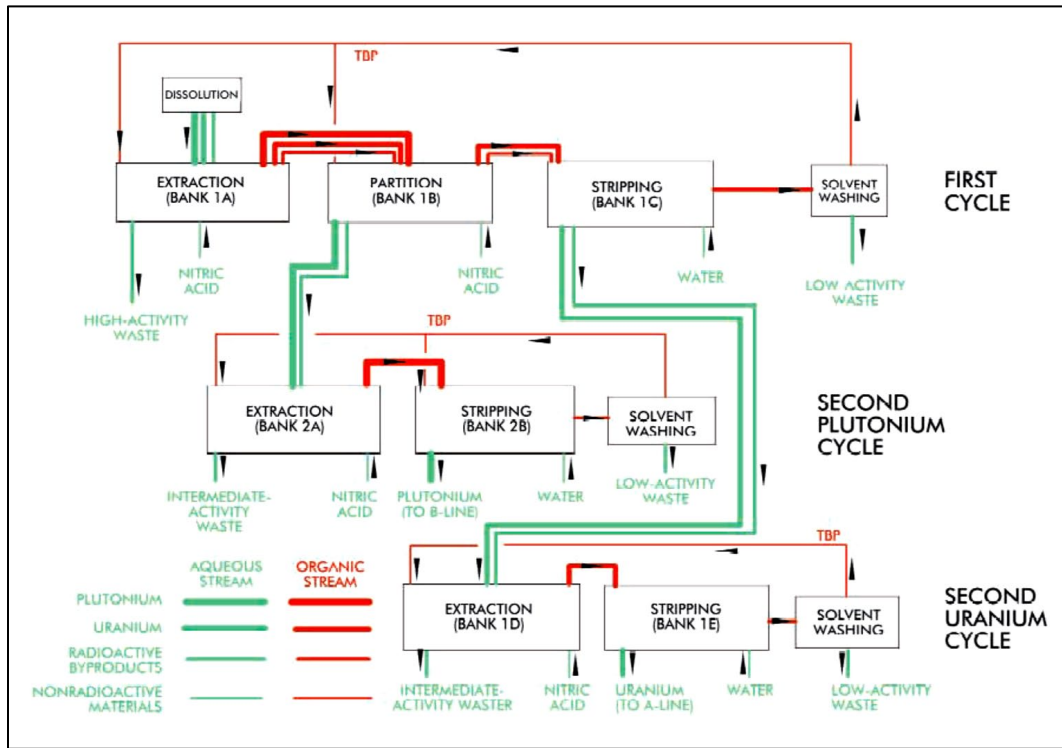


Figure 2-19. PUREX process steps [Strack 2002, p. 373].

**2.2.2.2.1 Dissolution**

Solid irradiated materials received at the canyon were converted to liquid form. Workers placed fuel assemblies in a dissolver, to which caustic solutions were added to remove the aluminum cladding. Workers then transferred the resulting solution to canyon waste tanks and added nitric acid to aid dissolution of the assemblies. Once dissolved, the liquid was centrifuged to remove zirconium and niobium fission products as well as additional solids [Strack 2002, p. 373]. Noble gases and released iodines were routed through sorbent materials coated in silver nitrate to reduce iodine emissions. Feed solution from the FB-Line special recovery process was fed into the head-end process [LaBone 1996, p. 10]. After 1961, the uranium target assemblies contained recycled uranium with trace amounts of fission and activation products [WSRC 2000a, p. 15].

**2.2.2.2.2 PUREX Cycles**

The first cycle of solvent extraction (head-end treatment) was developed by Knolls Atomic Power Laboratory and TNX and performed in the hot canyon in mixer-settler banks (1A, 1B, and 1C as shown in Figure 2-19). It separated plutonium and uranium from fission products using tributyl phosphate in a hydrocarbon diluent. Table 2-6 lists radionuclides of concern for the head-end stream.

Table 2-6. 221-F Canyon, Head-End Stream (Stream 1).<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	221-F, F-Canyon
<b>Period</b>	Operations from November 1954 to 2006
<b>Radionuclides of concern</b>	Ce-144, Zr-95, Ru-106, Cs-137, Sr-90, Pu-239

a. Sources: LaBone [1996, p. 11]; Reed et al. [2013, p. 57]; SRNS [2015, p. 36].

In the second cycle, plutonium from the first cycle was oxidized to the tetravalent state for feed to the cycle and mixed in Bank 2A into countercurrent aqueous and organic streams. Most of the residual fission contaminants remained with the aqueous stream. SRS reduced plutonium to the aqueous

phase in Bank 2B. SRS transferred that aqueous stream to a holding tank and eventually transferred it to the B-Line to be converted into plutonium metal. Table 2-7 lists radionuclides of concern for the second cycle plutonium stream.

Table 2-7. 221-F Canyon, Second Cycle Plutonium Stream (Stream 3).<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	221-F, F-Canyon
<b>Period</b>	Operations from November 1954 to 2006
<b>Radionuclides of concern</b>	Pu-241, Pu-239, Zr-95, Ce-144, Ru-106, Pu-238

a. Sources: LaBone [1996, p.12]; Reed et al. [2013, p. 57]; SRNS [2015, p. 36].

In the second cycle uranium step in Banks 1D and 1E, SRS purified the uranium content through Bank 1C, further extracted plutonium, and removed fission byproducts. Concentrated uranyl nitrate solution from Bank 1C was fed to Bank 1D to undergo another series of mixing and settling [Strack 2002, pp. 373–374]. Table 2-8 lists the radionuclides of concern for the second uranium cycle stream.

Table 2-8. 221-F Canyon, Second Uranium Cycle (Stream 2).<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	221-F, F-Canyon
<b>Period</b>	Operations from November 1954 to 2006
<b>Radionuclides of concern</b>	U-238, Zr-95, Ru-106, Ce-144, Pu-239, Cs-137, Sr-90, U-235

a. Sources: LaBone [1996, p.12]; Reed et al. [2013, p. 57]; SRNS [2015, p. 36].

Aqueous waste from the first extraction cycle, the second uranium cycle, acid strip condensate, ventilation system sumps, and concentrate from the second-stage high-activity waste evaporator were all combined and evaporated in the first stage continuous evaporator [LaBone 1996, p. 10].

### 2.2.2.3 F-Area A-Line Facility

Even though there were A-Lines in both F and H canyons, the one in H was abbreviated and did not perform the same function as the 221-F A-Line which processed uranium for both F and H areas for many decades [Reed et al. 2013, p. 181].

The A-Line Facility accepted dilute depleted uranyl nitrate solution from the 221-F Facility and converted it into uranium trioxide powder. Dilute uranyl nitrate solution was sent to A-Line from the solvent extraction process in the 221-F Facility where it was initially decanted to remove tributyl phosphate n-paraffin solvent and concentrated by evaporation. A second evaporation yielded a hydrate of uranyl nitrate that was heated in a denitrator to yield the uranium trioxide product.

The A-Line Facility had a dissolver to prepare a uranyl nitrate solution from the uranium trioxide product and nitric acid when it was required as a process feed stream back into the solvent extraction process in the 221-F Facility. The dissolver was also used to recycle offstandard A-Line product. The uranium trioxide product was placed into drums for storage on site or into packages for offsite shipment. Radionuclides of concern for the A-Line facility are listed in Table 2-9.

Table 2-9. 221-F A-Line Facility.<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	221-F A-Line Facility
<b>Period</b>	Operations from July 1955 to 2006
<b>Radionuclides of concern</b>	U-234, U-235, U-236, U-238

a. Sources: LaBone [1996, p. 7]; Till et al. [2001, p. 84].

The A-Line Facility can be broken down into unit operations that performed specific functions in the creation of the final product [LaBone 1996, pp. 5–6]. These processes consisted of:

- Evaporation. Decanted uranyl nitrate solution from the 221-F Warm Canyon was concentrated by steam heating.
- Purification. Purification was an optional step. Once the uranyl nitrate solution was concentrated it was sometimes sent to the silica gel purification system, which absorbed residual zirconium and niobium in the uranyl nitrate solution to ensure the uranium trioxide product met certain specifications. The waste from this process was either neutralized and sent to underground storage or collected in a low oxalic acid waste tank and transferred to the seepage basin.
- Hydrate evaporation. Hydrate evaporation is the process in which uranyl nitrate from the silica gel beds or evaporator was fed to the hydrate evaporators and again concentrated. Condensate from the evaporation was sent to the seepage basin.
- Denitration. During denitration, material from the hydrate evaporation was received and converted from uranyl nitrate hexahydrate to uranium trioxide.
- Material handling. The uranium trioxide powder was removed from the denitrators using suction wands that sent the product to a dust collector where it was placed into storage bins from which it was loaded into containers. All process equipment in A-Line that handled uranium trioxide powder was vented to the oxide fines recovery system. The purpose of this was to minimize the amount of material that could reach the atmosphere.
- Fume recovery system. The fume recovery system converted oxides of nitrogen into nitric acid for reuse using exhausters, a venturi scrubber, offgas coolers, an absorption column, and various pumps and tanks. The venturi scrubber removed uranium trioxide particles from the denitrator offgas and was the only piece of the system of radiological concern.
- Oxide dissolution. Occasionally, 221-F required a uranyl nitrate acid solution for use as a process stream. The dissolver in A-Line prepared this solution by dissolving uranium trioxide in nitric acid.

#### 2.2.2.4 221-F B-Line Facility

Two “B-Lines” for plutonium production were originally constructed within the 221-F canyon building [Reed et al. 2013, p. 181]. F-canyon was shut down for improvements to the PUREX process in February 1957 and reopened in March 1959. During this period, a new, larger B-Line called the “Jumbo B-Line,” or “JB-Line,” was constructed onto the roof [Reed et al. 2013, p. 186].

This B-Line began operations in 1959 [Natural Resources Defense Council (NRDC) 1992, p. 44]. SRS decommissioned and closed FB-Line by 2005 [Hadlock 2006, p. 5]. FB-Line converted plutonium nitrate into plutonium metal or plutonium oxide. It also had the capability to recover plutonium from onsite and offsite scrap. Initial plutonium production in this facility was primarily of  $^{239}\text{Pu}$  in a dilute nitric acid solution. The form of  $^{239}\text{Pu}$  changed through various chemical processes.

The individual processes in the FB-Line consisted of [LaBone 1996, pp. 7–9]:

- Cation exchange. The dilute plutonium product solution was transferred from the canyon storage tanks to the FB-Line receiving tanks. To ensure criticality did not occur, the product was cooled and filtered to remove dibutylphosphate and plutonium. The material was then



sent through a cation exchange column that removed the remaining plutonium along with some other impurities.

- Precipitation and filtration. The product solution from the cation exchange system was initially precipitated using hydrofluoric acid, which created plutonium trifluoride crystals to be filtered from the solution. The plutonium trifluoride that remained attached to the precipitator tanks could oxidize and produce plutonium tetrafluoride, an undesirable product, which required strict controls to minimize its formation. The solution left from this process was tested and transferred without evaporation to the F-Area Tank Farm.
- Drying and conversion. In this process, the plutonium trifluoride from the filtration process was dried and roasted to produce a mixture of plutonium dioxide and plutonium tetrafluoride.
- Reduction. The mixture of plutonium dioxide and plutonium tetrafluoride was weighed, mixed with metallic calcium, placed into a magnesium crucible the inside of which was filled with magnesium oxide sand, sealed, pressurized with argon, and heated to a molten state. The plutonium metal flowed to the bottom of the crucible and formed a button-shaped solid that was cooled, separated from the calcium slag, and die-stamped for identification.
- Button finishing. Once the plutonium metal button was stamped, it was submerged in nitric acid to remove particles of sand, slag, and calcium metal; rinsed in water; and air-dried. One or 2 g of the plutonium button were removed using a drill; the drill turnings were sealed and sent to the 772-F laboratory for analysis. The button was weighed and placed in a can; the can was crimp-sealed, placed in a plastic bag, and placed in a second can. That can was crimp-sealed and sent to storage. After successful completion of the laboratory analysis and seal tests, the can was to be sent off site.
- Recovery. The recovery process took solid scrap from onsite and offsite sources, as well as miscellaneous solutions from FB-Line, and extracted the plutonium using an anion exchange resin. The scrap was initially dissolved using a mixture of aluminum nitrate and nitric acid, after which the plutonium scrap was sorbed on the anionic resin and sent back to the canyon for processing.
- Special recoveries. Certain plutonium forms from both onsite and offsite sources were not suitable for processing in the slag and crucible dissolver process discussed above. The special recovery process was used to process solutions containing plutonium that could not be prepared for anion exchange.
- Storage and vaults. All plutonium-bearing materials, both final products and recovery feed materials, were stored in the FB-Line storage vaults. Stored materials included standard and offstandard plutonium buttons, plutonium scrap metal, slag and crucibles, cabinet sweepings, plutonium laboratory solutions, plutonium alloy buttons, and plutonium oxide.

Many different mixtures of plutonium isotopes are possible in the FB-Line, but it is assumed that the mixture was either 6 wt% or 12 wt% plutonium. The radionuclides of concern are in Table 2-10. See Table 5-4 of ORAUT-TKBS-0003-5, *Savannah River Site – Occupational Internal Dose*, for specific activities and activity ratios [ORAUT 2024a].

Table 2-10. 221-F B-Line Facility.<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	221-F, FB-Line/JB-Line
<b>Period</b>	Operations from July 1959 to February 2005
<b>Radionuclides of concern</b>	Pu-239, Pu-240, Pu-238

a. Sources: LaBone [1996, p. 9]; WSRC [2007, p. 2]; Till et al. [2001, p. 84]; NRDC [1992, p. 44].

### 2.2.2.5 Multi-Purpose Process Facility in Building 221-F

SRS envisioned producing californium for commercial and medical therapeutic uses. In 1970, SRS designed and constructed a large hot cell with a crane, manipulators, shielded windows, and separations equipment in a portion of the F Hot Canyon [DuPont 1984b; Reed et al. 2013; Stoddard 1982]. Construction was completed in 1972. The facility was called the Multi-Purpose Processing Facility (MPPF). Unfortunately, the commercial market for californium did not materialize; SRS placed the MPPF on standby in 1973. In July 1978, SRS modified the MPPF to process <sup>241</sup>Am to americium oxide for the Isotope Sales Division at ORNL. SRS completed this work by 1980 [Strack 2002, p. 455; Gray et al. 1980; Reed et al. 2013]. Some <sup>241</sup>Am oxide remained; SRS stored this material in two canisters in one of the MPPF hot cells where they remained until 2003. SRS removed the containers using a manipulator and hot crane. After packaging the containers in a single pail SRS sent the material to waste operations [WSRC 2003].

SRS anticipated using MPPF to separate and ship <sup>243</sup>AmO<sub>2</sub> and <sup>244</sup>CmO<sub>2</sub>, but that work did not materialize [Burney 1972; DuPont 1980a]. However, SRS transferred about 15,000 L of solution containing americium and curium recovered from <sup>242</sup>Pu production campaigns to F-Canyon/MPPF Tank 17-1 in 1980 for the proposed work. The solution was still in the tank in January 1999. SRS proposed to vitrify the solution using cells modified for the process in MPPF [Witt 1999]. Modification of MPPF cells for vitrification work occurred in the late 1990s with some documented potential exposures to americium [Cantrell 1999]. However, DOE directed SRS to transfer the solution to the Defense Waste Processing Facility (DWPF) for vitrification [Conway 2001].

### 2.2.2.6 New Special Recovery Facility

The New Special Recovery (NSR) Facility was designed to recover plutonium from a variety of sources including scrap plutonium from offsite sources and the FB-Line Facility. The NSR was constructed on top of 221-F with completion in 1992 but was never used [Strack 2002, p. 536]. See Table 2-11.

Table 2-11. NSR.<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	NSR (constructed on Building 221-F)
<b>Period</b>	NSR was built in 1992; never operational.
<b>Radionuclides of concern</b>	None

a. Source: Strack [2002, p. 536].

### 2.2.2.7 F-Area Outside Facilities

The F-Area Outside Facilities provided general support, principally to the processing of irradiated fuels and targets in Building 221-F. The term “Outside Facilities” is used to describe a wide variety of processes, utilities, and services that are ancillary to the primary 200-F Area operations.

Many of these processes and services did not have a connection to each other in terms of function. Therefore, it might be necessary in some cases to treat each process separately in terms of radiological protection.

There were six areas of radiological concern for the F-Area Outside Facility Operations: the Acid Recovery Unit (ARU), the general purpose (GP) evaporators, and the segregated solvent (plutonium and uranium), laboratory wastes, water handling, and basin transfer tanks [LaBone 1996, pp. 15–17].

The main F-Area Outside Facility processes of radiological concern are as follows.

### Water Handling System

The water handling facilities are in the Building 211-F waste complex between the chemical storage area and the ARU. The primary equipment for water handling consists of tanks, skimmers, and coolers, all of which can contain radioactive materials. This facility was used to provide process water and acidified water streams for the canyons, retain discard water in holding tanks for analysis, and decant spent solvent from wastewater. The radionuclides of concern for the Water Handling System are shown in Table 2-12.

Table 2-12. F-Area Water Handling System.<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	221-F Water Handling System
<b>Period</b>	Operations from November 1954 to 2006
<b>Radionuclides of concern</b>	Ru-106, Nb-95, Ru-103, Zr-95

a. Source: LaBone [1996, p. 18]; Reed et al. [2013, p. 57]; SRNS [2015, p. 36].

### Acid Recovery Unit

This facility concentrated nitric acid for reuse and is in the Building 221-F complex between the GP evaporators and the water handling facilities. All material in this location can contain radioactive material. The radionuclides of concern in the ARU are shown in Table 2-13.

Table 2-13. F-Area ARU.<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	221-F ARU
<b>Period</b>	Operations from November 1954 to 2006
<b>Radionuclides of concern</b>	Zr-95, Nb-95, Ru-106, Ru-103, U-238, U-235, U-236, U-234

a. Source: LaBone [1996, p. 18]; Reed et al. [2013, p. 57]; SRNS [2015, p. 36].

### General Purpose Evaporators

The GP evaporators concentrate aqueous waste having activity that is greater than disposal limits yet low enough to be evaporated in unshielded equipment. The GP evaporator is in the Building 221 complex between the ARU and the GP waste tanks. The radionuclides of concern in the GP evaporators are shown in Table 2-14.

Table 2-14. F-Area GP evaporators.<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	221-F GP evaporators
<b>Period</b>	Operations from November 1954 to 2006
<b>Radionuclides of concern</b>	Pu-241, Ru-106, Zr-95, Nb-95, Ru-103, Pu-239, Pu-238, U-238, U-235, U-236, U-234

a. Source: LaBone [1996, p. 18]; Reed et al. [2013, p. 57]; SRNS [2015, p. 36].

### General Purpose Waste Tanks

The tanks in this facility were used to collect GP evaporator feed and to receive and dispense evaporator overheads for recycle to the process.

### Waste Handling Facilities

The F-Area waste handling facilities were used for storage and transfer of high- and low-activity wastes that were primarily from the Metallurgical Building 235-F, Savannah River National Laboratory

(SRNL), Waste Concentration Building 776-A, the Production Control Facilities, Building 772-F, and 100-Area sources. Cold or very low-level wastes were transferred to the seepage basins by way of the GP waste tankage. Low-level activity wastes were transferred to the GP evaporators for concentration. High-level wastes were pumped to the laboratory waste evaporator feed tank in the canyon [LaBone 1996]. The laboratory waste radionuclide distribution is assumed to be representative of the waste for the waste handling facilities. The radionuclides of concern in the laboratory waste are shown in Table 2-15.

Table 2-15. F-Area Outside Facilities (laboratory waste).<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	221-F Outside Facilities
<b>Period</b>	Operations from November 1954 to 2006
<b>Radionuclides of concern</b>	Pu-238, Pu-241, Pu-239, Ce-144, Zr-95, Cs-137, Nb-95, Ru-106, Cs-134, Ru-103, Ce-141, U-238, U-235, U-236, U-234

a. Source: LaBone [1996, p. 18]; Reed et al. [2013, p. 57]; SRNS [2015, p. 36].

### Segregated Solvent Facilities

Segregated solvent facilities provided solvent purification and tank storage before the solvent was returned to F-Canyon for reuse. One of the purposes of the segregated solvent facilities was to remove the degradation end products and the radioactive contaminants from the solvent.

The radionuclides of concern for the segregated solvent facilities are shown in Table 2-16 for plutonium solvent and Table 2-17 for uranium solvent.

Table 2-16. F-Area Segregated Plutonium Solvent.<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	221-F, F-Canyon Segregated Plutonium Solvent
<b>Period</b>	Operations from November 1954 to 2006
<b>Radionuclides of concern</b>	Pu-241, Ru-106, Pu-239, Zr-95, Nb-95, Ru-103, Pu-238

a. Source: LaBone [1996, p. 18]; Reed et al. [2013, p. 57]; SRNS [2015, p. 36].

Table 2-17. F-Area Segregated Uranium Solvent.<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	221-F, F-Canyon Segregated Uranium Solvent
<b>Period</b>	Operations from November 1954 to 2006
<b>Radionuclides of concern</b>	Ru-106, Zr-95, Ru-103, Nb-95, U-238, U-235, U-236, U-234

a. Source: LaBone [1996, p. 18]; Reed et al. [2013, p. 57]; SRNS [2015, p. 36].

### Transfer Tanks

Eight transfer tanks in F-Area basins provide intermediate pumping stations for those solutions transferring from Building 221-F where there is insufficient power to transfer the material in a single operation.

The radionuclide distribution in the basin transfer tanks is assumed to be similar to the water handling distribution.

### Burning Pits

A pit for burning radioactive waste was constructed in the Radioactive Waste Burial Ground, 643-G. Burning pits were created by bulldozing open trenches between 8 to 12 ft deep and from 250 to 400 ft long. The waste was burned about once a month [Till et al. 2001]. At 643-G, solvent was burned in an 8- by 8- by 4-ft steel pan with a rain shed over the top; due to corrosion, the pan was replaced in 1964 with a longitudinally halved 400-gal tank [DuPont 1973a].

Most of the radioactive materials burned at the Radioactive Waste Burial Ground were spent and degraded solvents such as tributyl phosphate and kerosene collected from the separations areas. Before being burned, the solvents were washed as part of solvent recovery [DuPont 1958a]. SRS implemented an improved solvent washing procedure in May 1955 that used a higher concentration of NaOH in the caustic wash followed by an additional water wash to remove residual caustic material and the associated activity more efficiently [DuPont 1955a,b]. A continuous solvent wash system was implemented in the 200-Area in 1956 [Christl 1956, p. 2]. As an example of solvent washing results, the Works Technical Department reported that as of May 28, 1956, hot solvent gamma activities in F-Area for the previous 4 weeks had averaged  $2.2 \times 10^4$  cpm/mL unwashed and  $4.5 \times 10^3$  cpm/mL washed [DuPont 1956a]. Residues from the washing process that were too highly radioactive to burn were stored in tanks to enable decay of short-lived fission product activity. After allowing the short-lived fission product activity in the solvent to decay for several years, most of the solvent was burned [Till et al. 2001, p. 849]. Residues were monitored by gross gamma, gross beta, and gross alpha analysis as needed before being burned [DuPont 1956b,c, 1957b,c, 1958b, 1965]. The gamma activity level in 1957 was 4,000 cpm/mL [DuPont 1957c]. By 1960, the Health Physics Section had implemented a gamma activity limit of 1,000 cpm/mL; solvents exceeding that amount could not be burned [DuPont 1960].

Information and data about the burning of solvents have been reported in some Works Technical Department monthly progress reports. A total of 170,000 gal of solvent was burned before February 1964, and it is believed that the best estimate of the total amount burned between 1956 and 1972 was 400,000 gal [Till et al. 2001].

Estimates of the amounts of solvent that was burned and of air concentrations can be found in ORAUT-RPRT-0051, *Analysis of Potential Exposures Resulting from the Burning of Degraded Solvents at the Savannah River Site* [ORAUT 2012a].

### **Sump Collection Tanks**

Two sump collection tanks occupy two separate underground vaults. One tank receives radioactive waste condensate by jet stream from sumps or tanks related to the canyon air exhaust system. It then transfers the waste to the hot or warm canyon waste evaporator feed tanks. Discharges to hot canyon cell sumps are first transferred to the second sump collection tank and then to the canyon for analysis and disposition.

### **Recycle Sump**

The Recycle Sump is between the ARU and the GP evaporators and collects drainage and overflow from the canyon auxiliary tanks and holds contaminated or recycled liquids.

### **Auxiliary Systems and Support Facilities**

The only auxiliary system and support facility of concern is the plant laundry in Building 723-F.

#### **2.2.2.8 235-F, F-Area Materials Storage Building**

Building 235-F, F-Area Materials Storage Building (FAMS) and also called the Metallurgical Building, was constructed in the 1950s and remained unused until the mid-1960s. For dose reconstruction purposes, January 1, 1960, should be used as the start of operations. During the mid- to late 1960s, Building 235-F was used to house the Special Products Fabrication Facility (SPFF) that supported facilities processing  $^{238}\text{U}$ ,  $^{237}\text{Np}$ , and  $^{239}\text{Pu}$  oxide. These facilities were known as the Slug Facility and the Alloy Line. The Slug Facility processed uranium, plutonium, and other actinide-bearing materials into irradiation target components. The Alloy Line was a set of gloveboxes with blending, welding, machining, and decontamination capabilities. The Alloy Line cabinets were removed in 1984.

In the mid-1970s, the SPFF was decontaminated and decommissioned, and the building reconfigured to support the Actinide Billet Line (ABL) mission and make room for the  $^{238}\text{PuO}_2$  Fuel Form Facility (PuFF). The ABL consisted of a glovebox line and associated equipment, and it initially fabricated plutonium billets for special reactor applications. ABL was later modified to produce special billets from  $^{237}\text{Np}$  oxide powder for extrusion into reactor targets. The ABL continued operations until 1991. The next mission of Building 235-F was the PuFF and Plutonium Experimental Facility (PEF) including a Metallography Laboratory, commonly known as the Old Metallurgical Laboratory. The PuFF produced  $^{238}\text{Pu}$  heat source pellets from 1977 until 1983. The PEF contained a line of 12 gloveboxes and 2 hoods. PEF activities included research and development on processes that manufactured  $^{238}\text{Pu}$  heat sources. Plutonium-238 was processed in the PEF from 1979 to 1981 to develop the fuel pellet fabrication process performed in the PuFF.

All metallurgical processes in Building 235-F were shut down by 1991. A drawing of the first floor of 235-F highlighting PuFF, PEF, and the AB Line (Np Billet Line) is shown in Figure 2-20.

Vaults on the first and second floors were included in the construction of 235-F [SRNS 2020a, pp. 28–30]. These vaults were used to store process materials during metallurgical processing. The last mission of Building 235-F provided for the receipt, storage within the vaults, and repackaging of containerized plutonium and uranium-bearing materials.

The storage and repackaging mission ended in 2006, the vaults were deinventoried, and all stored plutonium and shipping containers were removed from the building [SRNS 2020a, pp. 7–9]. As of 2020, the building was being decommissioned to establish it as a “Cold and Dark Transition Surveillance and Maintenance Facility” [SRNS 2020a, p. 6]. Process areas on both floors remain contaminated with some areas designated “High Contamination/Airborne Radiation Area” [SRNS 2020a, pp. 32–33].

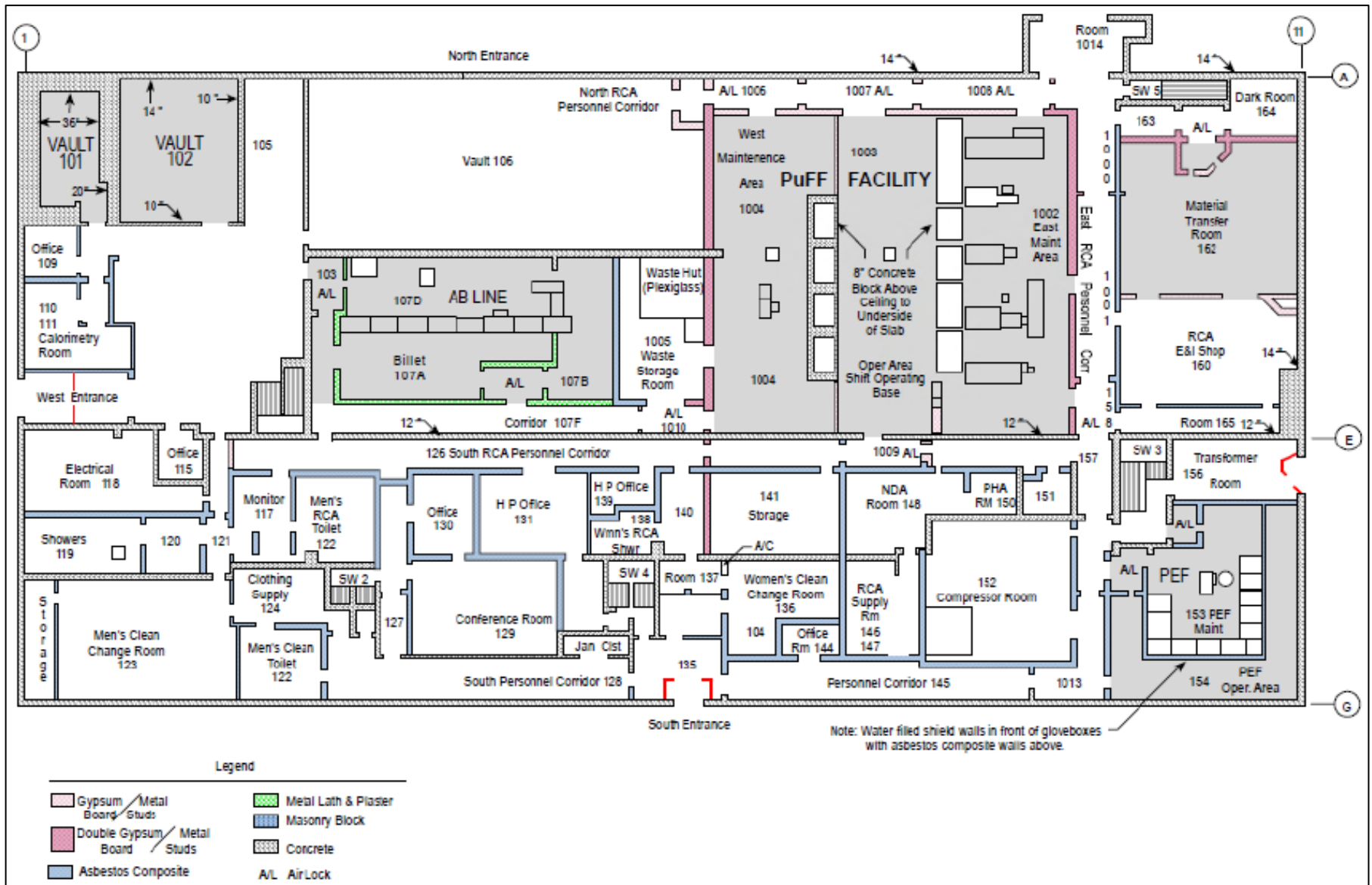


Figure 2-20. Building 235-F first floor plan [Taylor and Phifer 2012, p. 55].

### 2.2.2.9 PuFF Process

The PuFF was on the first and second floors of Building 235-F, and the PEF was on the first floor in Room 153. PEF was also known as “Little Puff” [Reed et al. 2013, p. 224]. SRS placed both facilities into production in July 1977 [DOE 1991, p. 11]. SRS processed plutonium in PuFF and PEF through 1983 when SRS put them both in standby. A layout of 235-F is given in Figure 2-21. The primary function of PuFF was to produce encapsulated plutonium oxide fuel forms by converting  $^{238}\text{Pu}$  oxide powder into fuel forms for heat sources using a powder ceramic and metallurgical process. The final products were compacted  $\text{PuO}_2$  fuel shapes. The PEF provided space for the testing of processes before they were used in the PuFF. PEF was staffed by SRNL. The plutonium oxide blends used in the PuFF and PEF were very similar.

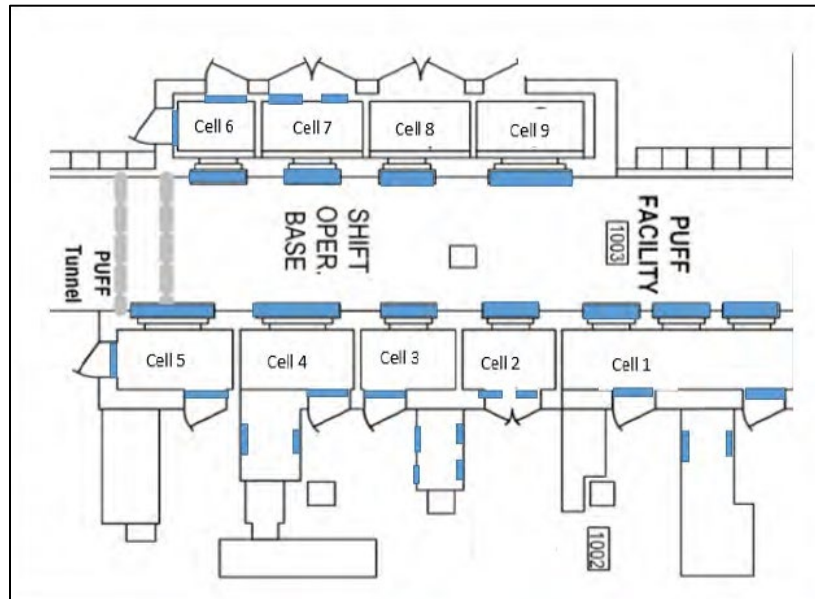


Figure 2-21. Layout of 235-F [Clendenning 2019, p. 9].

SRS fabricated and encapsulated  $\text{PuO}_2$  in six manipulator cells filled with inert gas, three air-filled manipulator cells, five wing cabinets, and one hood. The process enclosures were on the first floor of the PuFF. The manipulator cells were divided into two parallel and facing lines with an operating area between and separate maintenance area behind the lines as shown in Figure 2-21. Although operations with plutonium ceased in 1983, the hot cells remained contaminated with  $^{238}\text{Pu}$  [DOE 1991, p. 42]. SRS has removed most other equipment from PuFF.

SRS used 4.5 kg (0.45 mCi) of unirradiated thorium oxide reflector pellets as a surrogate to test the effects of heat treatment on physical characteristics and dissolution of heat sources [DOE 1978a]. This work, performed in the high-level caves, involved mechanical grinding of  $\text{ThO}_2$ . Beginning in 1980 and continuing into 1981, thorium was used as a doping agent of the iridium-welding agents for the fabrication of General Purpose Heat Sources [NIOSH 2012].

SRS cast and machined alloys of uranium including highly enriched uranium in the mid-1960s to 1970 in the Special Alloy Line in the building before construction of PuFF. However, plutonium was documented as the dominant source of exposure [DOE 1996; Farrell and Findley 1999].



### 2.2.2.10 PEF Process

To test the PEF for potential  $^{238}\text{Pu}$  heat source development, SRS used about 300 g  $\text{ThO}_2$  as a surrogate in 1978 [DuPont 1978]. By April 1978, all PEF equipment except for the hot press had been tested using  $\text{ThO}_2$  as a surrogate in gloveboxes [DOE 1978b, p. 3].

The PEF converted  $\text{PuO}_2$  powder into fuel forms for heat sources by using powdered ceramic and metallurgical processes [Morris et al. 2000, p. 2]. The final products were dense  $\text{PuO}_2$  fuel forms including pellets. The test fuel  $^{239}\text{Pu}$  and  $^{238}\text{Pu}$  was fabricated in 13 gloveboxes, 8 of which were supplied with recirculated inert gas. The interconnected gloveboxes were removed from 235-F in the 2000s.

The radionuclides of concern at the PuFF and the PEF are shown in Table 2-18.

Table 2-18. PuFF and PEF.<sup>a</sup>

Area	F
Description	235-F, PuFF and PEF
Period	235-F operations began in the mid-1960s. PuFF operations between July 1977 and 1987; PEF operations between 1979 and 1981; Building 235-F is scheduled for decommissioning.
Radionuclides of concern	Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Th-232 (1975–1981), Np-237, Am-241

a. Sources: DuPont [1977a, p. 51]; LaBone [1996, p. 21]; DOE [1991, pp. 11, 42]; Defense Nuclear Facilities Safety Board [2013, p. 55]; WSRC [2001b, pp. 86–87]; U.S. Environmental Protection Agency (EPA) et al. [1993–2022, p. 267]; [SRNS 2020a, pp. 7–9].

### 2.2.2.11 235-F Vaults

The 235-F Facility received plutonium or plutonium dioxide and fabricated it into a variety of items for the nuclear programs. SRS built vaults in 235-F to store plutonium at different production steps before completion. They were made to provide safe storage and establish safeguards for security purposes. Europium, thorium, and uranium have also been stored in these vaults. Three vaults were in 235-F; SRS emptied all three vaults in 2006 [Taylor and Phifer 2012, p. 20]. The F-Area material storage in 235-F is planned to be decommissioned [EPA et al. 1993–2022, p. 267]. For dose reconstruction purposes the start of operations 1960, should be the same as used for 235-F, January 1, 1960.

#### Plutonium Oxide Vault

The Plutonium Oxide Vault was used to store plutonium oxide powder from the PuFF, PEF, and the ABL facility. Neptunium-237 powder and billets were also stored in this facility [LaBone 1996, p. 22].

#### Finished Product Vault

The Finished Product Vault was used for the storage of uranium and transuranic (TRU) materials in powder form like the plutonium oxide vault. In addition, this vault contained a water-cooled shielded storage area for finished  $^{238}\text{Pu}$  products produced in the PuFF and PEF in the form of canned encapsulated pellets and spheres welded into cans [LaBone 1996, p. 22].

#### Scrap Vault

The Scrap Vault was used to store metal and oxide scraps of uranium, thorium, and plutonium in the form of chips, sweepings, powder, blended samples, or metal turnings [LaBone 1996, p. 22].

The radionuclides of interest in the three vaults are as shown in Table 2-19.

Table 2-19. 235-F Plutonium Oxide Vault, Finished Product Vault, and Scrap Vault.<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	235-F, 235-F vaults
<b>Period</b>	Operations began in the mid-1960s and ended October 2006. Building 235-F remains contaminated and is scheduled for decommissioning.
<b>Radionuclides of concern</b>	Np-237, U-235, Pu-239, Pu-238, Th-232

a. Sources: LaBone [1996, p. 22]; Taylor and Phifer [2012, p. 20]; EPA et al. [1993–2022, p. 267]; [SRNS 2020a, pp. 7–9].

### 2.2.2.12 Neptunium Oxide Facility

SRS began processing <sup>237</sup>Np into oxide on a pilot scale in 1958 [DuPont 1958c], then began production processing into fuel billets and tubes in Building 235-F in 1963 [Poe et al. 1963; SRNS 2020a, p. 9]. Operations to fabricate neptunium-oxide billets continued through 1984 [Fuller et al. 2013]. In the beginning, SRS separated neptunium from PUREX waste in F-Canyon but discontinued this process in 1965; thereafter SRS separated neptunium from irradiated neptunium targets in a specialized cell in H-Canyon [Groh et al. 2000]. SRS separated neptunium from <sup>238</sup>Pu in the HB-Line into an oxide form. SRS transferred the neptunium oxide to Building 235-F to be blended with aluminum powder and fabricated into irradiation targets for SRS reactors. Neptunium oxide shipments to 235-F would typically be in batches of about 1 kg each with total shipment weights ranging between 2 and 25 kg [DuPont 1977a,b,c,d]. Batches of neptunium oxide from HB-Line were contaminated with <sup>238</sup>Pu.

Plutonium contamination is important to note because, as the plutonium-to-neptunium ratio increases, the plutonium becomes the alpha emitter of radiological concern as demonstrated in Table 2-20, which assumes specific activities for <sup>238</sup>Pu of 17.1 Ci/g and for <sup>237</sup>Np of 0.00069 Ci/g.

Table 2-20. Plutonium-to-neptunium alpha ratio.

Np wt%	Pu wt%	Pu:Np alpha ratio
99.0	1.0	250:1
99.5	0.5	125:1
99.9	0.1	25:1
99.95	0.05	12:1
99.99	0.01	2.5:1
99.995	0.005	1.2:1
99.999	0.001	0.25:1

SRS fabricated the first target in the form of a canned slug. The slug consisted of a compacted blend of neptunium oxide and was clad in an aluminum can [ORAUT 2016]. In the mid-1960s, the design of the neptunium target was changed to unalloyed tubular elements as more neptunium became available for irradiation. The tubular design improved heat transfer, which increased production and purity of the <sup>238</sup>Pu by allowing operation at higher neutron fluxes. The designs of the tubular element were modified three times between 1966 and 1978. The target designs used differing amounts of neptunium so that the final tubular targets varied from 120 to 190 g of neptunium per foot of active tube length, and the tube diameter varied between 3 and 3.7 in. Starting in 1968, billets were fabricated into alloyed tubes in Building 235-F. To create a contamination shield, billets were placed in the space between the two sleeves. A top fitting with a breather tube was welded in place. The use of the contamination shield allowed the billet to be removed from containment cabinets in Building 235-F and transported to Building 321-M for outgassing and extrusion [ORAUT 2016].

SRS suspended fabrication of neptunium targets from October 1979 to April 1980 while using the neptunium billet line and related processes to make plutonium targets for the <sup>242</sup>Pu program [DuPont 1980b,c]. After restart, SRS continued fabrication of neptunium targets until 1984 [DuPont 1980a; Fuller et al. 2013; Jordan et al. 2009; Rogers et al. 1989].

Neptunium nitrate solutions were received from H-Canyon and F-Canyon and transferred to a receipt tank in the Old HB-Line. Solutions of plutonium were normally received from the Scrap Recovery Facility. On a limited basis, dilute  $^{239}\text{Pu}$  solutions were received from H-Canyon. The solutions were precipitated to yield neptunium and plutonium oxalate. The neptunium and plutonium oxalate cakes were then loaded into a furnace and calcined to  $\text{NpO}_2$  and  $\text{PuO}_2$ . The flow of neptunium at SRS is diagramed in Figure 2-22.

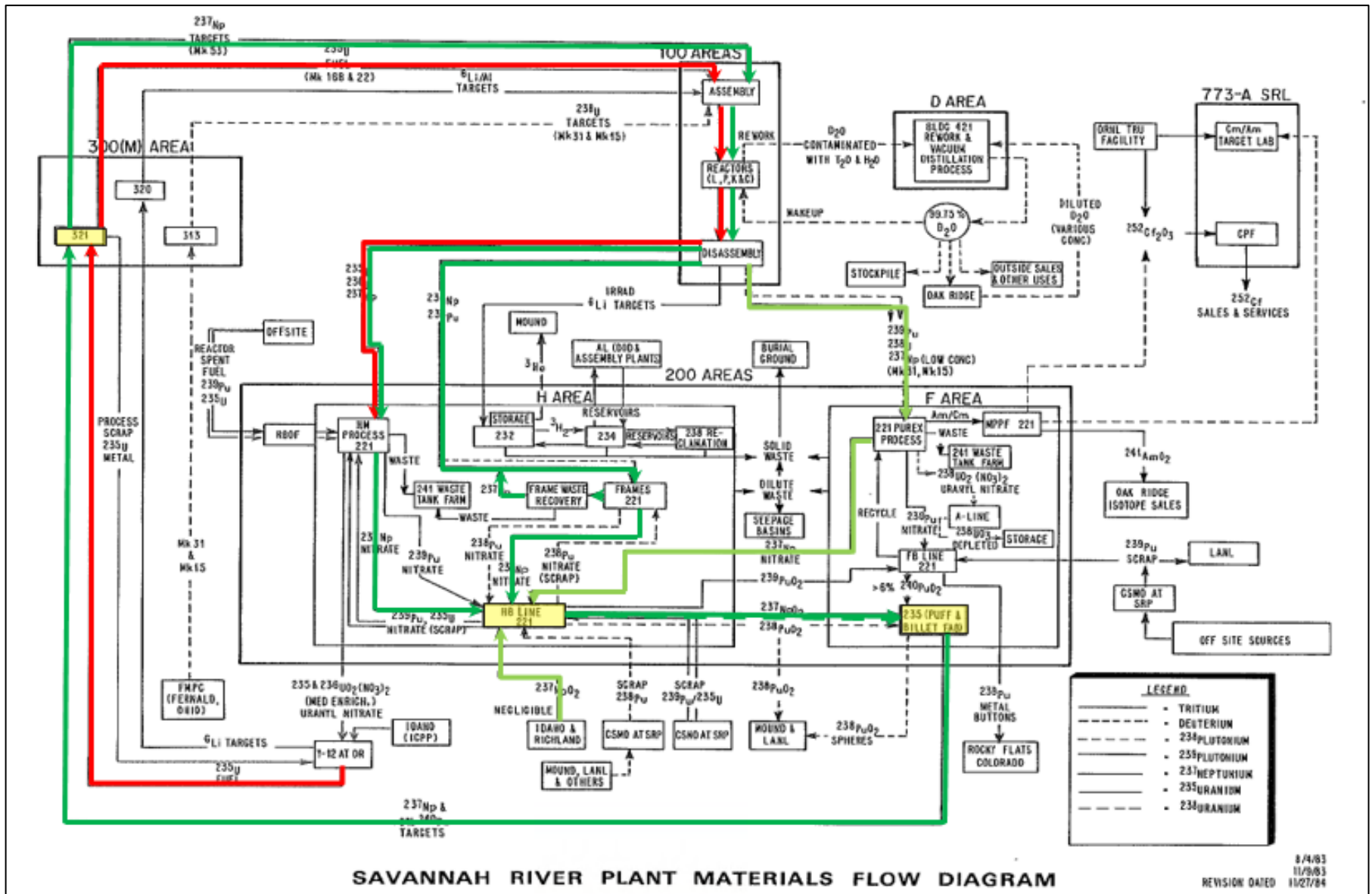


Figure 2-22. Flow of neptunium [modified from DuPont 1984c, p.3].

### 2.2.2.13 772-F and 772-1F Central Laboratory Facility

The production control laboratories, 772-F and 772-1F, were near the 221-F Canyon and provided analytical support for the 200-F and 200-H separations processes. These laboratories have also been referred to as the Central Laboratory Facility (CLAB) and F/H-Lab. The main purpose of the laboratory was to ensure the purity of the uranium and plutonium from the separation processes [Reed et al. 2013, p. 151]. Support was also provided to Waste Management, Reactors, and Raw Materials. Building 772-F was connected to the F-Canyon with a 167-ft-long covered causeway referred to as “the tunnel” that was used to transport shielded samples to the laboratory [Reed et al. 2013, p. 152].

Buildings 772-F and 772-1F are scheduled for decommissioning [EPA et al. 1993–2022, p. 272].

The radionuclides of concern are primarily of plutonium, americium, and fission products as well as, to a lesser degree, uranium and tritium (Table 2-21). Radionuclides of concern are given by waste stream location in Farrell and Findley [1999].

Table 2-21. 772-F and 772-1F Production Control Laboratories.<sup>a</sup>

Area	F
Description	772-F and 772-1F Production Control Laboratories
Period	772-F constructed in 1953; 772-1F constructed in 1987; operations ended February 2021; scheduled for decommission.
Radionuclides of concern	Fission products, Am-241 (with Pu only), U, Pu, H-3

a. Sources: LaBone [1996, p. 23]; DOE [2005, p. 46]; Durant and Pritchard [1986, pp. 162–163]; EPA et al. [1993–2022, p. 272]; SRNS [2023].

The major sources of external exposure were <sup>238</sup>Pu spontaneous fission neutrons and betas from <sup>238</sup>Pu progeny along with gamma, X-ray photons, and betas from fission products. While Farrell and Findley [1999] directly applies to work in 1999 and later, the information can be used retrospectively. It stated:

- **CLAB Waste Stream A.** Stream A consisted of waste from Building 772-1F Laboratory Modules L120, L122, L126, L127, and L130. The Stream A radionuclides of concern were plutonium isotopes.
- **CLAB Waste Stream B.** Stream B consisted of waste from Building 772-1F Laboratory Modules L125, L128, L129, L132, L137, L156, 772-F Cells (including cell exhaust system), and 772-1F Filter Room. The Stream B radionuclides of concern were plutonium isotopes.
- **CLAB Waste Stream C.** Stream C consisted of waste from Building 772-1F Laboratory Modules L108, L121, L133, L149, L150, L152, Off Gas Exhaust, Radiological Control Operations Exhausts, 772-1F drain line exhaust, 772-1F vacuum systems and Building 772-F Shielded Area/Fan Room, Liquid Byproduct Tanks, Laboratory Module L162, and the 772-4D Area. The Stream C radionuclides of concern were plutonium isotopes.
- **CLAB Waste Stream D.** Stream D consisted of waste generated in Building 772-1F Laboratory Module L134. The Stream D radionuclides of concern were plutonium and uranium isotopes.
- **CLAB Waste Stream E.** Stream E consisted of waste from Building 772-1F Laboratory Modules L131, L138, L139, L146, L147, L154, L155, and L171. The Stream E radionuclides of concern were <sup>241</sup>Am and plutonium isotopes.
- **CLAB Waste Stream F.** Stream F consisted of waste from Building 772-1F Laboratory Modules L126, L130, L158, L159, and L163. The Stream F radionuclides of concern were plutonium isotopes.

- CLAB Waste Stream G. Stream G consisted of waste from the Building 772-F/772-4F filter room. The Stream G radionuclides of concern were plutonium isotopes and  $^{241}\text{Am}$ .
- CLAB Waste Stream H. Stream H consisted of waste from Building 772-F Laboratory Modules L142, L143, L146, L147, L174, L175, and L183. The Stream H radionuclides of concern were  $^{241}\text{Am}$  and plutonium isotopes.
- CLAB Waste Stream I. Stream I consisted of waste from Building 772-F Laboratory Modules L110, L111, and L119. The Stream I radionuclide of concern was tritium.
- CLAB Waste Stream J. Stream J consisted of waste from the Building 772-F Laboratory Modules, which is disposed of in the 772-F Low Curie Salt drum. The Stream J radionuclides of concern were  $^{241}\text{Am}$ , tritium, and plutonium isotopes.

#### 2.2.2.14 200-F Separations Area Plutonium Storage Facility

The Plutonium Storage Facility (PSF) is in the 200-F Separations Area and was designed to receive, store, monitor, retrieve, and ship packaged plutonium as needed. The PSF was never used, and in 1993 it was placed in “indefinite cold standby” [DOE 1996, p. 210]. Table 2-22 shows there are no radionuclides of concern for the PSF.

Table 2-22. PSF, receipt, storage, monitoring, retrieving, and shipping plutonium.<sup>a</sup>

<b>Area</b>	F
<b>Description</b>	200-F, PSF
<b>Period</b>	PSF was built in 1987, never operational.
<b>Radionuclides of concern</b>	None

a. Source: DOE [1996, p. 210].

#### 2.2.2.15 F- and H-Areas Effluent Treatment Facility

The Effluent Treatment Facility (ETF) consists of large, impermeable storage basins (with lift stations in both F- and H-Areas) and a treatment facility, Building 241-84H in H-Area. ETF was designed to remove hazardous chemical and radioactive contaminants from the 200-Area liquid effluent waste streams and concentrate them for disposal. The effluent processing steps include [LaBone 1996, pp. 25–26]:

- Feed adjustment. The 200-Area effluent stream is pretreated to enhance the ETF process and eliminate potential process problems. Pretreatment operations consist of oxidation, grit collection and removal, and pH adjustment.
- Filtration. The filtration system consists of three parallel trains with three stages of filtration. Feed is pressurized by the filter pumps and passes through the tubular, ceramic, and crossflow elements. At appropriate intervals, the filters are back-pulsed with air to remove the deposited materials.
- Organic removal. Organic removal consists of three mercury removal ion exchange columns in parallel followed by two activated carbon beds in series.
- Reverse osmosis. The bulk of the solid material in the 200-Area liquid effluent waste is removed using a three-stage reverse osmosis process that concentrates dissolved solids. Reverse osmosis concentrates dissolved solids without adding material to the flow stream. The water that passes through the membranes is directed to the ion exchange pH adjustment tank. The concentrated liquid on the upstream side of the third membrane, containing the

dissolved material that the membranes have concentrated, is either recycled or directed to the evaporator feed tank.

- Ion exchange. The filtrate from the reverse osmosis process is the feed solution to a system of four cation columns and two mercury removal columns. The cation and mercury resins remove ionic species of radionuclides and serve as a final polishing step in the process.
- Evaporation. Feed for the evaporation process consists of concentrate from the filtration and reverse osmosis systems, regeneration solution from the ion exchange process, and the cleaning solution from the filtration and the reverse osmosis processes. Evaporator steam condensate is collected in a condensate receiver and pumped to the H-Area segregated water system for analysis. The overheads are transferred to the organic removal feed tank or recycled to the wastewater collection tanks for reprocessing. The evaporator bottoms are sent to the H-Area Tank Farm for transfer to the Saltstone Disposal Facility.

Table 2-23 shows the radionuclides of concern for the F- and H-Areas ETF.

Table 2-23. F- and H-Areas ETF.<sup>a</sup>

<b>Area</b>	F and H
<b>Description</b>	241-84H, ETF
<b>Period</b>	Operations began in 1988. Currently operating.
<b>Radionuclides of concern</b>	Ce-144, Zr-95, Sr-89, Ru-106, Cs-137, Cs-134, Sr-90, Pu-241, Pu-238, Pu-239

a. Sources: LaBone [1996, p. 26]; Strack [2002, p. 541]; SRNS [2023].

### 2.2.3 H-Area

H-Area was designed to be very similar to F-Area. Both areas had support facilities for the generation of process steam, water and air supply, filtration, and materials storage, but an analytical laboratory was not built in H-Area. Analytical analysis for H-Area samples were performed in Building 772-F in F Area [Strack 2002, pp. 379–380]. The F-Canyon was deactivated in 2006. H-Canyon continues to operate [SRNS 2015, p. 36].

#### 2.2.3.1 221-H H-Canyon

The initial design of the H-Canyon building was similar to that of F-Canyon. A photo of 221-H is shown in Figure 2-23.

Since 1959, H-Canyon has been devoted primarily to the recovery of enriched uranium from fuels irradiated in SRS reactors using a modified PUREX process, the HM process. Once separated, highly enriched uranium product solution is transferred out of H-Canyon and shipped to the Y-12 Plant at Oak Ridge for conversion into uranium metal to be recycled as driver fuel for the SRS production reactors. Enriched uranium has been recovered from fuels from domestic research reactors and from foreign research reactors fueled with material produced or enriched by the United States [WSRC 2000a, pp. 21–22]. The HM process was modified to recover neptunium by isolating most of the neptunium in a new 1BP stream (1BP Aqueous Product from 1B Bank in first cycle solvent extraction) to reduce the loss to the 1AW stream by adjusting the 1AF stream with additional ferrous sulfamate. The flowchart for recovering neptunium from enriched uranium is shown in Figure 2-24.



Figure 2-23. 221-H Canyon building [Reed et al. 2013, p. 23].

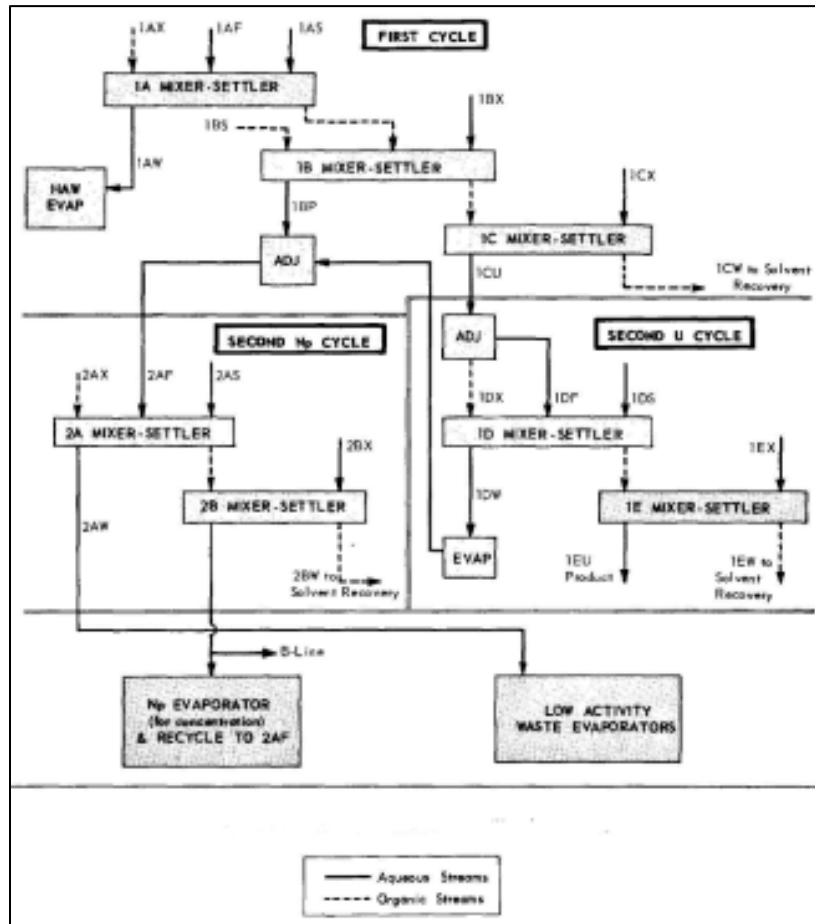


Figure 2-24. Process for neptunium recovery from enriched uranium [Poe et al. 1963, p. 19].



### 2.2.3.2 Separations Processes

In terms of the radionuclides of concern, H-Canyon separates uranium, neptunium, plutonium, and fission products from irradiated uranium fuel. While americium is not listed in Tables 2-24 through 2-27, Farrell and Findley [1999] lists  $^{241}\text{Am}$  as part of the H Canyon waste stream and should be assigned to dose reconstructions for workers in H Canyon facilities.

The radionuclides of concern vary with the age of the material.

#### 2.2.3.2.1 Uranium Cycles

In the first uranium cycle, standard uranium fuel is dissolved in nitric acid catalyzed by mercuric nitrate. The dissolved fuel is clarified in the head end, and the adjusted solution is fed into the solvent extraction system. Uranium, plutonium (not named but first plutonium cycle), and neptunium are extracted from the fission products and separated from each other in the first cycle. Each solution then goes through two separate cycles of solvent extraction for further purification. The uranium product solution is transferred out of the building for further processing. The neptunium product solution is transferred to the HB-Line for oxide conversion [ORAUT 2013a,b, 2016]. Table 2-24 lists the radionuclides of concern for the first uranium cycle, and Table 2-25 lists those for the second cycle.

Table 2-24. H-Canyon, First Uranium Cycle Stream.<sup>a</sup>

<b>Area</b>	H
<b>Description</b>	221-H, H-Canyon First Uranium Cycle Stream
<b>Period</b>	Operations began July 1955 and ended June 2021.
<b>Radionuclides of concern</b>	Ce-144, Zr-95, Sr-89, Cs-137, Ru-106, Sr-90, Pu-241, Pu-238, Pu-239, U-234, U-236, Np-237, U-235, U-238

a. Sources: LaBone [1996, p. 42]; Strack [ORAUT 2016]; SRNS [2023].

Table 2-25. H-Canyon, Second Uranium Cycle Stream.<sup>a</sup>

<b>Area</b>	H
<b>Description</b>	221-H, H-Canyon Second Uranium Cycle Stream
<b>Period</b>	Operations began July 1955 and ended December 2020.
<b>Radionuclides of concern</b>	Ce-144, U-234, Zr-95, Sr-89, U-236, Cs-137, Ru-106, Sr-90, U-235, Np-237, Pu-241, Pu-238, U-238, Pu-239

a. Sources: LaBone [1996, p. 43]; ORAUT [2016]; SRNS [2023].

#### 2.2.3.2.2 Neptunium Cycle

Irradiated neptunium oxide targets, originally in a slug design but later changed to a tube, contained unconverted  $^{237}\text{Np}$ , reactor product  $^{238}\text{Pu}$ , and fission products. Irradiated targets were cooled at least 45 days to allow decay of the short-lived fission products. Dissolution of targets required up to 48 hours and then only about 85% dissolution was obtained. The remaining undissolved target material was carried over and dissolved with the subsequent batches of targets but with significant plutonium contamination. Anion exchange was used to separate aluminum, remaining fission products, and other impurities from the  $^{238}\text{Pu}$  product and residual  $^{237}\text{Np}$ . Plutonium and neptunium were absorbed by the resin column to separate them from aluminum, fission products, and other cationic impurities. Plutonium was separated from the neptunium by washing the column with nitric acid containing ferrous sulfamate and hydrazine to reduce the plutonium to the (III) valence state. The separated neptunium and plutonium were processed by one or more additional cycles of anion exchange, as needed, to obtain the required purity of  $^{238}\text{Pu}$  product and residual  $^{237}\text{Np}$  [Groh et al. 2000].

SRS developed a frame assembly for neptunium recovery from irradiated targets in H-Canyon. All canyon maintenance work was designed to be done remotely [ORAUT 2013a,b]. Five modular frames were installed in canyon spaces, each approximately 10 ft square by 17 ft high with a standard arrangement of pipe nozzles to supply services to process waste streams and irradiated targets. SRS installed five columns and a dissolver for neptunium-plutonium separation from irradiated neptunium targets. The size of the ion exchange columns was small in comparison with regular canyon equipment; this permitted development of a concept in which a number of columns and tanks could be mounted in a frame that could be installed as one modular unit. Electrical and other services were piped to equipment pieces as part of the permanent frame structure. Additional services were required though, primarily for pneumatic liquid level and specific gravity instrumentation for the several vessels in each frame. Bundles of up to six small stainless-steel tubes were drawn through 3-in. pipes embedded in the canyon shielding walls. No significant problems were reported with the use of these frames and piping [Poe et al. 1963, p. 6; Groh et al. 2000, p. 9]. A photo of a canyon frame is shown in Figure 2-25.

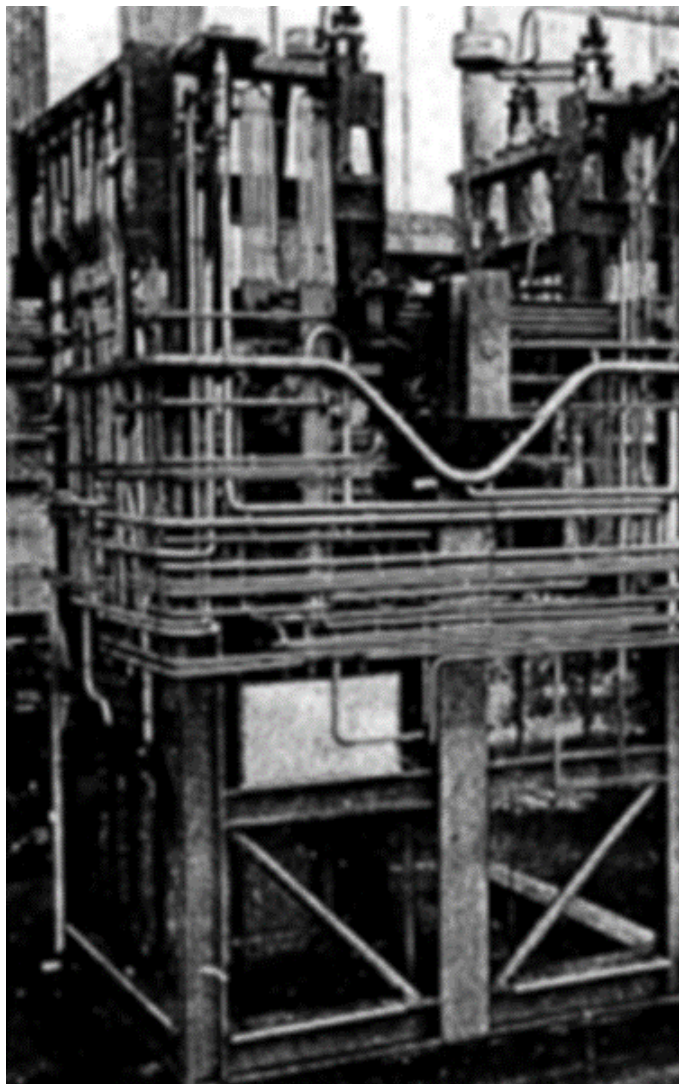


Figure 2-25. Neptunium-plutonium frame without ion exchange columns [Groh et al. 2000, p. 9].

Table 2-26 lists the radionuclides of concern for the neptunium second cycle.

Table 2-26. H-Canyon, Neptunium Second Cycle.<sup>a</sup>

<b>Area</b>	H
<b>Description</b>	221-H, H-Canyon Neptunium Second Cycle Stream
<b>Period</b>	Operations began July 1955 and ended December 2020.
<b>Radionuclides of concern</b>	Zr-95, Ce-144, Sr-89, Ru-106, Pu-241, Cs-137, Pu-238, Sr-90, Np-237, Pu-239, U-236, U-235, U-234

a. Sources: LaBone [1996, p. 43]; Strack [2002, p. 564]; SRNS [2023].

The separations process flowchart for the recovery of neptunium and plutonium from irradiated neptunium targets is shown in Figure 2-26. The plutonium oxide was either shipped off site or sent to Building 235-F for formation into heat sources. The neptunium oxide was sent to Building 235-F for refabrication into billets. Once an adequate neptunium inventory was accumulated by 1965, isolation and recovery of neptunium from the PUREX process was discontinued [Groh et al. 2000].

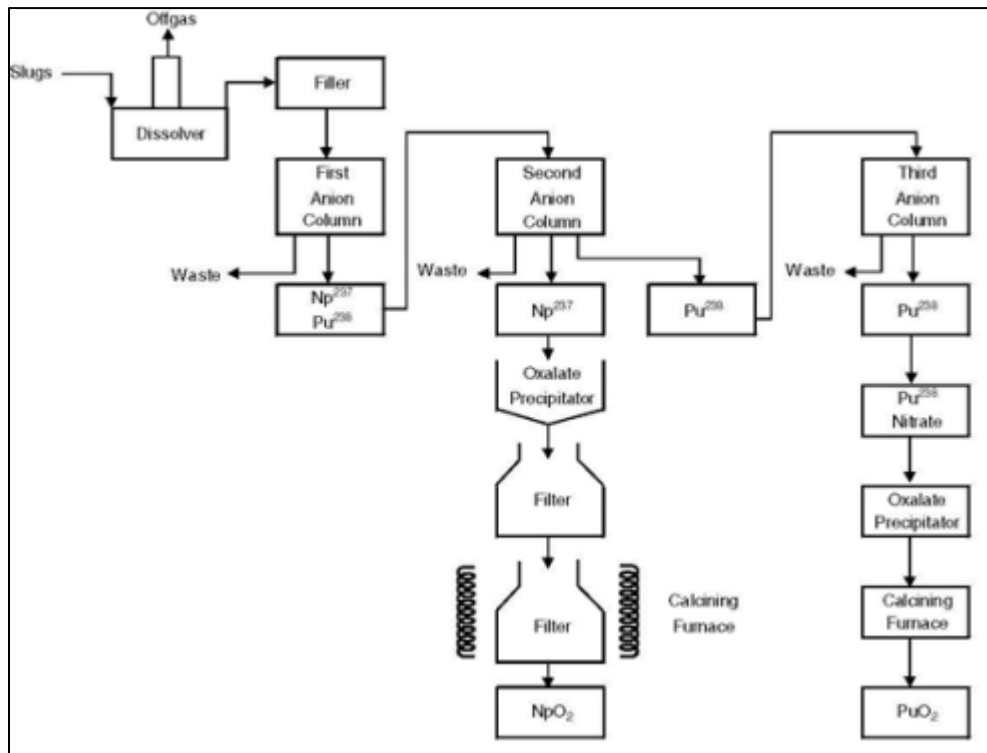


Figure 2-26. Process for recovery of neptunium from irradiated targets [Groh et al. 2000, p. 7].

Table 2-27 lists the radionuclides of concern for the H-Canyon work stream.

Table 2-27. H-Canyon, Work Stream.<sup>a</sup>

<b>Area</b>	H
<b>Description</b>	221-H, H-Canyon Work Stream
<b>Period</b>	Operations began July 1955.
<b>Radionuclides of concern</b>	Ce-144, Zr-95, Sr-89, Cs-137, Ru-106, Sr-90, Pu-241, Pu-238, Pu-239, Np-237, U-234, U-236, U-235, U-238

a. Sources: LaBone [1996, p. 44]; Strack [2002, p. 564].

### 2.2.3.2.3 Special Programs

In addition to the main process operations, various special programs have been conducted. For example, a plutonium mixture containing appreciable  $^{238}\text{Pu}$  was recovered by solvent extraction from the fission product waste stream of the uranium cycle. A thorium nitrate solution was purified and concentrated for offsite use. Irradiated thorium was processed in the solvent extraction system to recover  $^{233}\text{U}$  [LaBone 1996, p. 42].

### 2.2.3.3 HB-Line Plutonium Oxide Facility (Old HB-Line)

Plutonium nitrate solution from H-Canyon was converted to plutonium oxide powder by the oxalate precipitation and calcinations method. The product was shipped to Building 235-F or to offsite facilities for fabrication into heat sources [LaBone 1996, p. 40]. The radionuclides of concern in the Plutonium Oxide Facility are shown in Table 2-28.

Table 2-28. Old HB-Line,  $\text{NpO}_2$ , and  $\text{PuO}_2$ .<sup>a</sup>

<b>Area</b>	H
<b>Description</b>	221-H, Old HB-Line
<b>Period</b>	Operations from July 1955 to 1984
<b><math>\text{UO}_2</math> material radionuclides of concern</b>	U-233
<b><math>\text{NpO}_2</math> material radionuclides of concern</b>	Np-237
<b><math>\text{PuO}_2</math> material radionuclides of concern</b>	Pu-238, Pu-241, Pu-239

a. Sources: Separated U-233 between 1964 and 1969 [Till 2001, p. 102]; LaBone [1996, p. 41]; Reed et al. [2013, p. 220]; Strack [2002, p. 564].

The radionuclides of concern in the scrap material are either  $^{238}\text{Pu}$  in oxide form or a  $^{235}\text{U}/^{239}\text{Pu}$  mixture.

### 2.2.3.4 HB-Line Facility (New HB-Line)

The new HB-Line was designed to replace the aging HB-Line production facility (Old HB-Line). The line, built into the H-Canyon building, consists of three separate facilities: (1) the Scrap Recovery Facility, (2) the Neptunium Oxide Facility, and (3) the Plutonium Oxide Facility.

#### Scrap Recovery Facility (Phase I)

The Scrap Recovery Facility is designed to routinely generate nitrate solutions of  $^{238}\text{Pu}$  or  $^{235}\text{U}/^{239}\text{Pu}$  scrap suitable for purification by anion exchange or solvent extraction in the canyon. Scrap is received and dissolver batches prepared based on assay of the scrap. The solid scrap is dissolved in hot nitric acid containing trace fluoride ions, transferred through a filter bag, collected in a tank, sampled for accountability and process control, then diluted with nitric acid and transferred to the proper canyon vessel as a nitrate solution. Duplicate lines are installed in this recovery facility to provide the required flexibility and maintain recovery operations to return all recovery products to the main streams.

The radionuclides of concern in the scrap material are composed of either  $^{238}\text{Pu}$  in oxide form or a  $^{235}\text{U}/^{239}\text{Pu}$  mixture. The  $^{235}\text{U}/^{239}\text{Pu}$  mixture normally contains greater than 20% enriched uranium. Table 2-29 shows the radionuclides of concern for the H-Canyon, B-Line Facility  $^{238}\text{Pu}$  scrap material and for the  $^{235}\text{U}$  scrap material [LaBone 1996, pp. 40–41].

Table 2-29. HB-Line Scrap Recovery Facility.<sup>a</sup>

<b>Area</b>	H
<b>Description</b>	221-H, HB-Line Scrap Recovery Facility
<b>Period</b>	Operations began in 1985 and ended in 2011.
<b>Pu-238 scrap radionuclides of concern</b>	Pu-238, Pu-241, Pu-239
<b>Pu-239 scrap radionuclides of concern</b>	Pu-241, Pu-239
<b>U-235 scrap radionuclides of concern</b>	U-234, U-235, U-236, U-238
<b>Np scrap radionuclide</b>	Np-237

a. Sources: LaBone [1996, pp. 40–41]; Reed et al. [2013, p. 220]; SRNS [2008, p. 3]; DOE [1997, p. 37]; SRNS [2023].

### Neptunium Oxide Facility (Phase II)

Phase II, which is the <sup>237</sup>Np/<sup>239</sup>Pu Oxide Line, can produce powdered oxide material from <sup>237</sup>Np or <sup>239</sup>Pu nitrate solutions. Phase II started operations in November 2001. The plutonium material was shipped to FB-Line for packaging in 3,013 containers for long-term storage and then to K-Area for interim storage. The neptunium material was shipped to the Idaho National Laboratory for further processing and conversion to reactor targets for future <sup>238</sup>Pu production and space exploration. Since startup, approximately 325 kg of neptunium oxide have been shipped. Operations with all neptunium materials were completed as of 2012; Phase II of HB-Line is no longer used with neptunium. The Phase II process equipment was upgraded in 2012 and tested to support plutonium oxide production of feed material for the Mixed Oxide Facility [Groh et al. 2000, p. 11; SRNS 2012].

### Pu-238 Oxide Production Line (Phase III)

Phase III was originally designed to be a replacement for PuFF (Building 235-F), but that use did not materialize. Phase III has been converted into a facility to process open storage containers when necessary and oxidize metals to allow for them to be dissolved in the Phase I process area or the H-Canyon dissolvers. Phase III is specially equipped to support the preparation and repackaging of excess plutonium and uranium metals and oxides in various forms and packaged in various containers and configurations [SRNS 2012].

#### 2.2.3.5 H-Area Outside Facilities

The H-Area Outside Facilities include several processes and services that support the separations function of the 200-H Area. The main processes and services of radiological concern are A-Line, Water Handling Facilities, ARU, GP evaporator, Segregated Solvent Facilities, Transfer Tanks, Sump Collection Tanks, Recycle Sumps, Auxiliary Systems, and Support Facilities.

Note that HP Location Codes alone are not sufficient to isolate a worker to certain individual locations such as A-Line, and without other evidence radionuclides given for H Canyon should be assumed.

Many of the processes and services do not have any connection to each other in terms of function in the Outside Facilities. The processing of radioactive material at the 211-H Outside Facilities are as follows [LaBone 1996, pp. 45–46]. While americium is not listed in Tables 2-30 and 2-31, Farrell and Findley [1999] lists plutonium and americium as part of the H Area Outside Facilities waste stream. Therefore, americium should be assigned to dose reconstructions for workers in H Area Outside Facilities along with <sup>237</sup>Np.

## A-Line

The H-Area A-Line receives a dilute aqueous solution of uranyl nitrate hexahydrate enriched in  $^{235}\text{U}$  from the canyon where it is temporarily stored in A-Line tanks, from which it is pumped into tank trucks for offsite shipment. The radionuclides of concern in the A-Line are shown in Table 2-30.

Table 2-30. H-Area Outside Facilities, A-Line.<sup>a</sup>

<b>Area</b>	H
<b>Description</b>	H-Area Outside Facilities, A-Line
<b>Period</b>	Constructed after 1982. Operations ended 2011.
<b>Radionuclides of concern</b>	U-234, U-235, U-236, U-238

a. Sources: LaBone [1996, p. 46]; DuPont [1984d, p. 456]; Farrell and Findley [1999, p. 11]; SRNS [2023].

## Water Handling System

The water handling facilities are in the Building 211-H Complex between the chemical storage area and the ARU. The primary equipment for water handling consists of tanks, skimmers, and coolers, all of which can contain radioactive materials. The facility is operated to provide process water and recycle water for the canyons, retain recycled water before analysis to permit disposal, decant and discharge skimmed solvent, and store weak acid feed for the ARU. The radionuclides of concern in the water handling facilities are shown in Table 2-31.

Table 2-31. H-Area Outside Facilities, Water Handling System.<sup>a</sup>

<b>Area</b>	H
<b>Description</b>	H-Area Outside Facilities, Water Handling System
<b>Period</b>	Operations began July 1955. Currently operating.
<b>Radionuclides of concern</b>	Nb-95, Zr-95, Ru-106, Cs-137

a. Sources: LaBone [1996, p. 47]; Strack [2002, p. 564]; SRNS [2023].

The Water Handling System includes the following facilities and processes:

- **ARU.** Nitric acid is concentrated in the ARU for reuse. The ARU is in the Building 211-H Complex between the control house and the water handling facilities. All material in this location can contain radioactive material.
- **GP evaporator.** Low-activity aqueous waste is concentrated in the GP evaporator, reducing the volume before sending the waste to the tank farm. The GP evaporator is in the Building 211-H Complex between the control house and the segregated solvent facilities.
- **Segregated solvent facilities.** The final purification and storage of the solvent occurs in the segregated solvent facilities before its return to H-Canyon for reuse. The segregated solvent facilities are in Building 211-H north of the GP evaporator in H-Area. Activity in the spent solvent is caused primarily by degradation of n-paraffin and tributyl phosphate when solvent is exposed to high levels of radiation. The degradation produces complex zirconium and ruthenium, which subsequently contaminate both solvent and product streams. One of the purposes of the segregated solvent facilities is to remove the degradation end products and radioactive contaminants from the solvent.
- **Transfer tanks.** Seven transfer tanks are along the east side of H-Canyon. These tanks provide intermediate pumping stations for solutions transferred from H-Canyon and are in open basins, each containing a sump and a pump. General leaks are pumped to the tanks that feed the GP evaporator, but occasionally the pump discharge must be sent to the seepage basins. The basins are considered a place of potential exposure to airborne radioactivity.

- Sump collection tank. This tank is in an underground vault and receives radioactive waste condensate by steam jet from sumps and tanks related to the canyon air exhaust system and sends it by jet stream to other tanks that feed the hot or warm canyon waste evaporators.
- Recycle sump. The recycle sump is an underground stainless steel lined concrete sump that is near the southwest corner of the original Building 211-H control house. The sump collects the drainage and overflow for all H-Canyon tanks that contain contaminated or recycled liquids. Being underground and shielded, the sump can receive radioactive solutions that are above the limits for unshielded tanks.
- Recycle vent system. The recycle vent system provides a means to filter contaminated air from tanks and vessels before discharge to the atmosphere. The system services tanks in the H-Area A-Line, H-Canyon Facilities, and the first and third levels of H-Canyon. The vent system travels through Building 292-H where it is filtered and sent to the Building 291-H stack.
- Auxiliary systems and support facilities. The process well water system is the only system of concern for these facilities. There are two cooling water streams that are of importance: (1) the segregated cooling water stream, which is discarded, and (2) the recirculating cooling water stream, which is pumped through a cooling tower and reused. Both streams are continuously monitored for radioactivity and can be diverted into retention or seepage basins to prevent a release of radioactive material.

### 2.2.3.6 Receiving Basin for Offsite Fuel and Resin Regeneration Facility

The Receiving Basin for Offsite Fuel (RBOF) and the Resin Regeneration Facility (RRF) have hosted several processes, utilities, and services that support the separations areas. SRS ceased operating RBOF and RRF in 2003 [Reed et al. 2013, pp. 230–231].

#### 2.2.3.6.1 Receiving Basin for Offsite Fuel

The RBOF (244-H) received and stored fuel elements from various offsite reactors and, on occasion, from SRS reactors. After storage, the fuel elements were sent to either of the two SRS separations plants for processing.

The RBOF facility handled casks and fuels of various shapes, sizes, and content. The RBOF equipment and procedures provided for the safe receipt and storage of irradiated nuclear fuels in a system of water-filled basins connected by canals. It included facilities for inspecting the fuel by mechanical and optical means, disassembling fuel, cutting fuel by crosscutting or slitting, and weighing fuel before and after packaging or other alterations. A photo showing irradiated fuel tubes submerged on tracks in water is shown in Figure 2-27.

Casks of spent fuel were received and washed in the cask basin to remove any dirt that might have accumulated during transit. After the cask lid bolts were loosened, the cask was lowered into the cask basin. The lid of the cask was removed and all or part of the loaded fuel basket within the cask could be transferred to a fuel transfer bucket. The transfer bucket could be moved underwater via a canal system to the inspection basin, disassembly basin, repackaging basin, or storage basin [LaBone 1996, p. 48]. Although operations ended in 2003 [WSRC 2008, p. 2], it still has a “significant radionuclide inventory” [Hiergesell et al. 2008, p. 130] and has not been decommissioned and decontaminated as of 2019 [SRNS 2019, p. 19].

Table 2-32 shows the radionuclides of concern for the RBOF.

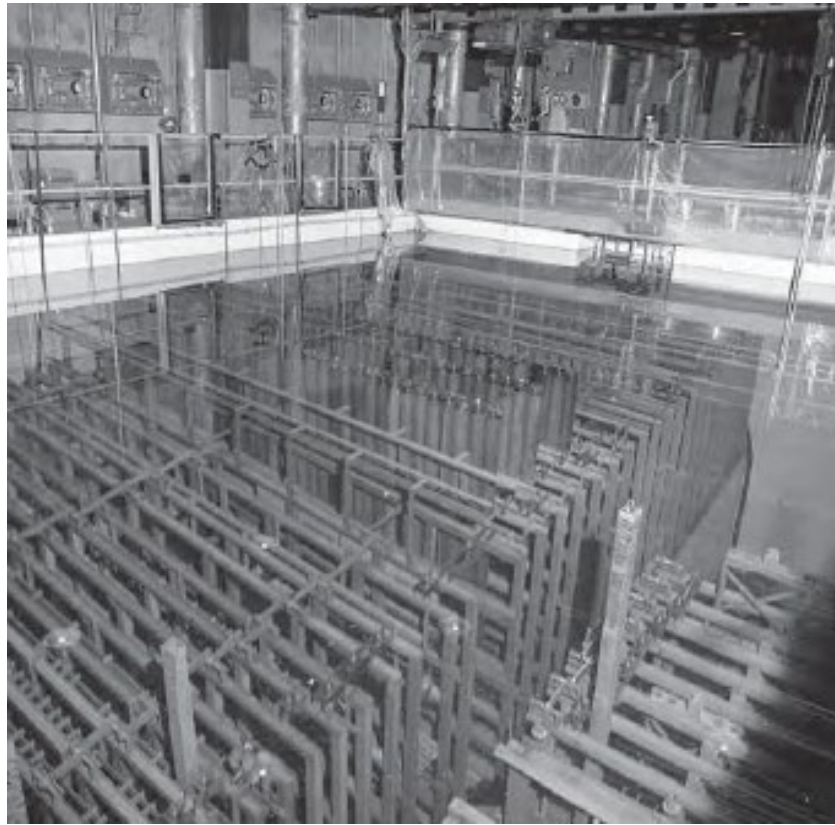


Figure 2-27. Irradiated fuel tubes on racks underwater in 244-H, June 26, 1968 [Reed et al. 2013, p. 231].

Table 2-32. RBOF, irradiated fuel receipt and storage.<sup>a</sup>

<b>Area</b>	H
<b>Description</b>	244-H, RBOF
<b>Period</b>	Operations from 1964 to present.
<b>Radionuclides of concern</b>	Kr-85, I-131, Xe-133, Pu <sup>b</sup> , Am <sup>b</sup>

a. Sources: LaBone [1996, p. 48]; Farrell and Findley [1999, p. 12]; WSRC [2008, p. 2]; Strack [2002, pp. 534, 569].

b. Although the 1996 Facility Descriptions document indicated Kr-85, I-131, and Xe-133 had the greatest activity at RBOF, the 1999 site characterization study indicated bioassay for Pu and Am to monitor RBOF exposures.

The irradiated fuel received at RBOF contained uranium and fission and activation products. Because the work was performed underwater, the inert gases that escaped the fuel bundles are of primary radiological concern.

In addition to irradiated fuel, RBOF performed other operations such as basin purification, resin regeneration, and target cleaning. Tritium, fission and activation products, and plutonium could be present during these activities [LaBone 1996, p. 48].

#### **2.2.3.6.2 Resin Regeneration Facility**

The RRF was capable of regenerating both anion and cation type resins. Normally, it was utilized to regenerate the resin for deionization of Building 244-H water and the resin used in the potable deionizer to deionize the 100-Area water systems. However, it could also regenerate any other resin that could be brought to the facility. Other operations performed at RRF included the removal and disposal of the filter cake from the 100-Area filter and chemical cleaning of target slugs before processing in Tritium Facilities [Shedrow 1997, pp. 13–14].



The RRF, in Building 245-H, received a mixture of anion and cation resins from the basin water deionizer. Resins were transferred from the deionizer to tanks in the RRF. The mixed resin was first depleted with NaOH and then the cation and anion resins were separated, after which they were regenerated, rinsed, and returned to the deionizer. The RRF regenerated either mixed or unmixed resins shipped from the reactor facilities.

Target elements were also cleaned at RRF. Irradiated aluminum-clad target elements of LiAl alloys were transported by truck to the RRF in a cask that was connected to a hose box for in-cask cleaning to remove hydrated alumina from the target elements [LaBone 1996, p. 48].

Table 2-33 shows the radionuclides of concern at the RRF.

Table 2-33. RRF, spent ion exchange resin regeneration.<sup>a</sup>

<b>Area</b>	H
<b>Description</b>	245-H, RRF
<b>Period</b>	Operations began in 1969 and ended July 30, 2010.
<b>Radionuclides of concern</b>	Co-60, Cs-137, Cs-134

a. Sources: LaBone [1996, p. 49]; SRNS [2011, p. 5]; SRNS [2023].

## 2.2.4 **F- and H-Areas Cooling Water and Retention Basins**

The cooling water and retention basins are support facilities for the separations and waste management areas. Low-level radioactive liquid wastes from the separations areas are routed to the basins when activity is detected [LaBone 1996, p. 26].

### 2.2.4.1 **Cooling Water Basins**

When radioactivity was encountered in a cooling water system, immediate action was taken to divert the water to a retention basin and isolate the leak. When flushing had reduced the level of radioactivity to below the guideline for release to the separations waste basins, diversion to the retention basin was terminated [LaBone 1996, p. 27].

### 2.2.4.2 **Retention Basins**

Both F- and H-Areas use a retention basin when necessary for temporary storage of storm drainage that might be contaminated. The retention basin also collects the water from the overflow from the cooling water basin. Stormwater is automatically diverted to the retention basin when radiation monitors detect contamination levels that are greater than those specified in process operating limits. After sampling and analysis, the water in the retention basin might be further processed or it might be transferred to the ETF or Four Mile Creek. The liquid waste sent to the cooling water and retention basins comes from several different facilities and processes such as cesium removal columns and overhead condensate. Table 2-34 shows the radionuclides of concern for the cooling water and retention basins [LaBone 1996, p. 27].

Table 2-34. F- and H-Areas cooling water and retention basins.<sup>a</sup>

<b>Area</b>	F and H
<b>Description</b>	281, F/H cooling water and retention basins
<b>Period</b>	Constructed in 1951.
<b>F-Area retention basin radionuclides of concern</b>	Cs-137, Sr-90, Pu-239, U-238, Ru-106, Ce-144, Pu-238
<b>H-Area retention basin radionuclides of concern</b>	Cs-137, Sr-90, Ru-106, Ce-144, Pu-238, Pu-239, U-238

a. Sources: LaBone [1996, pp. 27–28]; Strack [2002, p. 380].

### 2.2.5 F- and H-Area Tank Farms

High-level liquid radioactive wastes from SRS were received and managed in large underground tanks in the Tank Farms in 241-F and 241-H. The two farms contained 51 large subsurface tanks and related facilities required for safe handling, processing, and temporary retention of liquid wastes. The liquid waste came primarily from fuel reprocessing operations in F- and H-Areas with smaller quantities from the RBOF, the RRF, SRNL, and the 100-Area. Liquid wastes were classified as either high-heat waste (HHW) or low-heat waste (LHW) depending on whether the concentration of radioactivity was such that forced cooling was required to maintain waste temperature within operating guidelines [LaBone 1996, p. 28]. The F-Area Tank Farm is defined as all the buildings and waste tanks in the 221-F Area. The H-Area Tank Farm is defined as all the buildings and waste tanks in the H-Canyon and 242-H Area. A drawing of the F-Area Tank Farm is given in Figure 2-28. Drawings of the H-Area Tank Farm (west) and the H-Area Tank Farm (east) are given in Figures 2-29 and 2-30, respectively.

High-level waste tanks were continuously constructed from the 1950s through the 1980s. The designs of the tanks changed over that time frame in a general evolution from a single-walled tank to a double-walled tank. Tank designs were labeled as Type I, Type II, Type III, and Type IV [Reed et al. 2013, pp. 245–248].

The first liquid waste storage tanks at Savannah River became known as Type I tanks. One of the most critical aspects of construction was the care that had to be taken in making and inspecting the welds of the plates that comprised the tanks. The Pittsburgh Testing Laboratory spent nearly a year and a half X-raying welds to assess quality. From February 1952 to August 1953, approximately 50,000 ft of welded seams were checked by the laboratory. Type II liquid waste storage tanks were constructed in H-Area. The modified design was prompted by experience from construction of the first tanks, by new data about waste behavior from Hanford, and from research at the Savannah River Laboratory. The Type IV tanks were more economical than Types I and II due to simpler design and building cost. They were used to store wastes that generated less heat [Reed et al. 2013, p. 246].

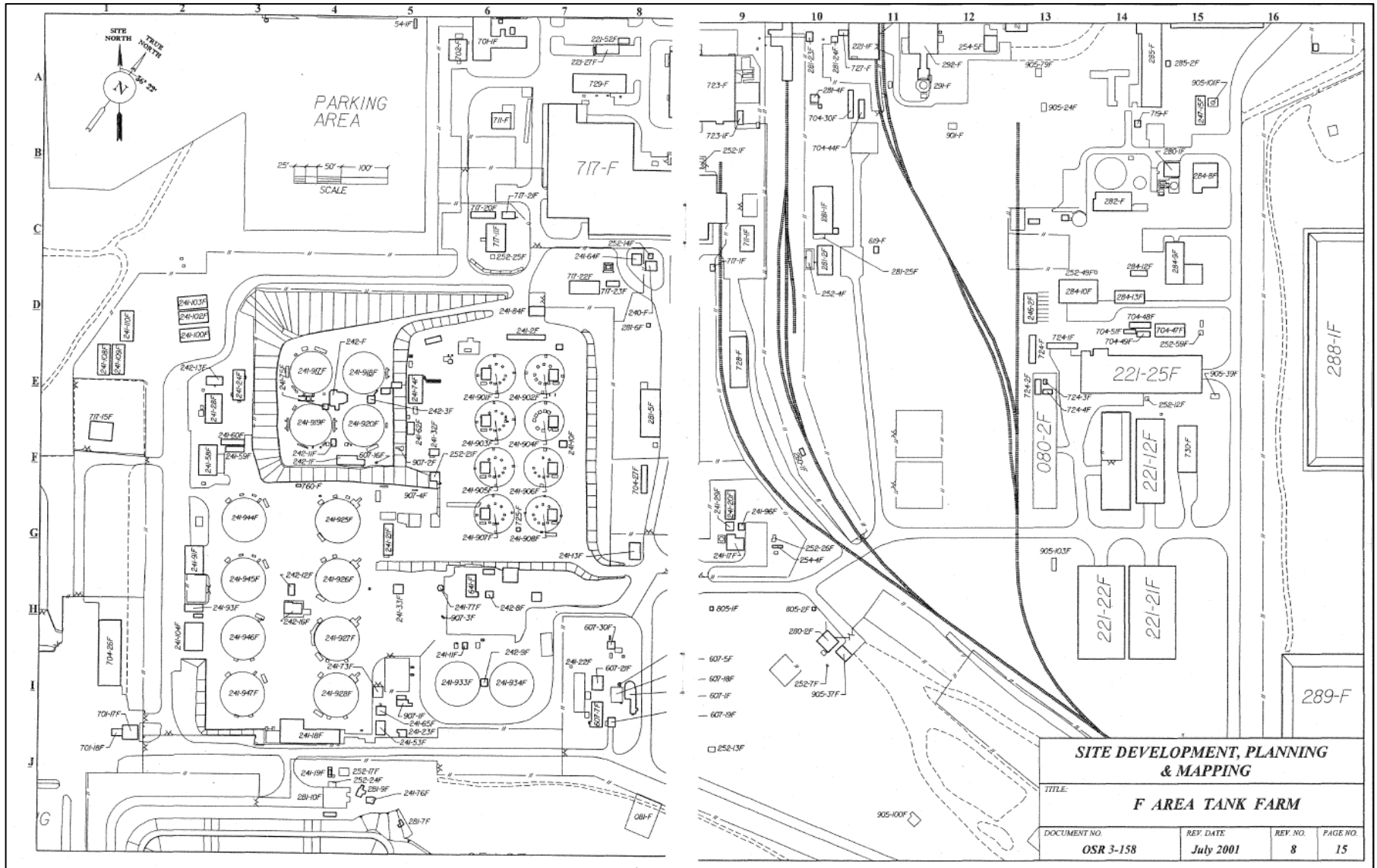


Figure 2-28. F-Area Tank Farm [WSRC 2001a, p. 19].

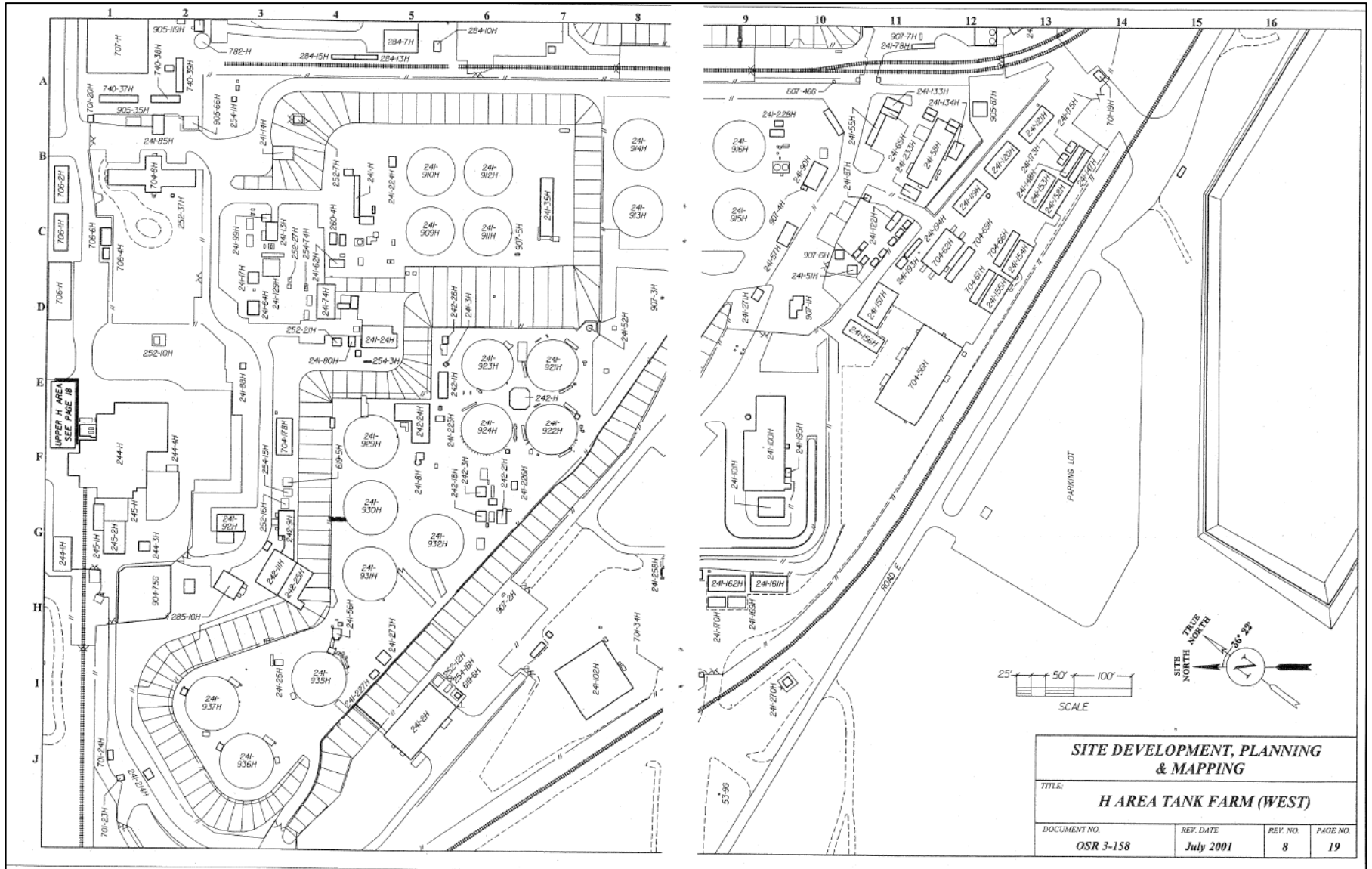


Figure 2-29. H-Area Tank Farm (west) [WSRC 2001a, p. 23].

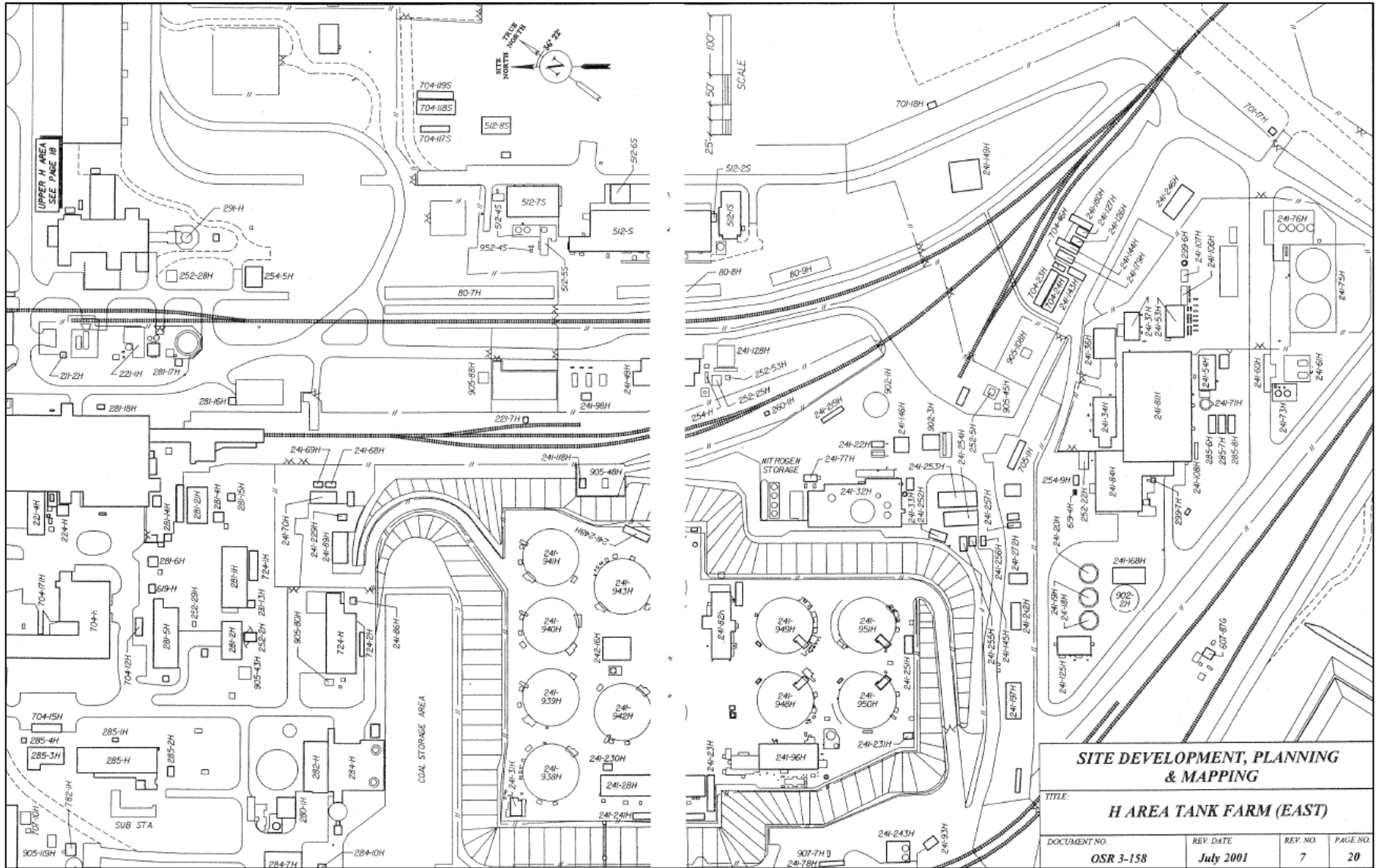


Figure 2-30. H-Area Tank Farm (east) [WSRC 2001a, p. 24].

### 2.2.5.1 F- and H-Area Waste Reprocessing

Liquid waste in Buildings 221-F and H-Canyon is made up of many waste streams generated during the recovery and purification of the reprocessing operation (i.e., the HHW from 221-F and H-Areas and the LHW from F- and H-Areas). The HHW from 221-F is transferred to pump Tank 3 and then to receipt Tank 33 or 34. The F-Area LHW is transferred to pump Tank 2 and to evaporator feed Tank 26. In H-Area, the HHW from H-Canyon is transferred to pump Tank 5 and then to receipt Tank 39. The H-Area LHW is transferred to pump Tank 6 to evaporator feed Tank 43. The evolution of tank design is displayed in Figure 2-31. Type I was used in both F-Area (eight tanks) and H-Area (four tanks) from 1951 to 1953. Type II was used in H-Area (four tanks) only in 1955. Type III was installed in both areas from 1967 to 1981 with 10 tanks in F-Area and 17 tanks in H-Area. Type IV was constructed in F-Area in 1958 and in H-Area in 1962 with four tanks in each area.

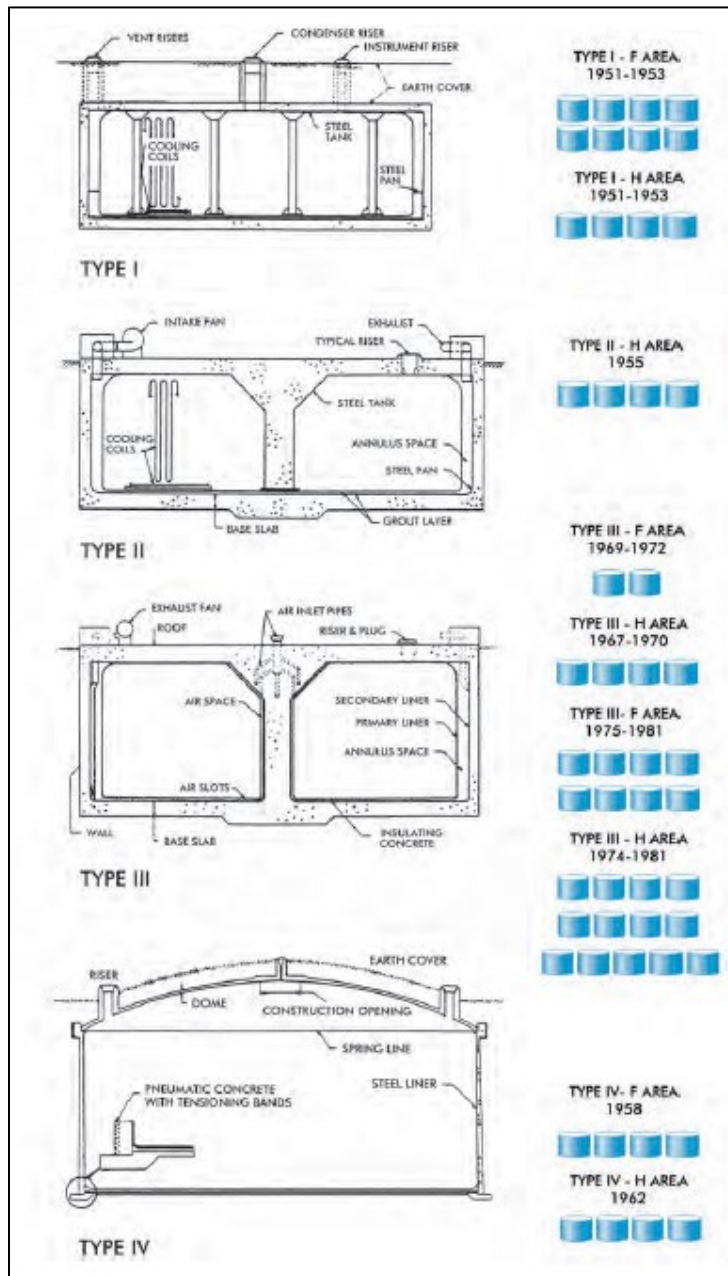


Figure 2-31. Evolution of HHW tank design [Reed et al. 2013, p. 246].

### 2.2.5.2 Waste from Receiving Basin for Offsite Fuel and Resin Regeneration Facility

Waste from RBOF (244-H) and RRF (245-H) was produced during regeneration of ion-exchange beds, backwashing filters, cleaning and handling of fuel and target elements, and similar incidental operations. Radioactivity in these liquids was approximately 400  $\mu\text{Ci/gal}$ , and the volume ranged from 1.0 to 2.3 million gal/yr. Liquid waste from RBOF and RRF operations in H-Area was pumped to Tank 21 or hold Tank 23. Waste liquid that was too high in radioactivity for adequate decontamination by the cesium removal column was routed to Tank 21 for subsequent evaporation.

### 2.2.5.3 Waste from Building 773-A and 100-Area

Batches of liquid waste from Building 773-A (SRNL) and the reactor areas that were not suitable for discharge to SRS seepage basins were delivered in tank trailers to the 211-F waste complex where the waste was evaporated either in the laboratory waste evaporator in 221-F (HHW) or in the 211-F general evaporator (LHW). The waste was adjusted for alkalinity if necessary and transferred to the 241-F waste storage tanks as LHW. Aqueous waste from the reactor area was handled much the same as laboratory waste unless it contained insoluble materials such as sludges or slurries. When solids were present, the waste was unloaded from the reactor area trailer directly to Tank 47 [LaBone 1996, p. 29].

- Waste processing operations. Liquid waste from separations areas and other locations were routinely processed through a series of operations allowing segregation and consolidation of the waste components before interim storage.
- Waste evaporation. Radioactive waste received in the waste tank facilities was reduced by evaporation, and the concentrated solutions were immobilized by solidification of the residual salts. The evaporator condensate was continuously monitored and, if the radioactivity exceeded 1,500 dpm/mL gamma, the condensate was diverted and recycled.
- Radionuclide content of waste. The radionuclide contents depended on the age of the waste and varied with the source stream and previous processing.
- Air monitoring systems. The air monitoring systems at F- and H-Area Tank Farms were designed to measure radioactive releases from the tanks to the atmosphere. The system was not designed to measure the breathing air in the workplace. The monitoring was accomplished by passing a fraction of the air through a filter paper that is monitored by a Geiger-Müller detector with the signals amplified and read out in the tank farm control room. If no abnormal conditions were detected, the filter paper was processed through the Health Physics counting room to measure and maintain records of the low-level radioactive releases to the atmosphere.

Table 2-35 lists the radionuclides of concern for the F- and H-Area Tank Farms.

Table 2-35. F- and H-Area Tank Farms.<sup>a</sup>

<b>Area</b>	F and H
<b>Description</b>	241-F/241-H, F/H Tank Farms
<b>Period</b>	Operations began in 1954. Removal and closure to continue through the 2030s. <sup>b</sup>
<b>F-Area combined waste tank radionuclides of concern</b>	Cs-137, Sr-90, Ce-144, Ru-106, Pu-241, Pu-239, Pu-238, Am-241
<b>H-Area combined waste tank radionuclides of concern</b>	Sr-90, Cs-137, Ce-144, Ru-106, Pu-241, Pu-238, Am-241

a. Sources: LaBone [1996, pp. 30–31]; Strack [2002, p. 564] unless otherwise noted.

b. Source: Savannah River Remediation (SRR) [2015, p. 9].

### 2.2.6 E-Area Solid Waste Disposal Facility

Solid and liquid low-level waste generated at SRS and some wastes from off site are stored at the Solid Waste Disposal Facility (SWDF). The SWDF is in areas labeled 643-E and 643-7E. The old burial ground is 643-E and ceased operations in 1972. Burial operations were phased in during 1969 to 1972 in the present SWDF, 643-7E. Solid waste is packaged in 55-gal drums, carbon steel engineered boxes, shipping containers, or not at all depending on the radiation level and potential for reuse (such as large equipment). Solid wastes have been placed both above and below ground. Most of the solid waste above ground is TRU waste, mainly <sup>238</sup>Pu and <sup>239</sup>Pu contaminated waste. Contaminated process equipment in steel boxes is stored above ground until reuse. The solid non-TRU waste below ground is contained in carbon steel boxes, drums, or concrete, or is uncontained depending on its type and form.

Solid radioactive waste is classified as low-level alpha and beta/gamma waste, intermediate-level beta/gamma waste, low-level alpha waste, intermediate-activity TRU waste, high-activity TRU waste, and temporarily stored contaminated equipment [LaBone 1996, p. 23].

Due to the diverse nature of the radioactive waste, the radionuclide content of each type of waste has not been defined. The radionuclides of highest concentration are tritium and isotopes of cobalt, plutonium, and curium (Table 2-36).

Table 2-36. E-Area SWDF, liquid and solid waste disposal.<sup>a</sup>

<b>Area</b>	E
<b>Description</b>	643-E Old Radioactive Waste Burial Ground, 643-7E Radioactive Waste Burial Ground/SWDF
<b>Period</b>	643-E operations from 1953 to 1974; 643-7E operations began in 1969. Currently operating.
<b>Radionuclides of concern</b>	Various; the radionuclides of highest concentration were H-3 and isotopes of Co, Pu, and Cm.

a. Sources: LaBone [1996, pp. 23–24]; WSRC [2000b, pp. 821, 884]; SRNS [2023].

### 2.2.7 724-8E Waste Certification Facility

The Waste Certification Facility (WCF), Building 724-8E, was used to certify and package drums of TRU waste. The functions of the WCF were to assay and certify the contents of TRU waste drums and to package, load, and ship the drums.

Only physical handling of sealed drums was performed in the WCF. Drums were not opened, and the contents were not exposed in normal operation. Each drum was inspected for proper packaging. Health Physics routinely surveyed the WCF to ensure the area was not contaminated [LaBone 1996].



Examples of typical generic contents of the TRU waste drums included:

- Combustibles. Paper, gloves, sweepings, and cloth rags.
- Noncombustibles. High-efficiency particulate air (HEPA) filters, crucibles, and glassware.
- Chemicals. Evaporated sludge, caustic soda, soda lime, and spent resins.
- Equipment. Gloveboxes, motors, and scales.

Normally the drums contained either  $^{238}\text{Pu}$  or  $^{239}\text{Pu}$  depending on the source of the TRU waste [LaBone 1996]. Table 2-37 lists the radionuclides of concern for WCF.

Table 2-37. WCF Building 724-8E, certification and packaging of TRU waste.<sup>a</sup>

<b>Area</b>	E
<b>Description</b>	724-8E, WCF
<b>Period</b>	Building construction began in 1986; operations began in December 1986 decontamination and decommissioning completed March 2005.
<b>Radionuclides of concern</b>	H-3, Cm, Pu, fission products, activation products

a. Sources: LaBone [1996, p. 36]; SRNS [2017, p. 6]; DuPont [1988a, pp. 149–150].

### 2.2.8 Tritium Facilities

The initial Tritium Facility, Building 232-F, at SRS was built as an interim facility in 1953 to 1954 and began tritium extraction operations in October 1955 after the reactor and separations startups. During this period the demands for plutonium and tritium increased significantly. The 232-F Facility had a low capacity throughput of only one slug per furnace per cycle. SRS shut down Building 232-F in October 1958 [Blackburn 1994].

The Tritium Facilities include the Old Tritium Extraction Facility, H-Area New Manufacturing (HANM), H-Area Old Manufacturing (HAOM), and Reclamation Facility.

The five tritium processing buildings house the following unique operations and capabilities:

- Extraction and recycling (Old Tritium Extraction Facility),
- Loading, unloading, and recycling (HANM),
- Reservoir handling (HAOM),
- Helium recovery and reservoir burst testing (HAOM), and
- Reservoir reclamation (Reclamation Facility).

The Old Tritium Extraction Facility is the oldest of the five buildings and began operation in 1955. The facility is the location of the equipment that performs the functions of vacuum furnace extraction of tritium from the irradiated LiAl targets, purification, enrichment, and loading and unloading low-pressure product containers.

HANM is the newest and most technologically advanced of the five buildings and began operation in 1994. It houses the equipment to perform the functions of loading and unloading reservoirs, purification and enrichment of tritium, and initial separation of the tritium from the gas stream.

HAOM houses equipment that performs the functions of receiving new manufactured reservoirs and inspecting, testing, packaging, and shipping loaded reservoirs. Returned reservoirs are also accounted for in this building. HAOM and the Reclamation Facility provide a support mission to the three primary structures. HAOM began operation in 1964 and provides tritium recovery and reservoir pneumatic burst test capability. The Reclamation Facility provides reservoir reclamation capability [Buley et al. 1995].

Assume elemental tritium and tritiated water to be the primary radionuclides of concern at SRS tritium facilities. The activation product  $^{65}\text{Zn}$  was associated with the processing of lithium targets in Building 232-H [LaBone 1996].

### **2.2.8.1 Tritium Processing**

Tritium gas processing consists of five main areas: separation, purification, compression, pumping, and storage.

Separation is the process of removing unwanted gas impurities from the hydrogen. This operation is accomplished using a commercially available palladium diffuser.

The purification process involves taking a hydrogen stream and dividing the hydrogen isotopes (i.e., tritium, deuterium, and protium). This is performed by two methods: cryogenic distillation and the thermal cycle absorption process. Cryogenic distillation uses the difference in boiling points to facilitate the separation of isotopes. The thermal cycle absorption process uses thermal cycling of a palladium hydride to perform the gas separation.

The compression and pumping processes are mechanical operations using diaphragm compressors to produce numerous atmospheres of pressure for filling reservoir containers. Positive displacement bellows pumps, in series with orbiting scroll pumps, produce from full vacuum to the 2- to 3-atmosphere positive pressure needed for gas movement.

Gas movement is also provided by metal hydride beds. These beds serve primarily as tritium storage but also provide energy to evacuate and move low-pressure tritium gas. The storage capability provides a safer operation to contain tritium because it converts tritium gas into a solid form [Buley et al. 1995].

### **2.2.8.2 Special Tritium Compounds**

SRS determines if a compound is classified as a special tritium compound based on the degree to which the tritium atoms are held in the compound. Tritium atoms in special tritium compounds are bound more tightly than the tritium in tritium oxide. Some metal hydrides lack the potential to form as a special tritium compound. These include zeolite, magnesium, iron, and steel hydrides.

#### **2.2.8.2.1 Metal Tritides**

Compounds can be created by intentional combination of tritium with the desired materials or by inadvertent contamination of a material that has been subjected to the presence of tritium over time. Tritium combines with certain metals with the relative ease of tritium exchange. The property of interest of these materials is that they react with hydrogen isotopes to form stoichiometric hydrides in which the hydrogen isotope is contained in interstitial sites in the metal lattice [Nobile and Motyka 1991]. Metal tritides are essentially tritium in a particulate form, which can present dose reconstruction problems that are distinctly different from those of the more commonly encountered tritiated water vapor. The hydriding/dehydriding reaction of various metals can be used as a basis for the storage, pumping, compression, purification, and separation of hydrogen isotopes. One advantage of using

hydride technology in tritium processes is the compact size of hydride equipment because the volumetric density of hydrogen in typical hydrides is greater even than that of liquid or solid hydrogen [ORAUT 2017].

Metal tritides are classified as either stable or unstable depending on the ease with which they release tritium. Unstable metal tritides pose no problems because they dissociate themselves with ease in the environment and biologically form tritiated water vapor. Stable metal tritides, however, maintain their original form with a stable metal-tritium bond [ORAUT 2017].

The SRS Tritium Facilities have used a variety of metal hydride materials in process vessels or beds for the safe processing and storage of tritium [ORAUT 2017]. These materials are used in tritium processes as storage media and are the subject or byproduct of research activities [Moxley 2002]. Due to the sensitive nature of the work at SRS, information such as the quantities of material and the details of the work on metal hydrides and tritides are still classified [ORAUT 2017]. Insoluble stable metal tritides encountered at SRS are shown in Table 2-38.

Table 2-38. Stable metal tritides.<sup>a</sup>

Stable metal tritides	Absorption type
Lanthanum-nickel-aluminum	F
Palladium	F
Palladium-aluminum	M
Palladium-silver	F
Titanium	M
Uranium	F
Zirconium	M
Zirconium-iron-chromium	M

a. Source: ORAUT [2017, p. 26].

More detail on the production and use of tritium and hydride materials is provided in ORAUT-RPRT-0072, *Locations of Stable Tritide Use at the Savannah River Site* [ORAUT 2017].

#### 2.2.8.2.2 Organically Bound Tritium

Exposure to organic forms of tritium (commonly called organically bound tritium) in the same tritium facilities was possible. Milham and Boni [1976] described measurements of tritium species in effluents from the tritium processing areas: "The concentration of tritium in the organic form is generally 1% of the tritium released to the atmosphere." Howard [2000] mentions mercury diffusion pumps, tritiated vacuum pumps, and tritiated polymers and elastomers as possible sources of exposure to organically bound tritium.

#### 2.2.9 300-M Fuel and Target Fabrication Area

The 300-M Area was near the northwest perimeter of the SRS complex and contained testing and fabrication facilities for reactor fuel and targets. Two buildings, 305-M and 777-M (later renumbered as 777-10A), contained test reactors that were used to test the components manufactured in the 300 Area and to aid development and testing for SRS reactor design [Reed and Swanson 2006, p. 37].

Early reactor fuel elements, or slugs, were fabricated by the Feed Materials Production Center in Fernald, Ohio, and shipped to Savannah River in the 300-M Area. The slugs were solid natural uranium rods about 1 in. in diameter and 8 in. long, clad in aluminum. After the metallurgical structure of the uranium rods was adjusted (first at SRS and later at Fernald before shipment), the slugs were sealed in aluminum [Reed and Swanson 2006, p. 42].

Lithium target slugs were needed for the production of tritium and for use as control rods in the reactors. Lithium was sent from the Oak Ridge site to Building 320-M where it was alloyed with aluminum, cast into billets, extruded to the proper diameter, cut to the required length, and canned in aluminum. The LiAl slugs were also encased in aluminum sheaths called "raincoats." At SRS, tritium was initially produced as a reactor byproduct in the LiAl control rods. As AEC requirements for tritium increased, reactor elements specifically designed for tritium production were needed. Driver, or fuel, elements of highly enriched uranium were used to provide the neutrons for irradiating the LiAl target elements. Enriched uranium drivers were extruded in 320-M until 1957, after which they were produced in the newly constructed 321-M, built specifically for this process [Reed and Swanson 2006, p. 42].

Target assemblies containing plutonium were manufactured for the Curium I program in 1964, the Curium II program in 1965 to 1967, and the Californium I program in 1969 and 1970 [Reed and Swanson 2006, p. 51]. The isotopic mix for these assemblies is not known. Targets containing plutonium-aluminum alloy were produced from 1963 to 1965, 1968 to 1970, and in 1975 [DuPont 1974, pp. 116, 124, 135, 167, 176, 186; Reed and Swanson 2006, p. 188]. Targets containing PuO<sub>2</sub>-Al alloy were produced in 1975 [Reed and Swanson 2006, p. 189].

The M-Area continued to produce most of its own fuel and target assemblies until the end of the Cold War. Revisions and upgrades were made to the facilities as needed. One of the most important was the change from solid slugs to tubular elements. The production of solid slugs ended late in 1957. Production in the M-Area varied with the needs of the reactors. The last large increase was in 1983 when the operations in 321-M went to 24 hours a day. Operations fell off as the reactors closed, and for the most part have ceased altogether since 1989 when the last reactor was taken off line [Reed and Swanson 2006, p. 42].

Production years for products manufactured in the 300-M Area are shown in Figure 2-32. These products can be divided into four general categories:

- Natural and slightly enriched uranium fuel slugs, hollow slugs, and short tubes including Mark 1 Slugs (Canned), Mark VII Slugs (Canned), Mark VII-A Slugs (Canned), and Mark V Series Short Tubes;
- Enriched uranium fuel tubes and slugs including Extended Surface Fuel Tubes and Mark VIII (fuel slugs);
- Target assemblies including Blankets, Depleted Uranium, Cobalt, Thorium, and other target types; and
- Control and safety rods including LiAl Cast, Control Rods, and Shadow Slugs.

Year	Mark 1 Slugs (Canned)	Mark VII Slugs (Canned)	Mark VII-A Slugs (Canned)	Mark V Series Short Tubes	Extended Surface Fuel Tubes	Mark VIII (fuel slugs)	Blankets	Depleted Uranium	Cobalt	Thorium	Other	LiAl Cast	Control Rods	Shadow Slugs
1957	Red	Orange	Yellow		Green	Light Green						Blue	Light Blue	
1958	Red	Orange	Yellow		Green							Blue	Light Blue	
1959			Yellow		Green		Purple					Blue	Light Blue	
1960			Yellow	Yellow	Green		Purple					Blue	Light Blue	Light Cyan
1961			Yellow	Yellow	Green		Purple		Light Purple	Light Purple		Blue	Light Blue	Light Cyan
1962			Yellow	Yellow	Green				Light Purple	Light Purple		Blue	Light Blue	
1963				Yellow	Green				Light Purple	Light Purple		Blue	Light Blue	Light Cyan
1964				Yellow	Green				Light Purple	Light Purple	Light Purple	Blue	Light Blue	
1965				Yellow	Green				Light Purple	Light Purple	Light Purple	Blue	Light Blue	
1966				Yellow	Green				Light Purple	Light Purple	Light Purple	Blue	Light Blue	
1967				Yellow	Green			Purple		Light Purple	Light Purple	Blue	Light Blue	Light Cyan
1968				Yellow	Green		Purple	Purple		Light Purple	Light Purple	Blue	Light Blue	Light Cyan
1969				Yellow	Green			Purple	Light Purple		Light Purple	Blue	Light Blue	Light Cyan
1970					Green			Purple			Light Purple	Blue	Light Blue	Light Cyan
1971					Green			Purple			Light Purple	Blue	Light Blue	Light Cyan
1972					Green			Purple			Light Purple	Blue	Light Blue	Light Cyan
1973					Green			Purple			Light Purple	Blue	Light Blue	Light Cyan
1974					Green			Purple			Light Purple	Blue	Light Blue	Light Cyan
1975					Green			Purple			Light Purple	Blue	Light Blue	Light Cyan

Figure 2-32. 300-M Area Production statistics, 1957 to 1975 [Reed and Swanson 2006, p. 183].

Figure 2-33 shows a diagram of the 300-M Area dated January 1956. The diagram shows the primary buildings where radiological work was conducted, 305-M, 313-M, 320-M, 777-M, and 321-M, which was constructed in 1957 along with 322-M. Figure 2-34 shows an aerial photo of the 300-M Area, date unknown, with all major buildings included.



### 2.2.9.1 305-M Test Pile

Building 305-M was built to test bare slugs, canned slugs from 313-M, and control rods from 320-M to assess their quality for use in the production reactors. There was one primary piece of equipment in 305-M, an uncooled natural uranium test pile with a graphite moderator. The test pile was the centerpiece of the building; all openings and ancillary components were arranged to maximize performance. The test pile was based on a design like one at Hanford [Reed and Swanson 2006, p. 91]. Neutron absorbing rods controlled the pile. Gravity-driven safety rods were inserted into the top and the west face of the pile for use in case of emergencies. The graphite moderator was built of 47 layers of blocks. The pile contained 360 channels for uranium rods in an 8.375-in. lattice, although no uranium channels were provided in the outer four blocks around the four sides of the pile. The central core, bounded by the outermost uranium channels, was made of the highest grade graphite with the remainder of the 190 t in the pile consisting of unpurified graphite [Reed and Swanson 2006, p. 133].

From Reed and Swanson [2006, p. 136]:

*The actual testing process began with the shipment of components from 313-M and 320-M by covered truck via an air lock. The air lock was fitted with a monorail crane to remove the cover when foils and rods were shipped in boxes then stored in a Cold Assembly Storage area. Prior to insertion, they were placed in troughs for loading by the charging machines. An installed crane and other handling equipment facilitated the handling. Slugs and cans were shipped on trays and covered pallets that were unloaded by forklift that would also bring them onto the charging platform. Slug trays were loaded manually and tongs were used to transfer them to the troughs that led to the reactor.*

*Testing procedure changed given what was to be tested:*

- Bare uranium slugs would be placed ten to a hole in two test holes located in center of reactor. The charging action alternates from the north to the south faces of the reactor until completed.*
- Canned Uranium slugs followed the same testing procedure as above.*
- Canned LiAl slugs were charged into three fringe test holes. The number tested would depend on the reactivity of the pile as well as the alloy used.*
- Control Rods were tested in the same holes as the LiAl slugs. The test aimed at detecting abnormalities in the amount of lithium in any one slug or to identify if there were an excessive number of problem slugs.*
- Quatrefoils, Septifoils, [used in control rods, each foil having a combination of cadmium rods and producer rods of LiAl alloy] and slug cans were tested to ascertain the quality of the metal in their fabrication. A hole in the center of the reactor was used for these.*

The radionuclides of concern are a uranium mixture, thorium and its progeny, and possible activation products [Reed and Swanson 2006, pp. 133–137; DuPont 1961–1962, p. 16]. SRS operated the test pile from 1952 through 1981 and dismantling was performed in 1984. SRS operated the nuclear test gauge (NTG) from 1954 through 1981 [Reed and Swanson 2006, pp. 136, 196].

### 2.2.9.2 313-M Uranium Target Fabrication Facility

The Uranium Target Fabrication Facility in Building 313-M was designed and built to manufacture aluminum-clad targets for irradiation in the SRS reactors. SRS fabricated depleted uranium metal from the Feed Materials Production Center into target rods that were used for transmutation of  $^{238}\text{U}$  into plutonium. After 1961, the target assemblies contained recycled uranium. The recycled uranium contained trace amounts of fission and activation products. There were no machining operations on the uranium metal, so there was little contamination associated with the process [WSRC 2000a]. Thorium slugs were fabricated in Building 313-M. The first thorium canning campaign took place from January 1953 to March 1953; about 320 1-kg slugs were canned [DuPont 1953, p. 18]. About 1,785 slugs were canned in 1954, and 100 were canned in the first half of 1955. In September 1956, SRS received a request to irradiate about 5,000 slugs of thorium to produce  $^{233}\text{U}$  [NIOSH 2010b]. The plan was to produce material that would be shipped to ORNL in 0.75-t batches with minimal cooling time (less than 30 days after discharge) to enable testing of radiation damage to extraction solvents and to determine radiation levels during various stages of the extraction process [DuPont 1957b]. Sylcor canned these thorium slugs [DuPont 1956d, p. 133]. A total of 5,200 canned slugs were to be faced, welded, bond-tested, and autoclave-tested at SRS [DuPont 1957d, p. 198]. This fuel was planned for inclusion in reactor loads [DuPont 1957b,d]. By July 1957, SRS had shipped eight shipping casks containing irradiated thorium fuel to ORNL [DuPont 1957e]. From 1961 to 1963 SRS authorized Sylcor and Bridgeport Brass Company to extrude and can thorium metal slugs off site [DuPont 1961b, p. 16; DuPont 1974, p. 109]. In 1964, SRS installed vibratory compaction equipment in 313-M to produce thorium oxide (thoria) slugs; these were produced in 313-M through 1965 [NIOSH 2011]. See Table 2-39.

Table 2-39. 313-M Uranium Target Fabrication Facility.<sup>a</sup>

<b>Area</b>	M
<b>Description</b>	313-M, Uranium Target Fabrication Facility
<b>Period</b>	Operations from 1952 to 1988
<b>Radionuclides of concern</b>	U-238, U-234, U-235, Th-232, Ra-228, and Th-228

a. Sources: LaBone [1996, p. 51]; NIOSH [2011]; NIOSH [2010b]; Reed and Swanson [2006, p. 82]; Strack [2002, p. 519].

### 2.2.9.3 320-M, 322-M, and 341-M

The Alloy Building, 320-M, housed manufacturing process facilities for producing reactor control and circumferential blanket rods containing a LiAl alloy and reactor rods containing 35W5 slugs for the reactors in the 100-Area [DuPont 1957f, p. 118]. Enriched uranium fuel elements were extruded in 320-M until 1957 when production moved to 321-M [Reed and Swanson 2006, p. 42].

The Metallurgical Laboratory, 322-M, was created for process control and method development [Reed and Swanson 2006, p. 75]. In 1962, it was expanded to house facilities for the examination of contaminated samples for equipment failures [DuPont 1970a, p. 257].

The Dilute ETF, 341-M, processed all M-Area process rinse waters [WSRC 1991a, p. 58].

All radionuclides in the tables are of concern for both internal and external exposures. SRS also extruded thorium fuel components in 320-M from 1954 to 1968 [NIOSH 2011]. SRS ceased production activities at all 300-M Area buildings in 1987. SRS removed all equipment by 1997, and demolition of buildings started in 2003 [SRNS 2017, p. 9]. See Table 2-40.



Table 2-40. 320-M, 322-M, and 341-M, uranium processing facilities.<sup>a</sup>

<b>Area</b>	M
<b>Description</b>	320-M Alloy Building, 322-M Metallurgical Laboratory, and 341-M Waste Water Treatment Facility
<b>Period</b>	320-M operations began August 15, 1952; decontamination and decommissioning completed in March 2004. 322-M operations began January 1, 1956; decontamination and decommissioning completed in January 2004. 341 M was constructed beginning June 1, 1985; decontamination and decommissioning completed in March 2004.
<b>Radionuclides of concern</b>	Depleted uranium, enriched uranium, Pu (1963–1965, 1968–1970, 1975, 1979–1980)

a. Sources: LaBone [1996, p. 52]; Reed and Swanson [2006, p. 58]; SRNS [2017, p. 9].

SRS installed an NTG in 320-M around 1958 to test physics properties of irradiation targets [Reed and Swanson 2006, p. 160].

#### 2.2.9.4 321-M Fuel Fabrication Facility

The Fuel Fabrication Facility in Building 321-M was designed and built to manufacture aluminum-clad fuel elements for irradiation in the SRS reactors. In this process, enriched uranium metal was alloyed with aluminum in concentrations required for reactor irradiation. The alloy was cast into hollow cylindrical ingots from which preextrusion billet cores were machined. These cores were extruded into logs that were machined into sections, encased in aluminum, and coextruded into tubes. Cores for other types of tubes were fabricated and assembled into billets in other facilities (LiAl in Building 320-M and NpO<sub>2</sub>-Al in Building 235-F) before they were received at 321-M. The following processes were used on materials in Building 321-M [LaBone 1996]:

- Charge preparation. Enriched uranium metal from the Oak Ridge Y-12 Plant was weighed on an analytical balance in a hood in the charge preparation room then transferred to a criticality-safe storage container. The process monitor computer selected the correct amount of uranium, aluminum, and UAl scrap for each casting charge. After the uranium was weighed, it was wrapped in aluminum foil and placed in a charge can. UAl scrap was also weighed on scales in the casting and machine rooms and stored in the UAl scrap rooms.
- UAl alloy storage. UAl alloy storage facilities consisted of two rooms containing metal storage racks that provided spacing necessary for criticality safety. Large pieces, such as ingots and extrusion billet cores, were stored directly on the racks in one storage room. Smaller pieces, such as lathe turnings, were stored in metal cans on racks in the other storage room. The casting area contained a hooded press for crushing large UAl scrap for remelting. The hood exhaust ran at 1,500 cfm through a single HEPA filter.
- UAl casting. Uranium metal, aluminum, and UAl scrap were blended and cast into hollow cylinders. Alloys were melted in graphite crucibles in induction furnaces. The alloy was then cast in graphite molds. Used graphite crucibles were stripped of adhering UAl particles, assayed, and buried. UAl scrap was stored in cans in the UAl scrap storage rooms for recovery or returned to the Oak Ridge Y-12 plant.

The facility had three induction-heated furnaces for alloy and scrap recovery, two recirculation air electric resistance furnaces for heating graphite molds, and four induction heaters for graphite core removal. It also had scales for weighing castings, a material cooling hood, a filter cleaning glovebox, a pulse height analyzer, a californium shuffler for determining the amount of <sup>235</sup>U scrap and waste, and material handling carts.

Casting furnaces were installed in hoods. Two hoods were exhausted at 5,900 cfm and one at 5,400 cfm through a pair of HEPA filters connected in parallel. The gloveboxes and cooling hoods were exhausted at 5,600 cfm through three pairs of HEPA filters in parallel. The room exhaust system consisted of six HEPA filters in parallel exhausted at 8,500 cfm.

- **Core machining.** Enriched UAl castings were machined to dimensions for preextrusion cores on lathes. Machining scrap was collected and stored for recovery in the casting process. The core machine shop contained two hooded lathes. The hood covering each lathe was exhausted at 4,000 cfm through three pairs of HEPA filters in parallel. The facility was also equipped with a power hacksaw, a tube shearer, a balance for weighing cores and scrap, an ultrasonic cleaner, a mechanical chip compacter, and workstations for handling scrap. All this equipment except the tube shearer and ultrasonic cleaner was in hoods exhausted at 1,500 cfm through single HEPA filters.

The room exhaust consisted of three pairs of HEPA filters in parallel exhausted at 12,000 cfm.

- **Billet assembly.** UAl cores were assembled into extrusion billets by encasing them in aluminum components. LiAl target billets were brought into the facility at this point in the process. The exterior aluminum joints were welded and the weld integrity checked. Fumes from the processes in this facility were exhausted through two roof stacks.
- **Billet outgassing.** Billets were vacuum outgassed in outgassing furnaces at a temperature greater than the extrusion temperature.  $\text{NpO}_2\text{-Al}$  billets were brought into the facility at this point in the process. UAl and LiAl billets were outgassed in all the outgassing furnaces;  $\text{NpO}_2\text{-Al}$  billets were restricted to the furnaces equipped with HEPA-filtered exhaust systems.

The billet evacuation area consisted of six electric resistance furnaces, three of which were in metal enclosures with filtered exhaust systems. The exhaust system for two enclosures consisted of two single HEPA filters in parallel. The other system had two pairs of HEPA filters in parallel. The total airflow through each enclosure and the vacuum pump exhaust was 1,800 cfm.

SRS installed an NTG in 321-M around 1958 to test physics properties of enriched fuel elements [Parkinson and Baumann 2000].

SRS ceased production activities at all 300-M Area buildings in 1987. SRS removed all equipment by 1997, and demolition of buildings started in 2003 [Reed and Swanson 2006, pp. 198–199].

The radionuclides of concern are shown in Table 2-41. In addition, neptunium is a radionuclide of concern for 1961 through 2005 [ORAUT 2016].

Table 2-41. 321-M Fuel Fabrication Facility.<sup>a</sup>

<b>Area</b>	M
<b>Description</b>	321-M, Fuel Fabrication Facility
<b>Period</b>	Constructed in 1957; decontamination and decommissioning completed in December 2005.
<b>Radionuclides of concern</b>	U-234, U-236, U-235, U-238 (U-235 is assumed for recycled uranium intakes), Np-237 (1961–1998), Pu (1963–1965, 1968–1970, 1975, 1979–1980), Th-232 October 1972 and after

a. Sources: LaBone [1996, p. 54]; DOE [2005, p. 72]; SRNS [2017, p. 9]; ORAUT [2016]; Reed and Swanson [2006, pp. 188–189]; Bebbington [1990, p. 114], DuPont [1980d, pp. 34, 39, 97]; DuPont [1980a, p. 69].

### 2.2.9.5 777-M Pile Physics Laboratory

The Pile Physics Laboratory in Building 777-M, later designated 777-10A, contained various experimental reactors used for lattice and reactivity experiments. The building was originally designed to contain two experimental reactors: (1) the Process Development Pile (PDP), a nuclear mockup of the site's production reactors, and (2) the Standard Pile (SP), suitable for calibration purposes and for supplying neutrons for experimental purposes. The Subcritical Experiment, a tank filled with deuterium oxide, was closely coupled to the SP reactor. Many of the physics parameters of the SRS lattices could be determined in this facility using relatively little fuel. The reactors in 777-M were effectively shut down in the 1980s after most of their functions had been assumed by computer modeling [Strack 2002, p. 413].

### 2.2.9.6 Nuclear Test Gauge

SRS built the NTG in 1955 in Building 777-M to complement the test reactor in 305-M. The NTG was a subcritical multiplying facility measuring 6 ft<sup>3</sup> with an adjoined measurement area. SRS fueled the NTG with <sup>235</sup>UAl moderated by H<sub>2</sub>O. SRS applied calibration standards written for the reactor to the NTG. SRS tested components by passing them through an axial port. By 1955, both the NTG and reactor were used in tandem. SRS tested uranium metal in the reactor and alloy tubes and control rods in the NTG. Over time, SRS designed an even more efficient and smaller NTG. The new design was 10 times more efficient than the original NTG in 305-M. The NTG in 305-M was replaced by smaller NTGs in 320-M and 321-M [Strack 2002, pp. 350–351, 424; SRNS 2017; NIOSH 2011; ORAUT 2016; Parkinson and Baumann 2000].

The radionuclides of concern are shown in Table 2-42.

Table 2-42. 777-M Pile Physics Laboratory.<sup>a</sup>

<b>Area</b>	M
<b>Description</b>	777-M, 777-10A
<b>Period</b>	Construction completed in 1953; decontamination and decommissioning completed in March 2006.
<b>Radionuclides of concern</b>	Highly enriched uranium, fission products, Th (1954–1984), Np (1955, 1965–1968)

a. Sources: Strack [2002, p. 413]; SRNS [2017, p. 4]; NIOSH [2011]; ORAUT [2016]; Parkinson and Baumann [2000].

### 2.2.10 S-Area Defense Waste Processing Facility

Radioactive waste sludges are stored in large underground tanks at SRS. DWPF is in the 200-S Area north of the existing 200-F and 200-H separations areas.

#### DWPF Process

Radioactive wastes are stored in existing H-Area waste tank facilities in the form of settled sludge, supernatant liquid, and salt cake. The insoluble solids (sludge) that have separated from the aqueous fraction of the waste by settling undergo aluminum dissolving and washing operations in preparation for the feed to DWPF. The aqueous fraction undergoes treatment with sodium tetraphenylborate to precipitate the cesium and potassium salts. In addition, a slurry of sodium titanate is added to absorb the soluble strontium and plutonium. The precipitate is washed and concentrated in preparation for feed to the DWPF.

After recovery of the organic fraction of the tetraphenylborate precipitate, the aqueous product is blended with the sludge feed and adjusted with glass formers. The adjusted feed is vitrified to a borosilicate waste glass in a slurry-fed joule-heated melter. The waste glass is encapsulated in stainless-steel canisters and stored on site in the glass waste storage building.

Aqueous waste from DWPF is chemically treated with caustic and recycled to the H-Area Tank Farm. The radionuclide composition of the incoming DWPF feed stream varies depending on which tanks of waste are being processed. Two process streams represent the high and low ends of the range of  $^{238}\text{Pu}/^{90}\text{Sr}$  concentrations selected to design monitoring programs. The sludge-slurry feed stream (Stream 1) has a specific activity of 56 Ci/gal. Table 2-43 shows the radionuclides of concern in the sludge-slurry feed stream (Stream 1) [LaBone 1996].

Table 2-43. S-Area DWPF, Sludge-Slurry Feed Stream (Stream 1).<sup>a</sup>

<b>Area</b>	S
<b>Description</b>	200-S, DWPF, Sludge-Slurry Feed Stream (Stream 1)
<b>Period</b>	Operations began in 1996. Currently operating.
<b>Radionuclides of concern</b>	Sr-90, Ce-144, Ru-106, Pu-241, Cs-137, Pu-238

a. Sources: LaBone [1996, p. 58]; Reed and Swanson [2008, p. 58]; SRNS [2023].

The precipitate-slurry feed stream has a specific activity of 37 Ci/gal with the majority of the external dose being delivered by the same radionuclides as from the sludge-slurry stream. Table 2-44 shows the radionuclides of concern in Stream 201.

Table 2-44. S-Area DWPF, Precipitate-Slurry Feed Stream (Stream 201).<sup>a</sup>

<b>Area</b>	S
<b>Description</b>	200-S, DWPF, Precipitate-Slurry Feed Stream (Stream 201)
<b>Period</b>	Operations began in 1996. Currently operating.
<b>Radionuclides of concern</b>	Cs-137, Ce-144, Ru-106, Sr-90, Cs-134, Pu-238, Pu-241

a. Sources: LaBone [1996, pp. 58–59]; Reed and Swanson [2008, p. 58]; SRNS [2023].

## 2.2.11 Z-Area

Waste tank supernate (low-level waste) is transferred from H-Area to Z-Area via pipeline. At Z-Area, the salt solution (supernate) is mixed with cement, fly ash, and blast furnace slag, and the resulting grout is pumped to vaults where it sets into a saltstone monolith.

### Decontaminated Salt Solution

The chemical and radionuclide content of the decontaminated salt solution varies according to which tanks of waste are being processed. Average compositions have been determined from laboratory tests with actual waste from F- and H-Areas as well as calculations of fission product generation determined from the nuclear properties of SRS fuels and reactor exposure.

Two process streams have been identified based on which radionuclides would deliver the majority of the dose and which radionuclides could be used for tracers. The process streams are the decontaminated salt solution and the saltstone. Tables 2-45 and 2-46 show the radionuclides of concern in the decontaminated salt solution and in the saltstone [LaBone 1996].

Table 2-45. Z-Area, waste tank supernate, decontaminated salt solution.<sup>a</sup>

<b>Area</b>	Z
<b>Description</b>	Z-Area waste tank supernate, decontaminated salt solution
<b>Period</b>	Radioactive operations began June 1990. Currently operating.
<b>Radionuclides of concern</b>	Ru-106, Cs-137, Sr-90, Pu-238

a. Source: LaBone [1996, p. 60]; SRR [2009, p. 39]; SRNS [2023].

Table 2-46. Z-Area, Saltstone Disposal Facility.<sup>a</sup>

<b>Area</b>	Z
<b>Description</b>	Z-Area waste tank, Saltstone Disposal Facility
<b>Period</b>	Radioactive operations began June 1990. Currently operating.
<b>Radionuclides of concern</b>	Ru-106, Cs-137, Sr-90, Pu-238

a. Source: LaBone [1996, p. 60]; SRR [2009, p. 39]; SRNS [2023].

**2.2.12 A-Area**

The Savannah River Technology Center (SRTC) is in the A-Area and has historically been involved with analyses of fuel and target material and has dealt with various alpha-emitting radionuclides, including <sup>238</sup>Pu and isotopes of americium, curium, californium, and various other TRU elements with high specific alpha activities [Till et al. 2001, p. 257].

A-Area contained numerous buildings, some of which were administrative for the entire site. The A-Area (with building numbers in the 700s) also contains SRNL (Building 773-A), Main Laboratory Buildings (735-A and 735-11A), and the Liquid Waste Handling Facility (Building 776-A). The primary buildings of radiological concern are discussed further. A map of the portion of A-Area with those buildings is shown in Figure 2-35.

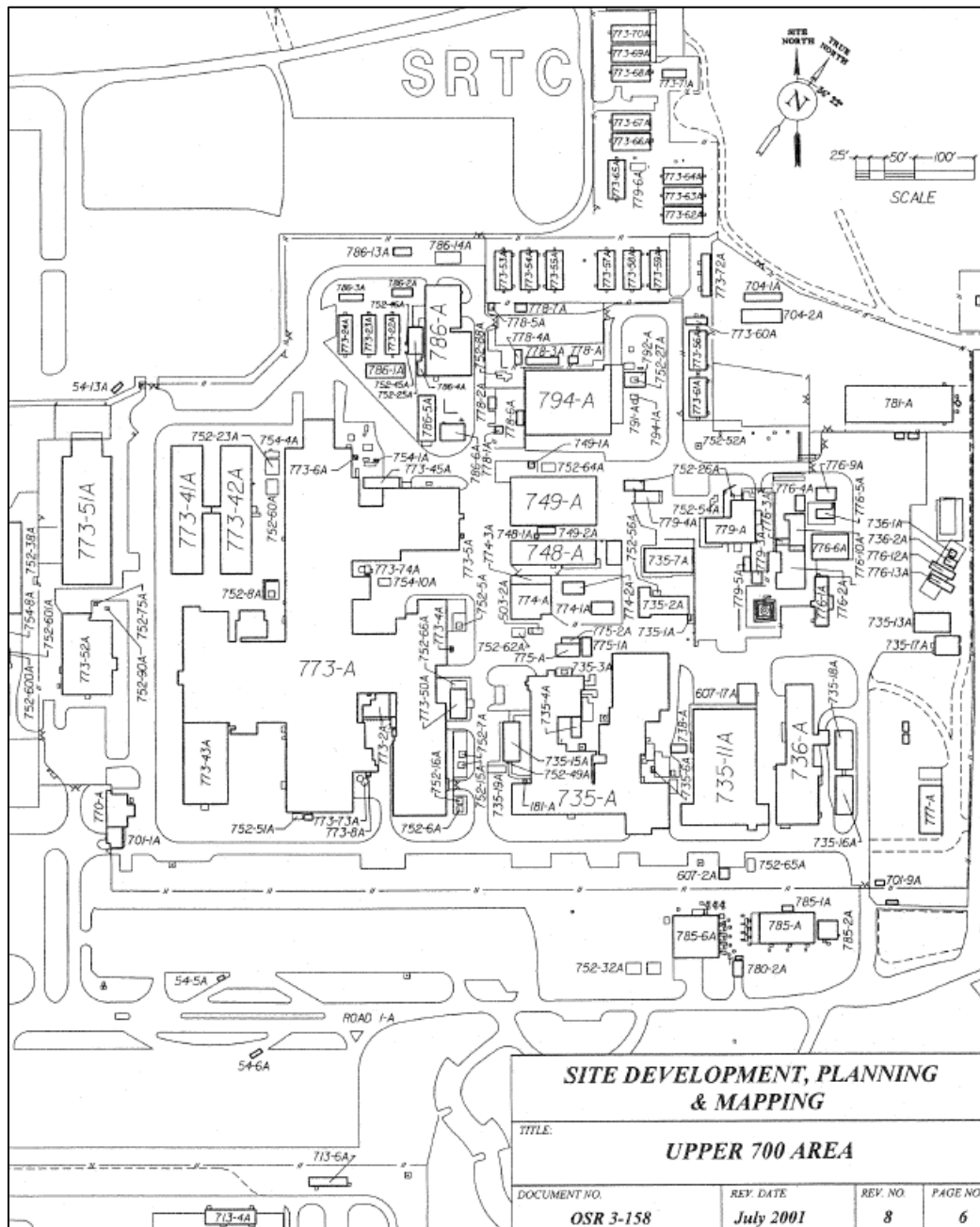


Figure 2-35. Upper 700 Area [WSRC 2001a, p. 10].

### 2.2.12.1 773-A Savannah River National Laboratory

Building 773-A was divided into three major parts. Part I was further divided into three sections. A layout of 773-A with labels is given in Figure 2-36. Descriptions follow [DuPont 1957g]:

- Section 1-A. The front wing of Building 773-A contains the administrative and technical services offices, the change area, and "cold" laboratories.
- Sections 1-B and 1-C. The main central portion with laboratory modules were back-to-back but separated by a utility corridor and flanked on the outside walls by auxiliary modules and personnel corridors. Both of these sections were equipped to handle radioactive materials.
- Section 2. Physical property and corrosion testing, metallographic and X-ray diffraction studies, and general physical and chemical studies were conducted in support of the reactor fuel development program.
- Part II (Caves). Contained facilities for physical, chemical, and metallurgical work with high-level radioactive materials. By October 1954, a variety of cave apparatus was available including dimensioning equipment, contour tracing gauge, weighing scales, analytical balances, binocular periscope, a stage cutoff machine for sectioning irradiated metal, chemical apparatus for dissolving fuel elements, and hardness testers.
- Part III, Main Technical Laboratory Metallurgical Extension. Part III provided increased space for the Metallurgical Section of the Technical Division. It housed equipment required for the development of new fuel elements. It was divided into the element fabrication area, electroplating and cleaning area, tube-cleaning area, vertical assembly area, nondestructive testing area, machine shop, and storage area. Mockups of fuel assemblies were made in this area.

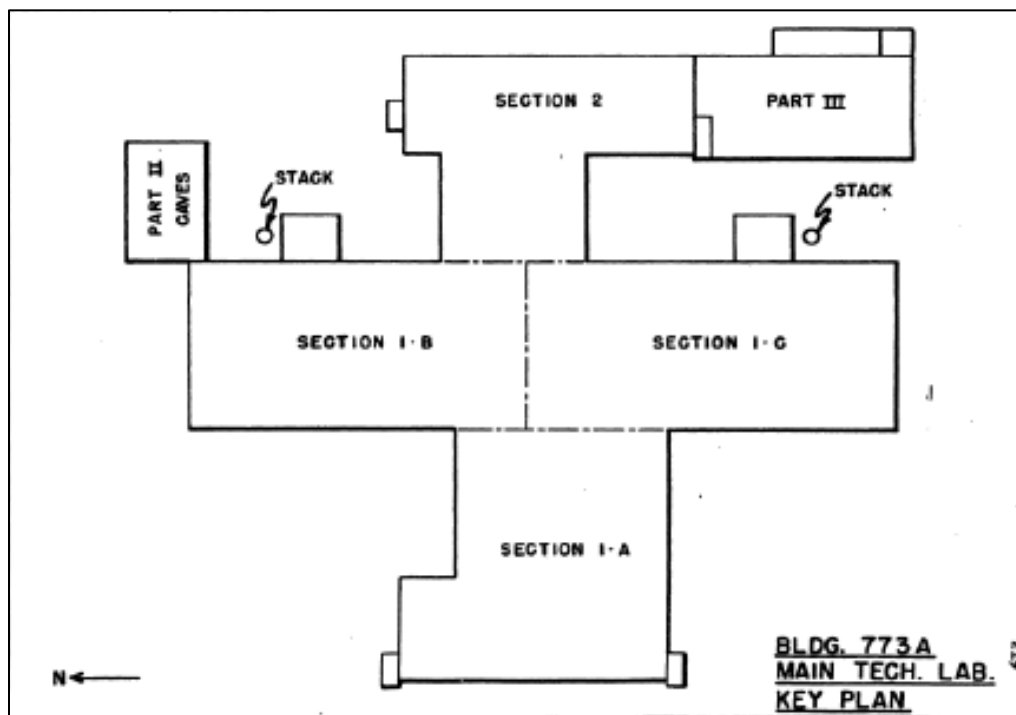


Figure 2-36. Building 773-A, Main Technical Laboratory Building [DuPont 1957g, p. 204].

SRNL continues to use 773-A. Over the years SRS conducted various research in the building's labs and hot cells. Some of the many processes included the following [LaBone 1996]:

- Actinide Technology. The Actinide Technology Division provided technical support for the plutonium and enriched uranium production facilities, Building 221-F, FB-Line, H-Canyon, and HB-Line. The work in the laboratories was primarily process troubleshooting and some independent research. The radionuclides of concern in these buildings and therefore in the Actinide Technology section of Building 773-A include plutonium, neptunium (1961 to 1988), americium, californium, thorium, and uranium.
- Analytical Development. Analytical Development is responsible for 32 of the 91 laboratories in Building 773-A. It provides analytical support for research programs and conducts research and development in the area of process control and analyzer development. The material used in this section might include fission products, uranium, neptunium (1961 to 1988), curium, americium, thorium, and/or californium.
- DWPF. The DWPF Division provides support and conducts research and development in the area of waste processing. Research is conducted using glass, saltstone, and cement to fix or entrap radionuclides and toxic materials into a matrix, thereby preventing or reducing the rate of release to the environment. The material used in this section was similar to the material processed in the S- and Z-Areas and should contain principally fission and activation products, americium, californium, neptunium (1961 to 1988), thorium, and plutonium.
- Hydrogen and Fuels Technology. The Hydrogen and Fuels Technology Division is involved in development of hydride process technology and testing materials for use in corrosive and hydrogen environments. The type of radioactive material used in this section is unknown.
- Interim Waste Technology. The Interim Waste Technology Division is developing the technology for the DWPF and the ETF. Work is performed on supernatant processing, precipitation of strontium and cesium and filtration and acid hydrolysis of the resulting sludges. The characterization of waste and the investigation of the treatment of effluents by ion exchange and reverse osmosis are also being developed. The material used in this section is assumed to be similar to the material processed in the S- and Z-Areas and the ETF in H-Area. The material should contain principally fission and activation products, americium, californium, neptunium (1961 to 1988), thorium, and plutonium.
- Environmental Technology. The Environmental Technology Division processes environmental samples. Iodine-129 is extracted from neutron-activated environmental samples and separated from radiobromine by fractional distillation. Carbon-14 is extracted from plant stack samplers, and the extracted  $^{14}\text{C}$  is counted in 735-A. The material used in this section contains  $^{129}\text{I}$  and  $^{14}\text{C}$ . Potential exposure to neptunium (1961 to 1988), thorium, and  $^{252}\text{Cf}$  should also be considered.
- Laboratory Services. The Laboratory Services Division areas of primary interest include the High Level Cells and the F-Wing Experimental Area. The High Level Cells provide the shielding and confinement necessary for analysis and testing of highly radioactive material from onsite and offsite facilities. The High Bay Experimental Area (F-Wing) has research- and pilot-scale production facilities for work with alpha and neutron emitters such as  $^{239}\text{Pu}$ ,  $^{244}\text{Cm}$ ,  $^{243}\text{Am}$ ,  $^{237}\text{Np}$  (1961 to 1988), thorium, and  $^{252}\text{Cf}$ .
- Materials Technology. The Material Technology Division consists of a Robotics Development Laboratory, Powder Metallurgy Facility, Onsite Uranium Reprocess Facility, and the Hot Machine Shop. The purpose of the Powder Metallurgy Facility is to develop a new process for

fabricating reactor fuel tube cores. The Onsite Uranium Recycle Process Facility is a prototype facility to assist in the design and operation of a plant-scale Fuel Production Facility. The Hot Machine Shop provides a variety of functions and services including machining uranium metals. The material used in this section includes fission and activation products, tritium, californium, thorium, neptunium (1961 to 1988), and depleted and/or enriched uranium.

Farrell and Findley [1999] categorized radionuclides of concern by a combination of location and operation rather than operation alone. While the document directly applies to work performed in 1999 and forward, the information can also be used retrospectively. Farrell and Findley stated the following:

- B-Process waste stream. The B-Process stream consists of HEPA filters, prefilters, and associated job control waste for the 773-A B-Wing air handling systems. The B-process radionuclides of concern were plutonium, americium, and curium isotopes.
- C-Process waste stream. The C-Process stream consists of HEPA filters, prefilters and associated job control waste for the 773-A C-Wing air handling system. The C-process radionuclides of concern were plutonium isotopes.
- F-Wing Boot waste stream. The F-Wing Boot stream consists of materials removed from the F-Wing air handling system. The F-Wing radionuclides of concern were americium isotopes and  $^{244}\text{Cm}$ .
- SCH-3-D waste stream. The SCH-3-D stream consists of job control waste from 773-A shielded cells and roof. The SCH-3-D radionuclides of concern were  $^{244}\text{Cm}$ , plutonium isotopes, and  $^{241}\text{Am}$ .
- Analytical Development Section (ADS)-1 waste stream. The ADS-1 stream is predominantly waste generated within the ADS laboratories. The ADS-1 radionuclides of concern were plutonium isotopes,  $^{241}\text{Am}$ , and  $^{244}\text{Cm}$ .
- Canyon waste stream. The Canyon stream consists of wastes from Chemical and Hydrogen Technology Section (CHTS) laboratories in 773-A. Most of the CHTS laboratories process weapons grade plutonium from F-Area. The C-process radionuclides of concern were plutonium isotopes.
- Non-Canyon waste stream. The Non-Canyon stream consists of wastes generated in 773-A laboratories outside of previously discussed ADS and CHTS laboratories. The Non-Canyon waste stream radionuclides of concern were plutonium isotopes and  $^{244}\text{Cm}$ .
- Casks waste stream. The Casks stream consists of high-level waste sludge and/or supernate sent to SRTC for analysis. The casks waste stream is broken into two separate streams, namely the Sludge substream and the Supernate substream. The Sludge radionuclides of concern were  $^{244}\text{Cm}$  and plutonium isotopes. The Supernate waste stream radionuclides of concern were plutonium isotopes.
- Californium waste stream. The Californium stream consists of job control waste associated with former californium processing facilities (the Californium Production Facility and Medical Source Facility) in F-Wing of Building 773-A. The radionuclide of concern was  $^{252}\text{Cf}$ .

Table 2-47 lists the radionuclides of concern for the Main Laboratory.



Table 2-47. 773-A, Main Laboratory.<sup>a</sup>

<b>Area</b>	A
<b>Description</b>	773-A, Main Laboratory Building
<b>Period</b>	Construction completed in January 1953. Currently operating.
<b>Radionuclides of concern</b>	H-3, Pu-238, Pu-239, Am-241, Cm-244, Cf-252, Th-232, Np-237, U-234, U-235, U-238, fission products, Th-232 October 1972 and after. <sup>b</sup>

a. Source: Strack [2002, p. 410]; SRNS [2023].

b. The laboratory processed samples from many locations. Amounts and mixtures varied depending on project.

### 2.2.12.2 Buildings 735-A and 735-11A Radiological and Environmental Science Facilities

Buildings 735-A and 735-11A house the Radiological and Environmental Science Facilities, previously called the Health Physics Laboratory. The work in these facilities involves the analysis of mainly low-level environmental and bioassay samples. Table 2-48 lists the radionuclides of concern for these facilities [WSRC 2001b, p. 119].

Table 2-48. 735-A and 735-11A Radiological and Environmental Science Facilities.<sup>a</sup>

<b>Area</b>	A
<b>Description</b>	735-A and 735-11A, Radiological and Environmental Science Facilities
<b>Period</b>	Building 735-A was constructed in 1953, 735-11A in 1964. Currently operating.
<b>Radionuclides of concern</b>	H-3, Pu-238, Pu-239, Am-241, Cm-244, Cf-252, Th-232, Np-237, U-234, U-235, U-238, fission products, Th-232 October 1972 and after.

a. Sources: DOE [2005, p. 39]; WSRC [2001b, p. 119]; SRNS [2023].

### 2.2.12.3 Building 776-A Liquid Waste Handling Facility

The SRS Liquid Waste Handling Facility is in Building 776-A. This facility collects the aqueous waste from Building 773-A. Facilities in Building 776-A are provided to strain solids from the waste streams, collect waste in batch receiving tanks, sample and adjust the pH of the waste tank contents, and transfer the waste to tank trailers for disposal. The radionuclides are believed to be a composite of all the material that is used in Building 773-A operations. Table 2-49 lists the radionuclides of concern for the Liquid Waste Handling Facility.

Table 2-49. 776-A Liquid Waste Handling Facility.<sup>a,b</sup>

<b>Area</b>	A
<b>Description</b>	776-A, Liquid Waste Handling Facility
<b>Period</b>	Constructed in 1953. Currently operating.
<b>Radionuclides of concern</b>	H-3, Pu-238, Pu-239, Am-241, Cm-244, Cf-252, Th-232, Np-237, U-234, U-235, U-238, fission products, Th-232 October 1972 and after.

a. Source: DOE [2005, p. 41]; SRNS [2023].

b. The types of radioactive material handled at the Liquid Waste Handling Facility is believed to be a composite of the material that is used in the Building 773-A operations listed above.

## 2.2.13 D-Area Heavy-Water Production and Reprocessing

The D-Area, also known as 400-Area, occupied an area of 159 acres with approximately 29 facilities and buildings that supported the major production of D<sub>2</sub>O activities [Reed and Swanson 2008, pp. 84, 87]. In late 1950, the supply of heavy water was recognized as a critical factor for the early startup of the proposed nuclear reactors. The heavy-water facilities in the 400-D Area were the first production facilities at SRS. The primary process, known as the Girdler-Sulfide (GS) process, required 144 heavy cylindrical towers ranging up to 6.5 ft in diameter and 120 ft tall in six parallel rows of 24 towers [Morris et al. 2000, p. 5]. A heavy-water production plant in the 400-D Area began operation in October 1952 to concentrate heavy water from Savannah River water to moderate and cool the site's reactors and to reconcentrate heavy water that had become diluted due to reactor operations [Till et

al. 2001, p. 67; Morris et al. 2000, p. 2]. A major part of the heavy-water production plant was closed in 1957 to 1958 because “there was an ample reserve supply on hand to supply foreseen demands” [Bebbington 1990, p. 226]. New or virgin heavy-water production continued until January 8, 1982 [Morris et al. 2000, p. 7]. Three units (tower columns) repurposed in 1957 for moderator rework with rework starting on December 6 [DuPont 1957h, p. 6; DuPont 1958d, p. 171]. In addition to the activation of native elements, water treatment additives, corrosion products, and fission products released by failed fuel elements, the heavy water in the SRS reactors contained significant amounts of tritium, which built up as the reactors operated [DOE 1997, p. 182]. Reconcentration of heavy water with a small staff continued until 1999 [WSRC 2004a, p. 5; SRNS 2009, p. 101].

A control laboratory was in D-Area. Originally the laboratory in 772-D processed heavy-water samples for tritium content, but by the 1970s samples were analyzed for beta, gamma, and alpha elements. Over time some portions of the 772-D were contaminated with measurable beta/gamma activity [WSRC 1991b]. Work in the 772-D laboratory continued until 1998 [WSRC 2004b, p. 6].

D-Area was one of the first building areas to be identified for deactivation and decommissioning. After 1982, buildings were removed and the towers were scrapped for their steel. The remaining process buildings were used as long as needed then deactivated. Many buildings that housed operations sat inactive and were demolished in the 2000s. The Control Laboratory, 772-D, was demolished in 2004; the Concentrator Building, 420-D, in June 2006; and the Rework Handling Facility, 420-2D, in September 2006 [Reed and Swanson 2008, pp. 124–125; SRNS 2017, p. 5].

### Heavy-Water Process

The heavy-water process and facilities description essentially involved three separate processes as depicted in Figure 2-37 and summarized below [Reed and Swanson 2008, p. 120]:

- GS (a.k.a. Counter-Current Gradient and Hydrogen Sulfide) Process (Buildings 411-D, 412-D, and 413-D). The process by which the heavy-water content of ordinary water was increased from about 0.015% to 15% to 20% by counter-current exchange of water and hydrogen sulfide gas first through cold towers and then through hot towers [Bebbington 1990, pp. 253–255].
- DW (Distillation of Water, a.k.a. DuPont Water) Process (Building 420-D). The distillation, under vacuum, to further concentrate heavy water from GS. The DW process concentrated heavy water from the GS process of about 20% to an average of 90% or a maximum of 98% heavy water [Hickey and Cragle 1985, p. 8].
- E (Electrolysis of Water) Process (Building 421-D). Batch electrolysis of heavy water from DW to bring it up to 99.8% heavy water [Hickey and Cragle 1985, p. 8].

Heavy-water processing was performed as two separate processes: (1) virgin heavy water and (2) reconcentration or rework heavy water. These processes were not housed in different buildings (Figure 2-37), but the processed materials were kept separate. As of 1959, the virgin heavy-water process was conducted in the remaining eight extraction units in Building 412-D and 9 of the 12 towers in the DW plant. The reconcentration or rework heavy water was conducted in the other three DW towers [Reed and Swanson 2008, p. 120].

Two process streams represent the high and low ends of the range of tritium, fission products, or induced activity concentrations.

Heavy water contains deuterium, a nonradioactive isotope of hydrogen, and does not represent a radiological hazard to workers. The major hazard associated with its production was from a byproduct chemical, hydrogen sulfide, created during the production process of the heavy water. Tritium was a

radiological hazard during the reprocessing of the degraded heavy water from the reactor's moderator and cooling systems [Hickey and Cragle 1985, p. 10].

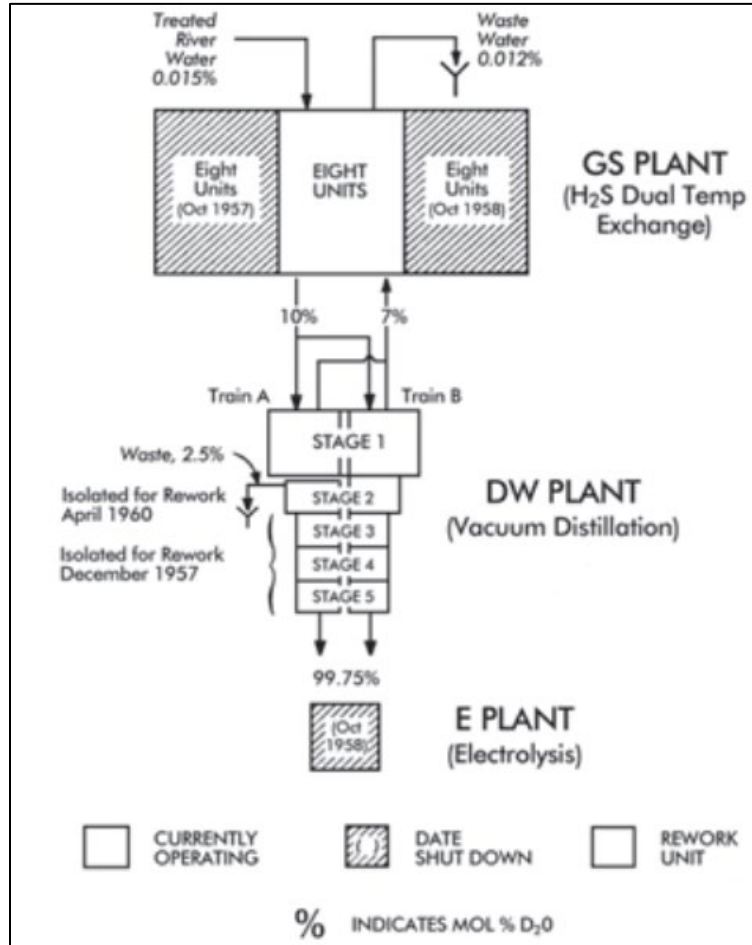


Figure 2-37. Heavy-water process and facilities [Reed and Swanson 2008, p. 120].

It has been known from the beginning of the heavy-water reprocessing that tritium was a contaminant [Hickey and Cragle 1985, p. 11]. As noted in the monthly meters and film badge reports for April to June 1954 [DuPont 1954a, p. 14]:

*For the second successive month no gas overexposures occurred.*

*The bioassay program initiated in April (1954) was concluded for Finishing Area personnel. No detectable tritium was found.*

*Slight tritium contamination has appeared in process material being returned to the Extraction Area from the DW Plant. Close checks are maintained on the concentrations involved.*

*Tritium contamination appearing in the final "E" Plant product reflected the contamination present in material added to the system from the Reactor Areas and other outside sources. Final product contamination averaged approximately 50 [ $\mu\text{Ci/mL}$ ] during recent weeks, with a maximum of 125 [ $\mu\text{Ci/mL}$ ].*

Therefore, deuterium workers were monitored via bioassay sampling for internal tritium exposure. Bioassay for fission products or induced activity is also found in the claim file records for deuterium workers [NIOSH 2022].

Bioassay for fission products or induced activity was routinely conducted for deuterium workers as indicated in the claim files [NIOSH 2022] and health physics procedures DPSOL 193-302 Revisions 2 to 5 and 7, and DPSOL 193-302T due to rework of reactor moderator [DuPont 1970b, 1971a,b, 1976, 1985].

There were two separate facilities or processes for heavy water, the heavy-water production plant that created virgin heavy water from the local water source (Savannah River) and the rework facility that reconcentrated the degraded heavy water from the reactors [Reed and Swanson 2008, p. 120]. Degraded moderator resulting from oil leakage, sample collections, and spills was collected in 55-gal drums and either shipped to 100-P for preliminary cleanup or shipped back to the rework facility in the 400-Area at a rate of approximately 5,000 drums per year [DuPont 1961c, p. 217].

PDP reactor heavy water was reconcentrated every 2 years. The moderator in the PDP test reactor and associated facility was partially replaced each month. In the early part of the program, the entire 220,000-lb charge was reconcentrated about once every 2 years. Under a later program, about 10,000 lb of moderator at a concentration of 99.0 to 99.6 mol heavy water was reconcentrated each month in the final stages of the DW Plant [DuPont 1961c, p. 219].

Heat exchanger leaks, reactor purges, and other discharges allowed radioactive water to escape the reactor's pool water many times. In addition to the activation of native elements, water treatment additives, corrosion products, and fission products released by failed fuel elements, the heavy water in the reactors contained significant amounts of tritium, which built up as the reactors operated [DOE 1997, p. 182].

Routine bioassay for fission products or induced activity occurred beginning in about 1965. Note that the first list of urine bioassay limits obtained was dated July 12, 1962, and included fission product and induced activity [Mann 1962, p. 2]. A "Bioassay Information" document dated December 10, 1959, did not include fission product or induced activity [Mann 1962, p. 3].

Degraded "half-heavy water" from dismantled weapons was recycled through the heavy-water rework facility for reenrichment. The heavy-water rework facility continues to produce deuterium gas for nuclear weapons from existing heavy-water stocks using an electrolytic process [DOE 1997, p. 161]. This is also shown in the heavy-water process flow chart [DOE 1997, p. 134].

Table 2-50 lists the radionuclides of concern for D-Area heavy-water production.

Table 2-50. 400 D-Area heavy-water production and reprocessing.<sup>a</sup>

<b>Area</b>	D
<b>Description</b>	400-D, heavy-water production and reprocessing
<b>Period</b>	Initial startup in the winter of 1951–1952. Full operation in May 1953. Production of new or virgin heavy water ended on January 8, 1982. Reconcentrated heavy water with a small staff until the shutdown of all SRS reactors in 1989. Rework continued through 1999. Laboratory work continued through 1998.
<b>Radionuclides of concern</b>	Minimal tritium, fission products, and induced activity from reactor water through 1999.

a. Sources: Bebbington [1990, p. 226]; Morris et al. [2000, p. 7]; Reed and Swanson [2008, pp. 124–125]; Till et al. [2001, p. 67].

### 2.2.14 Corrosion Mock-up Experiment and Semi-works Laboratory

Two technical development facilities were the first to be finished and put in operation. The CMX was a reactor support facility for determining if Savannah River water would present unexpected problems in reactor cooling. TNX provided a wide range of technical support and development for separations. Neptunium is a radionuclide of concern for 1958 to 1962.

CMX (679-T but originally 679-G), was used for hydraulic flow testing of all new types of fuel and target assemblies before they were approved for use in the reactors. There were three primary components initially located in the CMX building: the Water Laboratory, Health Physics, and Building Services. The Water Laboratory was housed in the tower with laboratory facilities in the long arm of the building. A boiler was in that same area and connected to an outside water treatment plant. The Health Physics Department (HPD) was initially located in CMX until its own facilities were constructed. Building services in CMX included the change rooms, lockers, showers, and bathrooms. CMX was closed in 1983, and the remaining active test facilities were moved to Building 773-A [Swanson and Reed 2006, p. 48]. A photo of CMX is given in Figure 2-38.



Figure 2-38. CMX Building 679-T [Swanson and Reed 2006, p. 68].

TNX (678-T but originally 678-G), was used as a staging area for receipt and testing of large process equipment destined for use in production facilities at SRS. Since 1978, the TNX Area expanded from 3 original buildings to 32 buildings. The 32 buildings include administrative offices, process buildings for large-scale experimental demonstrations, laboratories for both research and analytical work, pilot-scale facilities, bulk tank storage, industrial wastewater processing facilities, and warehouse storage for a wide range of chemical and specialty equipment [DOE 1988]. A photo of TNX is shown in Figure 2-39. Table 2-51 lists the radionuclides of concern for CMX and TNX.

Historically, within the TNX Area, some areas in Buildings 672-T, 677-T, and 678-T contained radioactive materials.

Building 672-T, constructed in 1983, consisted of a chemical processing facility, an electrical and instrumentation shop, and a control room. The chemical processing facility contained the Integrated DWPF Melter System, the Slurry Receipt Adjustment Tank/Slurry Mix Evaporator, the Ion Exchange Facility, and the Americium/Curium (Am/Cm) Melter though only non-radiological work was conducted in this building [DOE 1988].



Figure 2-39. TNX Building 678-T [Swanson and Reed 2006, p. 74].

Table 2-51. CMX and TNX.<sup>a</sup>

<b>Area</b>	T
<b>Description</b>	679-T CMX and 678-T TNX
<b>Period</b>	CMX and TNX began operations in 1951; decontamination and decommissioning were completed for Building 679-T in May 2004 and for Building 678-T in September 2004.
<b>Radionuclides of concern</b>	678-T TNX: depleted uranium

a. Source: Strack [2002, pp. 413–417]; DOE [1988, p. 6]; SRNS [2017, p. 11].

Building 677-T consisted of a maintenance shop, a laboratory where small-scale demonstration projects were operated, a bioremediation laboratory, and a bay area where larger scale pilot projects were demonstrated. Equipment in Building 677-T consisted of the Plutonium Waste Incinerator and Mobile Offgas System, the Uranium IV Facility, the Geometrically Favorable Dissolver, and the Centrifugal Contactor. Equipment and surrounding facilities housing the Geometrically Favorable Dissolver and the Centrifugal Contactor were radioactively contaminated. The Uranium IV Facility has a process history of radioactive work.

Building 678-T and the 678-T Annex, constructed in 1979, were a chemical processing facility and supporting laboratories. Equipment in 678-T included the Offgas Components Test Facility, the Mott Filter Test Stand, and a tank farm. Historically, the tank farm in this facility was used to store raw material feed makeup for the Precipitate Hydrolysis Experimental Facility, for development of melter feed, and for storage of aluminum nitrate for calibration of separations frames columns. Building 678-T was the former site of an evaporator that exploded in 1953. The tank farm, the former site of the evaporator, the laboratory, and a concrete pad outside the west wall of the building were all radioactively contaminated [DOE 1988, p.19].

Buildings 677-T and 678-T were contaminated with depleted uranium [DOE 1988, p. 6].

### **2.2.15 Central Shops**

Central Shops, in N-Area, was a collection of several dozen buildings in a grid of streets in a 100-acre tract located centrally to the entire site [DOE 2005, p. 23]. According to Strack [2002, p. 191]:

*Every brand of metal, rectangular, gable-roofed prefabricated buildings and structures could be found in Central Shops, where massive buildings housed men of all crafts, warehouses were filled with supplies, and streets arranged in a grid teeming with cars, trucks, and heavy machinery.*

*Components of plant equipment were fabricated in several large shops, some enclosing as much as two acres. These buildings were essentially factories and operated 24 hours a day, seven days a week. The buildings also contained the offices of engineers, foremen, and supporting personnel.*

The Construction Division was based in Central Shops and was responsible for facility maintenance and for new construction. The division included field project managers, engineers, and superintendents for most of the major crafts groups such as ironworkers, pipefitters, layout, electrical, sheet metal, and carpentry. Construction also had resident subcontractors who were specialists in piping, electrical work, insulation, and testing and inspection of completed works [Reed and Swanson 2006, p. 69]. Workers from Central Shops could be assigned to any area of SRS for construction or repair work [Anderson 2021, p. 2].

#### **2.2.15.1 Process Heat Exchanger Repair Facility**

The Process Heat Exchanger Repair Facility, also known as the Ford Building, was a one-story metal frame structure on a concrete slab covering approximately 9,700 ft<sup>2</sup> in the extreme southeastern tip of the Central Shops. The building, designated originally as 690-N but later designated 690-G, consisted of a machine shop with offices, storage rooms, restrooms, and a service area [Blankenship et al. 2021, p. 9]. This facility was used during the 1950s exclusively for testing of Ford Motor Company manufactured motor control packages for control rod drive mechanisms before their installation in the SRS reactors [Blankenship et al. 2021].

During the early 1960s, the SRS reactors were operating at higher power levels and the failure of the heat exchangers prompted conversion of this facility to perform maintenance, repair, and rework. The Ford Building was investigated in June 1961 as a repair facility for heat exchangers that were contaminated from use in the reactors [DuPont 1961c, p. 250]. By July 1961, contaminated heat exchangers were being repaired in the building using containment huts [DuPont 1961d, p. 17].

Building modifications to change the mission for Building 690-N from a clean facility to a 100-t heat exchanger maintenance and repair facility included the installation of an inner building, a sealed containment shell, and a HEPA filter system that maintained a slight negative pressure within the shell [Blankenship et al. 2021, p. 9].

Maintenance repair of the heat exchangers continued until the procurement of new heat exchangers for the reactors in the early 1980s. In the early through mid-1980s, the Ford Building housed construction crews that performed final assembly and testing of new, clean heat exchangers. For the remainder of the 1980s, the building was used to store equipment and miscellaneous supplies. During the early 1990s, the K Reactor had a minor leak in a heat exchanger that resulted in reactivating the facility. The facility operated again as a heat exchanger repair facility for about 6 months and was subsequently closed [Blankenship et al. 2021, p. 9]. Demolition of the Ford building was completed in November 2020 [SRNS 2020b, p. 2].

The Ford Building contained an underground storage tank for wastewater and a seepage basin, Building 904-91G constructed in 1964 for the disposal of wastewater from building processes. The seepage basin was contaminated with  $^{60}\text{Co}$ ,  $^{89/90}\text{Sr}$ ,  $^{137}\text{Cs}$ , and  $^3\text{H}$  [Simmons 1985]. Low levels of radioactivity were detected at the seepage area in 1988 while grounds maintenance work was being performed. Health Physics posted the area as a Soil Contamination Area to protect site workers from inadvertent exposure. Additionally, a larger area was designated as an Underground Radioactive Materials Area to indicate the possibility of buried material. However, further studies including ground-penetrating radar surveying and soil and core sampling showed that no buried waste was present. A study published in 1999 showed that any residual soil contamination at the unit does not pose a future risk to groundwater, a current or future risk to ecological receptors, or a current or future risk to human health [WSRC 1999].

### **2.2.16 Neptunium Operations**

Neptunium operations are detailed in ORAUT-RPRT-0065, *An Evaluation of Neptunium Operations at Savannah River Site* [ORAUT 2016].

### **2.2.17 Thorium Operations**

Natural thorium processes at SRS are well documented (such as in the 300/M-Area in 2006) and were conducted using the same buildings, equipment, and processing steps used to make natural uranium slugs.

#### **2.2.17.1 Slug Canning**

SRS canned thorium metal slugs for Hanford fuel tests in Buildings 313-M and 320-M. In mid-1953, SRNL started to accumulate a thorium inventory on the order of tons, which steadily increased over the next year and a half to about 5 t. It is reasonable to assume that the tons of thorium at the laboratory were used for the metallurgical research requested by a 1952 memorandum [Montenyohl 1952, p. 3] in support of the fuel and target development work [NIOSH 2010b].

In October 1961, Sylcor was authorized to extrude and can 600 hollow thorium metal slugs for use in the Mark IV fuel slug [DuPont 1961b, p. 227]. In 1962, Bridgeport Brass Company extruded 43,000 kg of thorium metal machined and canned at Sylcor. That work resulted in 14,000 Mark VII-T thorium slugs delivered to SRS for inspection and use in the L-8 Mar VI-B charge [DuPont 1974, p. 109]. This work continued sporadically at varying levels of production through 1963. In 1964, SRS installed vibratory compaction equipment in 313-M to facilitate the production of thorium oxide (thoria) slugs [NIOSH 2011, p. 18].

#### **2.2.17.2 Uranium-233 Production**

In 1964, SRNL developed procedures for processing irradiated thorium and thorium dioxide to separate  $^{233}\text{U}$  and recover thorium for recycling. These procedures were applied to two irradiation programs in progress: (1) 18 kg of  $^{233}\text{U}$  with 3 to 5 ppm of  $^{232}\text{U}$  was to be produced from 20 t of thorium metal, and (2) 100 kg  $^{233}\text{U}$  containing less than 65 ppm  $^{232}\text{U}$  was to be produced from 56 t of thorium dioxide [DuPont 1984d, p. 90].

Starting in August 1965,  $^{233}\text{U}$  was separated and recovered from irradiated thorium by the THOREX process in H-Canyon. This process consists of six steps: (1) dissolution; (2) acid reduction in the dissolver solution, if necessary; (3) coprecipitation (head end) of protactinium with manganese dioxide; (4) separation and recovery of thorium and uranium from fission products, protactinium, and aluminum by continuous solvent extraction with tributylphosphate; (5) concentration of purified thorium nitrate by evaporation; and (6) conversion of recovered  $^{233}\text{U}$  to uranium trioxide [DuPont 1984d, p. 90].



Once separated, the  $^{233}\text{U}$  was concentrated and treated by conversion to oxides suitable for shipment or storage in the Old HB-Line. This THOREX program was conducted through June 1967 as part of the THOREX I campaign and from 1968 through 1970 as part of the THOREX II campaign [Orth no date].

After the conclusion of the last THOREX campaign in 1970, the separations tanks were flushed many times and the targets were stored in the RBOF and in reactor basins, mainly the 100 K Basin. SRS sent the process and flush liquid wastes to the waste tanks and transferred them to railcars for shipment to the Feed Materials Production Center. Small amounts of solid waste from target manufacturing and SRNL were sent to 643-E (Old Burial Ground) and 643-7E (New Burial Ground) [Orth, no date, p. 11]. Thorium inventories for 1968 through 1971 show thorium byproduct materials from THOREX campaigns had been shipped off site by 1971. From 1968 through 1971, 138,516 kg of thorium was shipped to Fernald [Gibbs 1994].

### 2.2.17.3 MgO-Thoria

SRS produced 15 t of thoria ( $\text{ThO}_2$ ) during 1966 to verify the faster dissolution rate of the magnesium oxide-thoria observed in earlier plant tests at TNX. This MgO-thoria did not have the large particles found in the 600-lb pilot lots of thoria produced previously. Even though the packing density was low, the MgO-thoria was irradiated and dissolution testing was conducted in H-Canyon [DuPont 1984d, p. 161].

### 2.2.17.4 Thorium as Surrogates

Thorium materials were stored rather than processed in 773-A and other buildings [Burney 1972; ORAUT 2012b]. The Alpha Materials Laboratory in 773-A was placed in operation in 1973 and used thorium oxide as a surrogate for testing in gloveboxes with the  $^{238}\text{Pu}$  Fuel Form Program [DuPont 1972a, 1973b]. In 1973, gram quantity  $\text{ThO}_2$  shards were used in 773-A hot cells to test chemical vapor deposition of molybdenum [DuPont 1973b]. The thorium inventory in Building 235-F (that housed the Hot Press Facility for making fuel forms) was increased to about 1 kg (0.1 mCi) for use as a surrogate [DuPont 1972b].

During 1976, nominal use of thorium as a surrogate likely continued with remaining thorium inventories being stored or shipped off site. In 1977, thorium contamination (gram quantities) of the neptunium product in the Mark 16 fuel was successfully removed in H-Canyon. Thorium was also removed from dissolved plutonium scrap. Thorium oxalate powder prepared in 773-A was used as a surrogate for plutonium in testing of Old HB-Line exchange columns [DuPont 1977b, p. 60]. To test the PEF for potential  $^{238}\text{Pu}$  heat source development, SRNL staff put about 300 g of thorium oxide into the process line in March 1978 to functionally test the facility. No health physics problems were encountered [DuPont 1978]. By April 1978, all PEF equipment except for the hot press was tested using  $\text{ThO}_2$  as a surrogate in gloveboxes [DOE 1978b].

Beginning in 1980 and continuing into 1981, SRNL developed a process for fabrication of General Purpose Heat Source pellets as part of the  $^{238}\text{Pu}$  Fuel Form Program. In November 1980, SRNL began stocking thorium nitrate crystals in the 773-A chemical stores for use in research and as surrogate material with an inventory of 3.4 kg (0.34 mCi) [SRS 1976–1998]. Thorium was used as a surrogate for some of the work in hot cells in the PuFF in Building 235-F. Pellet fabrication was tested in the PEF. Thorium was used as a doping agent for the iridium welding agents. SRS found that some of the pellets made in PuFF were more extensively cracked than those made in PEF and many were fractured after heat-treating. Pore surfaces around the cracks contained unusually high levels of thorium, which was used as a doping agent. This work continued through 1995 [Burney and Congdon 1982; Kanne 1982; DuPont 1982; DuPont 1984e, pp. 118, 162–172; Franco-Ferreira and George 1995].

### 2.2.17.5 Thorium Fuel Cycle Technology

In 1977, SRS began new work in 773-A as part of the Thorium Fuel Cycle Technology (TFCT) Program to develop processing technology for spent thorium fuel. The scope of the program included broad evaluations to “identify viable thorium/uranium recycle strategies; research and development programs to confirm the feasibility of the selected fuel cycle or cycles; a design integration study to identify development areas and safeguards and proliferation aspects; and the development and testing of key systems, equipment, and components” [DuPont 1984a, p. 292]. Numerous thorium research and testing works were performed from 1977 until the TFCT Program was terminated in 1980.

SRNL procured unirradiated ThO<sub>2</sub> and ThO<sub>2</sub>-UO<sub>2</sub> fuel pellets for TFCT analysis. SRS received shipments of spent fuel from the Sodium Reactor Experiment, Dresden, and Elk River to support TFCT and the possible future expansion of the program [DOE 1977]. Those fuels were stored in the RBOF through 1998, but SRS never used them for research [SRS 1976–1998].

By 1978, nine cells in the high-level caves of 773-A were prepared for the Alternate Fuel Cycle Technology Program of which TFCT was a part [DuPont 1984b, p. 377]. Four and one-half kg (0.45 mCi) of unirradiated thorium oxide reflector pellets were received and used to test the effects of heat treatment on physical characteristics and dissolution. This work, performed in the high-level caves, involved mechanical grinding of ThO<sub>2</sub>. Testing of a conceptual THOREX process was evaluated using some of that same thorium oxide inventory [DOE 1978a]. Testing of the conceptual THOREX flowsheets continued at SRNL using irradiated thorium and uranium from spent fuels. Laboratory analyses were performed to evaluate alternative chemical reagents in the dissolution of thorium oxide. SRNL analysis of offgas removal work continued with simulated voloxidation (heating fuel in a furnace under oxidizing conditions) of UO<sub>2</sub> fuels in August 1978. Work at SRNL on these types of analyses continued through December 1978 [DOE 1978c].

In 1979, SRS and Hanford were planning and preparing a small number of 80% ThO<sub>2</sub>-20% UO<sub>2</sub> rod assemblies for irradiation in fiscal year 1980 and subsequent postirradiation characterization [DuPont 1984a, p. 292]. While 30 fuel rods (of varying thorium and uranium mixtures) were prepared at Hanford in 1979 and shipped to SRNL, the irradiation was canceled in May 1980 [DuPont 1984a, p. 307; Steimke 1980]. SRNL continued to use thorium in laboratory analyses in 1981 as a direct reagent and as a surrogate for other radionuclides [DOE 1987, pp. 179, 183; Monson and Hall 1981]. In 1982, SRNL conducted bench-scale roasting and dissolution of reactor-grade ceramic fuel (mixtures of ThO<sub>2</sub> and ThO<sub>2</sub>/UO<sub>2</sub>). This work was done as part of the TFCT Program to define the specific process conditions and equipment required to reprocess the thorium fuel materials [Pickett et al. 1982].

The TFCT Program ended in 1982.

### 2.2.17.6 Heat Source Surrogates

After 1982, SRS continued work using small amounts of thorium. SRS manufactured <sup>238</sup>Pu heat sources for the Galileo program through 1987 (average about 20 days per year). As part of this work, SRS workers made numerous welds for about 500 capsules. Manufacturing work was performed in PuFF and PEF, and some testing was performed in Building 773-A in hot cells. ORNL made the thorium doping agent. With the completion of the Galileo project, DOE moved the manufacture of <sup>238</sup>Pu capsules for the Cassini probe to Los Alamos National Laboratory (LANL) due to restructuring and changing missions. Plutonium-238 fuel was obtained from SRS which, along with ORNL and the Y-12 Plant, was assigned by DOE to assist in further development and testing of welds and welding inspections [Franco-Ferreira et al. 2000]. In July 1989 and April 1990, SRS received significant quantity shipments of thorium from ORNL (0.25 mCi) and LANL (0.017 mCi), most likely for use in the

effort to assist LANL production of the Cassini radioisotopic thermoelectric generators. All the thorium was stored in Building 773-A; the maximum quantity was 208 kg in February 1991. By December 1993, the thorium inventory had declined to about 170 kg and remained constant through March 1998. It then dropped to 4 kg in April 1998 [SRS 1976–1998]. A weld inspection report states that the development work ended by April 15, 1993 [Reimus et al. 1998].

#### **2.2.17.7 Welding Electrodes**

SRS used thoriated welding electrodes. Health Physics was asked to evaluate the need for radiological controls for work with these electrodes [Loring 1993]. Health Physics reported that as long as the SRS welding safety procedure was followed, no additional controls were required. However, Health Physics also stated that wastes from the tapering (grinding) of electrodes should be controlled to limit potential contamination and that Health Physics should review each electrode grinding location to ensure that adequate contamination control methods were used. Health Physics had to approve new or temporary electrode grinding locations before they could be used for that purpose [Loring 1993].

#### **2.2.17.8 Thorium Research**

Additional information and data on the use and storage of thorium materials and handling of thorium waste materials are given in Addendum 3 of the SEC petition evaluation report for petition SEC-00103 [NIOSH 2012].

#### **2.2.18 Worker Identification and Classification**

From the beginning through 1989, SRS used Payroll Identification numbers to identify and classify workers by type. These numbers consisted of a prefix letter or number for the type of worker payroll (referred to as Roll), an optional two-digit code for craft, and a 4- or 5-digit badge number. Roll prefixes changed over time. Rolls T, 0, and 1 designated DuPont salaried employees. Roll 2 designated local operations hourly workers. Roll 3 designated Wilmington CTWs (engineering and support workers). Roll 4 designated non-DuPont local hourly workers, most of whom were CTWs. Non-DuPont workers other than subcontractor CTWs were tracked in Roll 5. Over time, some non-DuPont CTWs were added to Roll 5. In 1985, Roll 6 was added for CTWs [DuPont 1954b, 1961a; WSRC, no date; DuPont 1988b; WSRC 1990]. The TR and TB prefixes were used for certain temporary workers, and the PS prefix was used for employees of Diversco [DuPont 1983–1985].

CTWs were deployed temporarily but frequently for short periods to perform specific tasks usually pertaining to facility construction and modification, system maintenance, and decontamination. Workers in both categories (DuPont and subcontractor construction) performed these types of jobs. Workers from both categories worked around the site, while production and operation staff normally worked at fixed locations. While workers assigned to Roll 2 were employed directly by DuPont Construction and Bechtel Savannah River, workers in Rolls 4 and 5, or subcontractors, were employed at SRS for periods ranging from a few days to years. Some tasks such as painting were mostly performed by workers in Roll 4 and some in Roll 5, while workers in Roll 2 mostly performed others such as instrument maintenance. Maintenance and decontamination tasks shared common exposure profiles where workers in some of the jobs could be exposed to higher levels of radiation from surface and/or airborne contamination. Lists of occupations and craft trades are provided in ORAUT-TKBS-0003-7, *Internal Dosimetry Co-Exposure Data for the Savannah River Site* [ORAUT 2024b, Section 3.2].

### 2.2.18.1 Health Physics Area Codes

The SRS radiological protection program established and implemented Health Physics Area (HPA) codes to assist in the control and accountability of issuing personnel dosimeters and the collection of bioassay samples. In addition, department codes were used, in some cases, to supplement or complement the HPA codes to better define the origin of the worker [DuPont 1988b; WSRC 1990]. HPA codes initially consisted of two alphanumeric characters beginning in the 1950s as shown on personnel dosimeter reports [DuPont 1958e]. They evolved into the three-character alpha and/or numeric facility codes in use as of 2023. In addition to personnel dosimetry reports by employee number and name, summary personnel dosimetry reports by HPA and department code have been available to SRS management since 1973 [Boiter 1973] to assist in controlling radiation exposure to workers.

HPA and department codes were used to identify workers routinely assigned to an area or location. Some CTWs were at times assigned to a specific area while others were assigned to a common HPA such as Central Shops. SRS used HPA codes for dosimeter assignment codes; workers retrieved and stored their daily dosimeter badges from badge racks in their assigned HPA areas.

HPA codes evolved over time and for various areas. Codes used between 1959 and 2015 are documented in a documented communication with the SRS Dosimetry Records Manager on HPA codes over time [ORAUT 2014]. The communication includes five code tables:

1. "Health Physics-Personnel Meters-IBM-1959,"
2. "Revised Health Physics Area (HPA) Location Codes-January 1973,"
3. "HP Area Codes-August 16, 1990,"
4. "HP Area Codes (1-28-1992)," and
5. "Facility Codes since 2004 from ProRad."

These five tables were combined into one document [ORAUT 2014]. These tables do not include some additions made to HPA codes during the early 1960s.

### 2.2.18.2 Health Physics Department Codes

HPD codes represent numerical designations developed for specific area and/or building locations. SRS used HPD codes to define the functional organization or group to which a worker was assigned and for radiation protection tracking. Both HPD and HPA codes were contained in personnel dosimetry reports.

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## GLOSSARY

### 6% plutonium and 12% plutonium

SRS designations for typical mixtures of plutonium that indicate the mass fraction of  $^{240}\text{Pu}$  in the mixture.

### absorption type

Categories for materials according to their rates of absorption from the respiratory tract to the blood, which replaced the earlier inhalation clearance classes. Defined by the International Commission on Radiological Protection, the absorption types are F: deposited materials that are readily absorbed into blood from the respiratory tract (fast solubilization), M: deposited materials that have intermediate rates of absorption into blood from the respiratory tract (moderate rate of solubilization), and S: deposited materials that are relatively insoluble in the respiratory tract (slow solubilization).

### annual dose equivalent

Dose equivalent received in a year in units of rem or sieverts.

### beta radiation

Charged particle emitted from some radioactive elements with a mass equal to 1/1,837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is a positron.

### byproduct

Material left over from a nuclear, physical, or chemical process designed to produce a particular substance. Examples include the tailings or wastes from the extraction or concentration of uranium or thorium from ore.

### curie

Traditional unit of radioactivity equal to 37 billion ( $3.7 \times 10^{10}$ ) becquerels, which is approximately equal to the activity of 1 gram of pure  $^{226}\text{Ra}$ .

### depleted uranium

Uranium with a percentage of  $^{235}\text{U}$  lower than the 0.7% found in natural uranium. As examples, spent (used) fuel elements, byproduct tails, residues from uranium isotope separation, and some weapons materials contain depleted uranium. Depleted uranium can be blended with highly enriched uranium to make reactor fuel or used as a raw material to produce plutonium. Depleted uranium was used for  $^{239}\text{Pu}$  production at SRS, and the listed isotopic activity fractions were:

<u>Isotope</u>	<u>Activity fraction</u>
$^{234}\text{U}$	0.0840
$^{235}\text{U}$	0.0145
$^{238}\text{U}$	0.9015

### dose equivalent

In units of rem or sieverts, product of absorbed dose in tissue multiplied by a weighting factor and sometimes by other modifying factors to account for the potential for a biological effect from the absorbed dose.

**gamma radiation**

Electromagnetic radiation (photons) that originates in atomic nuclei and accompanies many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Gamma photons are identical to X-ray photons; the difference is that X-rays do not originate in the nucleus.

**highly enriched uranium**

Uranium enriched to at least 20%  $^{235}\text{U}$  for use as fissile material in nuclear weapons components and some reactor fuels. Also called high-enriched uranium. SRS lists the isotopic activity fractions as:

<u>Isotope</u>	<u>Activity fraction</u>
$^{234}\text{U}$	0.9806
$^{235}\text{U}$	0.0194
$^{238}\text{U}$	0.0000

**operating area**

Major SRS operational work areas including fuel fabrication (e.g., 300-Area), reactor operations (e.g., R, P, L, K, and C), chemical separations (F and H), uranium separation (A-Line), plutonium finishing (H-Canyon, B-Line), research and development, transportation, communication, and general site support.

**photon**

Quantum of electromagnetic energy generally regarded as a discrete particle having zero rest mass, no electric charge, and an indefinitely long lifetime. The entire range of electromagnetic radiation that extends in frequency from  $10^{23}$  cycles per second (hertz) to 0 hertz.

**radiation**

Subatomic particles and electromagnetic rays (photons) with kinetic energy that interact with matter through various mechanisms that involve energy transfer.

**radioactivity**

Property possessed by some elements (e.g., uranium) or isotopes (e.g.,  $^{14}\text{C}$ ) of spontaneously emitting energetic particles (electrons or alpha particles) by the disintegration of their atomic nuclei.

**sievert**

International System unit for dose equivalent, which indicates the biological damage caused by radiation. The unit is the radiation value in gray (equal to 1 joule per kilogram) multiplied by a weighting factor for the type of radiation and a weighting factor for the tissue; 1 sievert equals 100 rem.

**thorium extraction (THOREX)**

Adaptation of the plutonium-uranium extraction process to separate  $^{233}\text{U}$  from irradiated thorium.

**X-ray radiation**

Electromagnetic radiation (photons) produced by bombardment of atoms by accelerated particles. X-rays are produced by various mechanisms including bremsstrahlung and electron shell transitions within atoms (characteristic X-rays). Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.

## ATTACHMENT A LIST OF NUCLIDES

Table A-1. Nuclides.<sup>a</sup>

Nuclide	Half-life	Decay mode	Alpha-emitted energy per transformation (MeV)	Electron-emitted energy per transformation (MeV)	Photon-emitted energy per transformation (MeV)	Total emitted energy per transformation (MeV)
Am-241	432.2 yr	Alpha	5.5712	0.0373	0.0293	5.6379
Am-243	7,370 yr	Alpha	5.3583	0.0234	0.0585	5.4402
Am-244	10.1 hr	Beta	Not applicable	0.333	0.8052	1.1382
Ba-130	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Ba-140	12.752 d	Beta	Not applicable	0.3202	0.1826	0.5029
Bk-249	330 d	Beta, alpha	<E-04	0.0324	<E-04	0.0325
Bk-250	3.212 hr	Beta	Not applicable	0.2949	0.8983	1.1932
Bi-209	2.1E+19 yr <sup>b</sup>	Alpha	Not applicable	Not applicable	Not applicable	Not applicable
Cs-134	2.0648 yr	Beta, electron capture	Not applicable	0.1639	1.5551	1.719
Cs-137	30.1671 yr	Beta	Not applicable	0.1884	<E-04	0.1884
Ca-40	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Ca-41	102,000 yr	Electron capture	Not applicable	0.0027	0.0005	0.0032
Ca-48	>5.8E+22 yr <sup>b</sup>	Beta	Not applicable	Not applicable	Not applicable	Not applicable
Cf-250	13.08 yr	Alpha, spontaneous fission	6.1168	0.0103	0.0111	6.2897
Cf-251	900 yr	Alpha	5.8803	0.1705	0.1245	6.1754
Cf-252	2.645 yr	Alpha, spontaneous fission	6.0177	0.2516	0.4572	12.8107
C-14	5,700 yr	Beta	Not applicable	0.0495	Not applicable	0.0495
Ce-139	137.641 d	Electron capture	Not applicable	0.0355	0.1599	0.1954
Ce-141	32.508 d	Beta	Not applicable	0.171	0.0768	0.2478
Ce-143	33.039 hr	Beta	Not applicable	0.4364	0.2796	0.7159
Ce-144	284.91 d	Beta	Not applicable	0.0916	0.0194	0.111
Cr-50	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Cr-51	27.7025 d	Electron capture	Not applicable	0.0038	0.0329	0.0367
Co-58	70.86 d	Electron capture, positron	Not applicable	0.034	0.9749	1.0089
Co-59	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Co-60	5.2713 yr	Beta	Not applicable	0.0969	2.5038	2.6007
Cm-242	162.8 d	Alpha, spontaneous fission	6.2041	0.0096	0.002	6.2156
Cm-244	18.1 yr	Alpha, spontaneous fission	5.8915	0.0079	0.0017	5.9014
Cm-245	8,500 yr	Alpha, spontaneous fission	5.4474	0.0824	0.1084	5.6382
Cm-246	4,760 yr	Alpha, spontaneous fission	5.4654	0.0085	0.005	5.5285
Cm-247	1.56E+7 yr	Alpha	5.0292	0.0114	0.3138	5.3544
Cm-248	348,000 yr	Alpha, spontaneous fission	4.7212	0.7716	1.3127	22.5608
Cm-249	64.15 min	Beta	Not applicable	0.2835	0.02	0.3035

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<b>Nuclide</b>	<b>Half-life</b>	<b>Decay mode</b>	<b>Alpha-emitted energy per transformation (MeV)</b>	<b>Electron-emitted energy per transformation (MeV)</b>	<b>Photon-emitted energy per transformation (MeV)</b>	<b>Total emitted energy per transformation (MeV)</b>
Dy-156	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Dy-158	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Dy-164	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Es-254	275.7 d	Alpha, beta, spontaneous fission	6.5244	0.0727	0.0208	6.6179
Er-162	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Er-170	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Eu-152	13.537 yr	Electron capture, beta, positron	Not applicable	0.1286	1.1759	1.3045
Eu-154	8.593 yr	Beta, electron capture	Not applicable	0.273	1.2493	1.5223
Gd-160	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Hf-177	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Hf-179	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Hf-180	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Ho-165	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Ho-166	26.8 hr	Beta	Not applicable	0.6963	0.0301	0.7264
H-2	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
H-3	12.32 yr	Beta	Not applicable	0.005	Not applicable	0.0057
I-129	1.57E+7 yr	Beta	Not applicable	0.0651	0.0252	0.0902
I-131	8.0207 d	Beta	Not applicable	0.1918	0.3828	0.5746
I-133	20.8 hr	Beta	Not applicable	0.4142	0.612	1.0262
Fe-54	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Fe-58	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Fe-59	44.495 d	Beta	Not applicable	0.1179	1.1883	1.3062
Kr-85	10.756 yr	Beta	Not applicable	0.2507	0.0022	0.2529
La-140	1.6781 d	Beta	Not applicable	0.5346	2.3084	2.8429
Lu-176	3.85E+10 yr	Beta	Not applicable	0.3026	0.4799	0.7825
Mn-54	312.12 d	Electron capture, beta, positron	Not applicable	0.0042	0.836	0.8402
Nd-146	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Nd-150	9.1E+18 yr <sup>b</sup>	Beta	Not applicable	Not applicable	Not applicable	Not applicable
Np-237	2,144,000 yr	Alpha	4.8499	0.0681	0.035	4.9529
Np-238	2.117 d	Beta	Not applicable	0.2519	0.5879	0.8398
Np-239	2.3565 d	Beta	Not applicable	0.2623	0.1846	0.4469
Ni-58	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Ni-62	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable

**ATTACHMENT A  
LIST OF NUCLIDES**

<b>Nuclide</b>	<b>Half-life</b>	<b>Decay mode</b>	<b>Alpha-emitted energy per transformation (MeV)</b>	<b>Electron-emitted energy per transformation (MeV)</b>	<b>Photon-emitted energy per transformation (MeV)</b>	<b>Total emitted energy per transformation (MeV)</b>
Nb-93	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Nb-95	34.991 d	Beta	Not applicable	0.0446	0.7645	0.8091
Os-190	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Os-192	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
P-32	14.263 d	Beta	Not applicable	0.6948	Not applicable	0.6948
Pu-236	2.858 yr	Alpha, spontaneous fission	5.8524	0.0128	0.0022	5.8674
Pu-238	87.7 yr	Alpha, spontaneous fission	5.5803	0.0107	0.0021	5.593
Pu-239	24,110 yr	Alpha	5.2357	0.0075	0.0011	5.2442
Pu-240	6,564 yr	Alpha, spontaneous fission	5.2434	0.0105	0.0019	5.2559
Pu-241	14.35 yr	Beta, alpha	0.0001	0.0052	<E-04	0.0054
Pu-242	375,000 yr	Alpha, spontaneous fission	4.9738	0.009	0.0017	4.9855
Pu-243	4.956 hr	Beta	Not applicable	0.1729	0.0259	0.1988
Po-210	138.376 d	Alpha	5.4075	<E-04	<E-04	5.4075
Pa-233	26.967 d	Beta	Not applicable	0.2151	0.2229	0.438
Pa-234	6.7 hr	Beta	Not applicable	0.4037	1.4718	1.8755
Ra-226	1,600 yr	Alpha	4.8603	0.0039	0.0074	4.8716
Ra-228	5.75 yr	Beta	Not applicable	0.0132	0.0031	0.0163
Re-187	4.12E+10 yr	Beta	Not applicable	0.0006	Not applicable	0.0006
Ru-103	39.26 d	Beta	Not applicable	0.066	0.4962	0.5622
Ru-106	373.59 d	Beta	Not applicable	0.01	Not applicable	0.01
Sm-144	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Sm-150	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Sm-154	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Sc-46	83.79 d	Beta	Not applicable	0.1121	2.0096	2.1217
Si-30	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Ag-107	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Ag-108	2.37 min	Beta, electron capture, positron	Not applicable	0.6071	0.0186	0.6256
Sr-89	50.53 d	Beta	Not applicable	0.5845	<E-04	0.5846
Sr-90	28.79 yr	Beta	Not applicable	0.1957	Not applicable	0.1957
Sr-91	9.63 hr	Beta	Not applicable	0.6549	0.7072	1.3621
Sr-92	2.66 hr	Beta	Not applicable	0.2025	1.3368	1.5393
S-33	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Ta-180	8.152 hr	Electron capture, beta	Not applicable	0.0565	0.0482	0.1048

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<b>Nuclide</b>	<b>Half-life</b>	<b>Decay mode</b>	<b>Alpha-emitted energy per transformation (MeV)</b>	<b>Electron-emitted energy per transformation (MeV)</b>	<b>Photon-emitted energy per transformation (MeV)</b>	<b>Total emitted energy per transformation (MeV)</b>
Ta-181	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Tb-158	180 yr	Electron capture, beta	Not applicable	0.1117	0.8048	0.9165
Tl-204	3.78 yr	Beta, electron capture	Not applicable	0.2372	0.0013	0.2385
Th-228	1.9116 yr	Alpha	5.4956	0.021	0.0036	5.5202
Th-232	1.405E+10 yr	Alpha	4.0688	0.0126	0.0015	4.0829
Th-234	24.1 d	Beta	Not applicable	0.0622	0.0105	0.0728
Tm-168	93.1 d	Electron capture, beta, positron	Not applicable	0.0847	1.2432	1.3279
Tm-170	128.6 d	Beta, electron capture	Not applicable	0.328	0.0041	0.3321
Tm-171	1.92 yr	Beta	Not applicable	0.0255	0.0006	0.0261
Sn-118	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Sn-121	27.03 hr	Beta	Not applicable	0.1156	Not applicable	0.1156
Sn-124	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
W-186	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
U-232	68.9 yr	Alpha	5.3948	0.0164	0.0023	5.4135
U-233	159,200 yr	Alpha	4.9013	0.0059	0.0013	4.9085
U-234	245,500 yr	Alpha	4.843	0.0137	0.002	4.8587
U-235	7.04E+8 yr	Alpha	4.4693	0.053	0.1669	4.6891
U-236	2.342E+7 yr	Alpha	4.5592	0.0114	0.0018	4.5723
U-238	4.468E+9 yr	Alpha, spontaneous fission	4.2584	0.0092	0.0014	4.2691
Xe-133	5.243 d	Beta	Not applicable	0.1379	0.0474	0.1854
Yb-168	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Yb-176	Stable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
Y-91	58.51 d	Beta	Not applicable	0.6032	0.0031	0.6063
Zn-65	244.06 d	Electron capture, positron	Not applicable	0.0069	0.5819	0.5888
Zr-95	64.032 d	Beta	Not applicable	0.1185	0.7321	0.8506
Zr-96	2.35E+19 yr <sup>b</sup>	Beta	Not applicable	Not applicable	Not applicable	Not applicable

a. Sources: Brookhaven Science Associates [no date], International Commission on Radiological Protection [2008].

b. Extremely long-lived with very low radioactivity. These are considered stable for all practical purposes.