

3. PROCESSES RESULTING IN WASTE EFFLUENTS

NUCLEAR PRODUCTION REACTORS (100 AREAS)

Three production reactors (P, K, and C) of the five originally constructed are currently operating at SRP. R and L reactors were shut down and placed in standby condition in 1964 and 1968, respectively. The primary products of the SRP reactors are plutonium (^{239}Pu) and tritium (T or ^3H). These are produced by absorption of neutrons in uranium (^{238}U) and lithium (^6Li), respectively. The versatility of the reactors has also led to the production of other nuclear materials.

Radioactivity releases to the environs in 1975 from reactor operations included: 159,000 curies of tritium, 65,000 curies of argon-41, 4,000 curies of krypton and xenon, and 39 curies of carbon-14 to the atmosphere; 42,000 curies of tritium and 1.4 curies of fission products and activation products to plant streams; and 22,000 curies of tritium and 0.3 curie of fission and activation products to earthen basins and cooling water impoundments. Heat in effluent cooling water is largely dissipated in onsite natural watercourses, swamps, and cooling water impoundments.

Description of Facilities

Each production reactor and its auxiliaries are housed in a reactor building that incorporates heavy concrete shielding to protect personnel from radiation. The reactor uses heavy water (D_2O) as a neutron moderator and as a recirculating primary coolant to remove heat from the nuclear fission process. Water pumped from the Savannah River or Par Pond, a cooling water impoundment, serves as a secondary coolant. The following paragraphs describe those components of a reactor system that are involved in the formation, transport, and release of radioactive materials and heat to the environs. Figure II-4 is a schematic diagram of the reactor process systems.

Hydraulic Systems

Primary Coolant. The D_2O primary coolant is circulated through the reactor by six parallel pumping systems. In each system, coolant is pumped from an outlet nozzle at the bottom of the reactor through two parallel heat exchangers into one of the inlet nozzles in the water plenum. From the plenum, the D_2O flows down through the fuel and target assemblies and then into the bulk moderator space in the reactor tank, from which it exits through the outlet nozzles.

A side stream from one of the circulating systems is supplied to the purification facility. This facility consists of prefilters, deionizers, afterfilters, evaporators, and distillation towers. Prefilters are designed to protect the deionizers from pluggage by particulate material, and afterfilters prevent loss of deionizer resin particles to the circulating coolant. The filters are replaced remotely when pressure drop indicates pluggage. Deionizers maintain ionic purity of the coolant by removing both normal and radioactive chemicals that arise from corrosion and chemical action.

Exposure of the coolant to humid air creates light water impurity (H_2O) in the heavy water (D_2O). To maintain coolant purity, a side stream from the deionizer effluent is fed to the distillation system for the separation of H_2O from the D_2O . Overhead streams containing primarily H_2O are drummed for further processing, and bottom streams containing the purified D_2O are returned to the coolant system. Evaporators are used to remove particulate and dissolved materials, both from drummed D_2O and from the distillation system. The condensate may be returned to the reactor circulating system or, if necessary, sent to the 400-D Area rework unit for further upgrading of D_2O purity.

Secondary Coolant. River water is pumped to 25,000,000-gallon cooling water reservoirs in P, K, and C areas. Water is also pumped to P Area from Par Pond (a 2700-acre reservoir designed to supplement the cooling water supply by evaporative cooling and reuse of effluent water). The water is then pumped to the reactor building to cool the D_2O in the main heat exchangers and for other area uses. The effluent cooling water then flows back to the river (from K and C) or to Par Pond (from P).

Gas Systems

Blanket Gas. A blanket of helium is maintained over the surface of the D_2O moderator to aid in the collection and treatment of gases produced by radiolytic decomposition of D_2O and corrosion of aluminum. The helium is passed in a closed circuit through the reactor gas plenum and then through a gas purification system to recombine deuterium (D_2) and oxygen evolved by radiolysis of the moderator and D_2 resulting from aluminum corrosion. Oxygen is added to the blanket gas when needed to maintain the D_2 concentration well below the flammability limit.

Building Air Flow. Clean outside air is drawn through the reactor building as shown in Figure II-5. The normal flow of 128,000 ft^3/min is the design value and is controlled over a range

of 10 to 20%. The distribution of flow through the three process areas (purification area, pump and heat exchanger area, and reactor room) is also controlled to assure that air flows from uncontaminated to potentially contaminated or contaminated areas before discharge through the stack.

During operation, all exhaust air to the stack passes through an activity collection (confinement) system designed to collect and retain radioactive particulates and iodine vapors. The system is designed primarily to limit the consequences of unlikely reactor accidents and is online at all times. It consists of (1) moisture separators (crimped wire mesh and teflon fibers intended to prevent entrained water particles from collecting on the particulate filters), (2) high-efficiency particulate filters, and (3) carbon absorbers designed to remove elemental iodine vapor and gaseous iodine compounds.

Component Handling

Components from the 300-M Area fuel and target fabrication facilities are received in the assembly area of the reactor building. Necessary assembly, testing, and inspection are done here, and the assemblies are then transferred to hangers for storage and subsequent charging to the reactor. Fresh assemblies are charged into the reactor, and irradiated assemblies are discharged to the fuel and target storage basin by two remotely controlled machines. The discharge machines are equipped with water cooling facilities to prevent an irradiated assembly from overheating if a machine becomes temporarily inoperative, and to flush discharged assemblies to reduce tritium carryover to the basin.

Fuel and Target Storage Basin

Irradiated fuel and targets are stored underwater in a basin which is 30 ft deep. Sections of the basin can be isolated from other sections by the use of temporary barriers placed in the openings in the concrete wall and by gates with inflatable seals. When a storage section is isolated, heat from the stored fuel and targets is removed by circulating the basin water through a heat exchanger system.

The basin water is recirculated through a sand filter to provide adequate visibility without a continuous fresh water purge. Deionizers, mounted on portable trailer units, are used to reduce radioactivity levels in the storage basin water. Six such deionizers are available and are placed in service as needed, depending on the amount and type of activity in basin water.

They are normally operated as mixed beds with 60 ft³ of resin in the H⁺-OH⁻ form. These are regenerated (in the separations area, as described on page II-36) [K-10] with nitric acid and sodium hydroxide.

Fuel and Targets

Various fuel and target assemblies, all clad with aluminum, have been irradiated in SRP reactors; typical fuel and target assemblies and their irradiation products are listed below.

Typical SRP Fuel and Target Material and Products

<i>Fuel/Target Material</i>	<i>Product</i>
Natural U	²³⁹ Pu
Depleted U (0.14% and 0.20% ²³⁵ U)	²³⁹ Pu
Slightly Enriched U (0.95% ²³⁵ U)	²³⁹ Pu
Enriched U	Neutrons for Various Target Absorptions
Recycled Enriched U	²³⁶ U, ²³⁷ Np
²³⁷ Np	²³⁸ Pu
²³⁹ Pu	²⁴⁰ Pu, ²⁴² Pu, ²⁴³ Am, ²⁴⁴ Cm
²⁴² Pu	²⁴⁴ Cm, ²⁵² Cf
²⁴¹ Am	²³⁸ Pu
²³² Th	²³³ U
²⁰⁹ Bi	²¹⁰ Po
²⁴³ Am- ²⁴⁴ Cm	²⁵² Cf
⁶ Li	³ H
⁵⁹ Co	⁶⁰ Co

Sources of Radioactive Release Within Facilities

Components, structural materials, heavy and light water systems, and gas systems have the potential of becoming radioactive from neutron activation. In addition, fuel and target materials build up concentrations of fission products and of desired radioactive products (and intermediates in their production).

Enriched Uranium-Aluminum Alloy Fuel

During irradiation of reactor charges containing enriched uranium fuel assemblies, fission products are released to the D₂O coolant from occasional defects in the fuel cladding. These defects are normally so small that their effects on assembly flow and temperature cannot be detected. In recent years, activity releases due to manufacturing defects in enriched uranium charges have been reduced by an extensive program to minimize the formation of defects in the casting process.

Depleted Uranium Metal Targets

Depleted uranium target element failures have been rare; when they occur, they proceed by first a breach of the cladding (e.g., pinhole leaks) and then a reaction of the hot uranium metal with the D₂O coolant that causes a distortion of the fuel and cladding. This distortion is sufficient to be detected by flow and temperature monitoring instrumentation. Thus, the location of the defective target element is identified before significant activity is released to the coolant. The reactor is shut down, and the assembly containing the failure is discharged to a container specially designed to provide cooling and contain leaking radioactivity. The container is stored underwater in the disassembly basin until it can be shipped to a special facility in the separations area for disassembly, examination, and processing. Prompt shutdown minimizes the amount of core metal and fission products released to the reactor system. Nongaseous radioactivity released from the failure before discharge is confined to the primary coolant system in the reactor and is removed by the filters and deionizers in the purification area.

Aluminum and Stainless Steel Corrosion

Trace nuclides resulting from aluminum corrosion are ⁵⁹Fe, ⁶⁰Co, ⁵⁶Mn, ⁶⁵Zn, and ⁵⁸Co. These are activation products of impurities and alloying metals. Nuclides associated with the irradiation of stainless steel reactor components are ⁵¹Cr, ⁶⁰Co, and ⁵⁹Fe. General stainless steel corrosion rates are low, and several instances of abnormal ⁵¹Cr concentrations have been traced to unusual wear of components, especially heat exchanger core rods.

Releases of stainless steel and aluminum activation products are minimized as a result of efforts to minimize corrosion. Acidification of the D₂O coolant with nitric acid to a pH of 5.0 to 5.2 results in the lowest attainable aluminum corrosion with an acceptably low general stainless corrosion rate. Practices

effective in minimizing stainless steel corrosion include avoiding large fluctuations of pH, preventing accumulation of chlorides from all potential sources such as makeup D₂O, and controlling dissolved oxygen at the lowest practicable level.

Ion Exchange Resin Degradation

The deionizers in the D₂O coolant purification system concentrate sufficient radioactivity that the resultant radiation dose causes degradation of the ion exchange resin. Degradation is evidenced by loss of exchange capacity, carbon dioxide in the blanket gas from radiolytic decomposition of the organics in the resin, and loss of organic sulfonate fragments of the cation resin to the coolant system. The amount of sulfur released is minimized by sorption on the anion resin, except when the anion capacity is nearly exhausted through normal depletion or radiation damage. Sulfur deposited in the neutron flux is activated and results in detectable quantities of ³⁵S and ³²P in the coolant.

Activation of Miscellaneous Impurities

Any materials added unintentionally to the D₂O coolant could result in the production of radioactive nuclides if their residence time in the reactor is sufficient. Such impurities are minimized by strict purity requirements on makeup D₂O, routing of this makeup water through the deionizers before mixing with the bulk coolant, and procedural control of chemical additions (e.g., nitric acid) to the coolant.

An example of impurity activity was the presence of ³²P in K-Area D₂O coolant starting in 1965. Phosphoric acid was used, experimentally, in place of oxalic acid for decontamination of a process heat exchanger before routine overhaul. Sufficient PO₄³⁻, probably as a deposited aluminum compound, remained in the heat exchanger tubes so that when the exchanger was returned to service, gradual dissolution of the compound resulted in detectable quantities (50-100 ppb) of PO₄³⁻ in the coolant. Neutron activation of the PO₄³⁻ resulted in higher ³²P concentrations than from the activation of resin fragments.

Argon Activation

There is an air space between the reactor tank and side shield systems. Neutron activation of the natural argon gas in this air produces ⁴¹Ar (2-hr half-life), which is subsequently

released through leakage paths to the process area ventilation systems. Activation of argon in the air around the reactors is reduced by absorbing more neutrons in "blanket" assemblies around the reactor cores.

Activation of the Heavy Water Coolant

Because the D₂O coolant is exposed to neutron fluxes, the tritium content of the D₂O is increased by neutron activation of the deuterium atoms. The buildup rate of tritium is affected by mean residence time of D₂O in the reactor, leakage-makeup rates of D₂O, tritium content of makeup, and variations in neutron flux.

Carbon-14 is produced from (n,α) reactions with naturally occurring ¹⁷O present in the coolant. Most of the ¹⁴C in coolant is removed, as carbonates, by an ion-exchange purification system. This reaction is the main source of ¹⁴C releases from reactor operations. Carbon-14 is also formed in heavy water coolant by an (n,p) reaction with ¹⁴N present in the form of dissolved gas and in the nitric acid used for coolant quality control.

Transport to Point of Release from Facilities

Gaseous Pathways

Fission product noble gases (isotopes of Kr and Xe), released from either the small defects in enriched uranium fuel or failure of depleted uranium targets, diffuse to the helium blanket gas system, and then, because of leaks in this slightly pressurized system, enter the process ventilation system and are released from the 200-ft stack. Argon-41 from the air space surrounding the reactor tank is also released from the stack.

Tritium (as DTO) is released to the ventilation system by evaporation of D₂O exposed to the air flowing through the process areas. During reactor operation, D₂O is evaporated from small leaks in flange joints and valve packing. During shutdown, infrequent D₂O spills, maintenance work on process water equipment, and the charge-discharge operation result in tritium releases to the ventilation system.

Aqueous Pathways

Fuel and Target Storage Basin. When fuel, targets, or other components are discharged to the storage basin, D₂O from the reactor tank is also transferred by two mechanisms. The first mechanism, physical transfer, involves adherence to the component

surfaces and entrapment in fittings. The second mechanism involves chemical transfer through hydrogen isotopes contained in the hydrated oxide film on aluminum surfaces. The tritiated heavy water exchanges rapidly with the light water of the basin and is uniformly distributed throughout the isolated fuel storage section.

The machines which remove fuel and targets from the reactor and discharge them to the storage basin are provided with a light water flushing system which removes 30 to 50% of the adhering D₂O and associated tritium during the discharge operation. This flush water is collected and is sent to the 400-D rework unit for D₂O recovery.

Fission products and activation products also concentrate in the aluminum oxide on component surfaces and are transferred to the storage basin with the discharged component. The fission and activation products then dissolve, diffuse, or exchange in the basin water and are collected eventually on deionizers or filters. Unlike tritium, these nuclides require several days to reach maximum concentrations in the basin. In most cases, the transfer by the oxide film is the dominant mechanism for release to the basin rather than continued corrosion of exposed fuel core material.

Process Heat Exchanger Leakage. Leakage of D₂O to the H₂O cooling water with consequent release of tritium (and other nuclides) to the effluent is approximately one-fourth of the total tritium releases from the reactor areas. Effluent cooling water from each process heat exchanger is continuously monitored by in-line gamma detectors. These are capable of detecting heavy water leakage from a single exchanger of about 0.3 lb/hr. This monitoring is supplemented by more sensitive tritium analyses on samples drawn once each week during operation.

Atmospheric Releases

A confinement system designed to contain radioactive particles and iodine that may be released to the ventilating air in accidents is provided for each reactor area. In normal operation, the particulate filters and carbon absorbers reduce radioactive releases, but the nuclides removed are only minor constituents of the stack air activity. The major radioactive constituents, tritiated water vapor and the noble gases ⁴¹Ar and isotopes of Kr-Xe, are unaffected by the confinement system.

The stack air is continuously monitored by drawing a small sample stream through a Kanne chamber. The Kanne signal is proportional to the sum of the tritium, ^{41}Ar , and Kr-Xe present and serves as a qualitative detector that an activity increase is occurring. Further investigation is required to determine the nature and source of the activity. Continuous monitors for noble gas activity are provided in all three reactor areas.

Tritium

The release of tritiated water vapor from the reactor stacks is calculated daily from tritium concentrations in samples of condensate from stack air and the humidity of the air.

Argon-41

Measurements of ^{41}Ar releases by gamma spectrometry were started in 1971. The values for previous years (Appendix A) were calculated from current releases and knowledge of the differences in neutron flux in the air space, which, in turn, was calculated from reactor loading patterns, charge designs, and power histories.

Kr-Xe

Measurements of fission product noble gases released from the reactor stacks were also started in 1971. Calculations of releases for previous years (Appendix A) are based on knowledge of the fission product nuclides in the reactor D_2O coolant.

Particulates and Iodine

Stack air sampling for particulate activity and radioiodine started in late 1960. A continuous stream of stack air is drawn through filter paper and charcoal absorbers, which are analyzed weekly.

Carbon-14

Calculations of ^{14}C releases are based on experimental stack monitoring results.

Aqueous Releases

Fuel and Target Storage Basin

A section of the fuel and target storage basin is isolated from other sections of the basin before each reactor discharge, and the water is recirculated through ion exchange resin during and following discharge using mobile deionizers mounted on trailers. When depleted, the resin beds are regenerated, and the radioactive wastes are processed in the separations areas. The regenerated units are returned to the reactor area for continued use. When the tritium content of the basin water has built up beyond a procedural control point (currently 0.2 to 0.4 $\mu\text{Ci/ml}$) and other activities are below their procedural limits, the deionizer effluent is directed to a plant stream and fresh water is added to the basin at the same rate. This purging is continued until a desired reduction in basin tritium content is obtained, thus reducing the airborne tritium concentration in working areas.

If deionizer effluent concentrations reach control values, the deionizer is either returned to the recirculation mode or taken off-line for regeneration. The current control values are 1.1×10^{-5} $\mu\text{Ci/ml}$ for ^{131}I and $^{134},^{137}\text{Cs}$, 2.3×10^{-5} $\mu\text{Ci/ml}$ for ^{51}Cr , and 3.4×10^{-5} $\mu\text{Ci/ml}$ for $^{95}\text{Zr-Nb}$ and $^{141},^{144}\text{Ce}$. These values may be reduced when tritium concentrations are sufficiently low that the purge may be interrupted without causing undue exposure of personnel to airborne tritium.

The primary factors governing aqueous releases are the annual operating guides for quantities of specific nuclides. A running account of the curies released is maintained during the purge and further limitations may be placed on effluent concentrations based on comparison with the operating guides. These estimated releases (including tritium) are verified by separate effluent monitoring described in Appendix E.

In K Area, the storage basin purges are directed to a 50,000,000-gallon earthen confinement basin, so the only activity released directly to plant streams comes from miscellaneous purges and leaks to cooling water, which is discharged to Pen Branch. This use of the confinement basin is a continuation of the practice begun in 1965 to dispose of ^{32}P activity (14-day half-life) through decay rather than releasing it directly to plant streams.

In P and C Areas, the purges are directed to plant streams. In C Area, activity released via the storage basin purges mixes with cooling water from the storage basin heat exchangers and distillation system cooling water. This stream is further diluted by the main heat exchanger cooling water and is discharged to

Four Mile Creek. The P-Area storage basin purges are routed to Steel Creek and mix with cooling water from the storage basin heat exchangers and the distillation system. This is the only dilution of purge water or leakage from the P-Area storage basin because the main heat exchanger cooling water is not discharged to a plant stream, but is recycled via the Par Pond reservoir.

Tritium. Tritium, transferred to the fuel storage basin with heavy water on the surfaces of discharged components, is released to plant streams or to the K-Area confinement basin when the storage basin water is purged. The purge is necessary to prevent exposure to operating personnel from airborne tritium. In 1975, purges from the reactor area basins included 30,000 Ci of tritium.

Fission Products and Activation Products. In 1975, about 1.4 Ci of non-tritium activities were released to the effluent streams from the three operating reactor areas. This release included 0.6 Ci of isotopes with radioactive half-lives less than one year; the longer-lived isotopes included 0.44 Ci ^{137}Cs , 0.11 Ci ^{90}Sr , 0.01 Ci ^{60}Co , 0.18 Ci ^{134}Cs , and 0.03 Ci ^{147}Pm .

Disposal of Miscellaneous Contaminated Water

Liquid wastes, primarily collected in sumps, tanks, and drums with D_2O content too low for economical recovery, are analyzed before disposal to determine the proper disposal point according to current operating procedures. The disposition depends on beta, gamma, and alpha activities and chemical content (conductivity, pH) and may be to plant streams, fuel and target storage basins, oil and chemical pits, or the separations area waste system. In the past, earthen seepage basins have been used for this material but are now inactive.

The principles and effects of seepage basin usage are given in Sections III and V. Descriptions of the previous use of reactor area seepage basins and the current use of the oil and chemical pit and the 50-million-gallon basin can be found in DP-1349, *Control and Treatment of Radioactive Liquid Waste Effluents at the Savannah River Plant*.¹⁰ The effluent monitoring program for aqueous releases to streams and basins from the reactor areas is given in Appendix E; quantities of radioactivity released since startup are given in Appendix A.

Releases of Nonradioactive Materials

About 1500 lb/yr of sodium chromate from chilled water systems leakage and about 350 lb/yr of phosphates (NaH_2PO_4) from heat exchanger inhibitor leakage reach plant streams. About 2000 lb/yr of boron reaches the effluents (in P Area the canal to Par Pond) from leakage of borated emergency cooling water.

A biodegradable inhibitor is being tested in the C reactor chilled water system as a replacement for sodium chromate.

Thermal Releases

Temperatures in the effluent cooling water are substantially higher than natural temperatures. This heat is largely dissipated in onsite natural watercourses and swamps (K and C reactors) and in cooling water impoundments (P reactor). Thermal effects on the river are within South Carolina standards and are discussed in Section III.

Disposal of Confined Activity and Other Materials

Efforts to minimize releases to the environs result in concentration of activity in various removal systems. The further disposal of this activity is described in the following paragraphs for the major systems involved.

Normal Processing

Most of the irradiation products formed in fuel and target assemblies are recovered in the separations area processes (or in some cases, shipped directly off-plant) after cooling in the storage basin to allow decay of short-lived fission products.

Miscellaneous Irradiated Material

Discharged components such as fuel and target housings and scrap metal from disassembly operations are sent to the burial ground for disposal.

Purification Equipment

The units from the D_2O coolant purification system are also shipped to the burial ground. Filters are buried in sealed concrete vessels, evaporators and deionizers are buried directly,

and the resin is encased in the stainless steel deionizer vessel. Before shipment to the burial ground, the D₂O in the deionizers is recovered by light water displacement. About 3 or 4 deionizers are used in each reactor area each year.

Storage Basin Sand Filter

Sludge from the sand filter settler is periodically pumped to a tank truck and transported for evaporation and storage in the separations area waste tanks. A program of cleaning the fuel storage basin floors is in progress; sludge is pumped to the settler for storage and is then sent to the separations area.

Heavy Water

The D₂O recovered from various systems designed to minimize heavy water losses or tritium releases is processed, if necessary, in reactor area deionization or evaporation facilities and then shipped to the heavy water plant recovery unit (420-D) for upgrading to reactor quality. The major sources of this water are:

- Fuel element flush.
- Deionizer dedeuterization.
- Distillation overheads.

Portable Deionizers

The deionizer units used to remove activity from the fuel storage basins are sent to the resin regeneration facility in the separations area for treatment and are returned to the reactor areas for continued use of the same resin. The next section of this report discusses the further disposal of the activity.

Lubricating Oil

About 2500 gal/yr of lubricating oil collected from leakage around bearings and piping is drummed and stored in the burial ground.

CHEMICAL SEPARATIONS FACILITIES (200 AREAS)

The chemical separations facilities consist of two main operating areas, 200-F and 200-H. Each main area has a large shielded "canyon" building for processing irradiated materials (fuel and/or targets), a waste concentration and storage system, seepage basins, powerhouses, and service facilities. In addition, F-Area contains the main analytical laboratory, the plutonium metallurgical building, and the Plutonium Fuel Form facility (PuFF) (presently under construction and scheduled for operation in 1978). H Area contains the tritium process buildings, the Receiving Basin for Offsite Fuels (RBOF), and the Resin Regeneration Facility (RRF). These areas are the major sources of radioactive liquid and solid wastes that are stored onsite.

Radioactive materials can leave these areas in several ways: in exhaust ventilation air from buildings, in off-gases from operations, in water that is discharged directly to natural waterways, and in radioactive liquid that migrates from seepage basins through the soil to a natural waterway.

In 1975, releases of radioactivity from the 200-F and 200-H facilities included 325,000 Ci of tritium,* 520,000 Ci of ^{85}Kr , 27 Ci of ^{14}C , 0.24 Ci of iodine (^{131}I and ^{129}I) and 0.0025 Ci of Pu to the atmosphere; 9000 Ci of tritium, 1.4 Ci of beta-gamma, and 0.007 Ci of alpha activity to plant streams; and 14,000 Ci of tritium, 26 Ci of beta-gamma, and 0.2 Ci of Pu activity to seepage basins.

Description of Processes and Facilities

In the chemical processing plants, the first step is to dissolve the irradiated components, which liberates volatile fission products and generates solutions with high concentrations of radioactivity. The initial separation yields 1) solutions of plutonium, uranium, or neptunium product, and 2) a high-heat liquid waste, containing the nonvolatile fission products. After the fission products are removed sufficiently from the product solutions, further processing can be done in unshielded areas, where the product elements may be converted from solution form to solid. The final steps involve handling powders of both ^{238}Pu and ^{239}Pu , which require additional precautions to avoid airborne releases. Figure II-6 is a schematic diagram of the separations process systems.

* Includes December 1975 release described in Appendix J.

²³⁹Pu Processes

In the 221-F canyon building, the Purex solvent extraction process is used to recover ²³⁹Pu from irradiated uranium, and the plutonium is converted to metal. Special or subsidiary programs include processing of highly irradiated plutonium and recovery of neptunium.

Processing begins with dissolution of the irradiated components. With the normal aluminum-clad, metallic uranium fuel, the aluminum cladding is dissolved in a sodium hydroxide-sodium nitrate solution, with evolution of ammonia and hydrogen. After the decladding solution with contained aluminum is removed, the uranium is dissolved in nitric acid. Noble gas fission products and some iodine are evolved during dissolving; these gases are routed through a vessel containing silver nitrate-coated sorbent materials (to reduce iodine emissions), a filter, and an acid recovery unit that converts some of the nitrogen oxides in the off-gas to reusable nitric acid, reducing nitrogen oxide emissions. The uranium solution is clarified, and some of the fission products are removed by adsorption on a precipitate of MnO₂.

Carbon-14 is produced during irradiation of enriched fuel and other fuel and core materials by an (n, p) reaction with ¹⁴N impurities. Most of the ¹⁴C so produced remains in the irradiated components until it is dissolved in the chemical separations areas, where most is evolved as ¹⁴CO and ¹⁴CO₂ and is exhausted to the atmosphere from 200-ft-high exhaust stacks.

In the first solvent extraction cycle, uranium and plutonium are extracted away from the remaining fission products with tributyl phosphate in a hydrocarbon diluent and then separated from each other. The uranium and plutonium are further purified with separate second solvent extraction cycles. The final uranium product solution is concentrated and sent outside the canyon building for subsequent processing in Building 221-1F. The plutonium product solution is transferred to a separate area within the canyon building where the plutonium is concentrated by ion exchange, precipitated as the fluoride, and, with some further treatments, reduced to the metal.

In the canyon building, neptunium and residual plutonium are recovered by ion exchange from the fission product waste solution from solvent extraction. Neptunium is further purified by ion exchange, and the final product solution is sent to the H-Area neptunium line. The plutonium is sent to the plutonium processing system. Also, irradiated plutonium fuel and target assemblies may be processed in special programs in the main solvent extraction system and in the plutonium finishing system.

Building 211-F. These operating areas constitute the supporting facilities to the 221-F canyon building. Cold feed solutions are prepared, uranium product solution is concentrated and converted to uranium oxide powder by denitration, nitrogen oxides from canyon dissolving and the denitration step are recovered as nitric acid, and miscellaneous radioactive wastes are handled, including Savannah River Laboratory wastes that are unloaded from the SRL waste trailer at a special station.

²³⁵U Processes

The 221-H canyon building is structurally very similar to the 221-F canyon building and originally was operated as a Purex solvent extraction facility. It was modified in 1959, and since then it has been devoted primarily to recovery of enriched uranium from SRP reactor fuel. Fuels from nondemonstration commercial power reactors such as PWRs and BWRs are not processed at the Savannah River Plant. Further modifications to the canyon added a small dissolver and ion exchange equipment for the recovery of ²³⁸Pu from irradiated neptunium. An electrolytic dissolver was installed for the handling of cermets and fuels clad with stainless steel or zirconium. Special processing programs have included recovery of thorium, ²³³U, and various mixtures of plutonium isotopes.

The solvent extraction operations are similar to those of the Purex system, with modifications in the chemical processes as required by differences in the fuels. The standard enriched uranium fuel is a uranium-aluminum alloy, clad in aluminum, and the whole fuel piece is dissolved in nitric acid catalyzed by mercuric nitrate, without a separate decladding step. As in F Area, the off-gases go through a silver nitrate reactor to reduce iodine emissions. The dissolved fuel is clarified, and the adjusted solution is fed to the solvent extraction system. Uranium and neptunium are extracted from the fission products and separated from each other in the first cycle; then each goes through two separate cycles of solvent extraction for further purification. The uranium product solution is not concentrated further, but is transferred out of the building, loaded into tank trucks, and sent offsite for further processing. The neptunium product solution is transferred to a separate area in the building for subsequent purification.

In a separate process system in the shielded canyon, irradiated neptunium targets are dissolved in nitric acid, and ^{238}Pu and neptunium are separated from fission products and each other by a series of anion exchange resin columns. The product solutions of plutonium and neptunium are transferred to the finishing area, where the two are concentrated, precipitated as oxalate, and calcined to oxides. The plutonium oxide is packaged for shipment offsite, and the neptunium oxide is sent to the metallurgical building (235-F) for refabrication into a billet to be made into reactor target elements.

In addition to these main process efforts, various parts of the system have been used for special programs. For example, highly irradiated plutonium has been processed in both the solvent extraction and canyon anion exchange systems and finally produced as oxide from the finishing area. A plutonium isotope mixture containing appreciable ^{238}Pu routinely has been recovered by solvent extraction from the fission product waste stream from enriched uranium processing. Irradiated thorium has been processed in the solvent extraction system to recover ^{233}U . A purified, concentrated thorium nitrate solution has been shipped offsite for reuse, and the uranium fraction from solvent extraction has been concentrated by cation exchange in the finishing area, precipitated as ammonium diuranate, calcined to the oxide, and shipped offsite.

Tritium Facilities

The tritium facilities (232-, 234-, and 238-H) are a complex of buildings in which tritium is separated from irradiated lithium-aluminum targets, further purified, and packaged. Operations are carried out in well-ventilated areas and process cabinets in which air movement sweeps any unavoidable tritium releases out the stacks.

In the processing system, the lithium-aluminum targets are loaded into furnaces, and the gases are extracted under heat and vacuum. The impure gas contains tritium, ^4He which has formed with tritium in neutron capture by ^6Li , ^3He from tritium decay, and ^1H and ^2H from the hydrated aluminum oxide surface of the targets. The hydrogen isotopes are first separated from the helium isotopes, and then tritium is separated from protium (^1H) and deuterium (^2H). Solid wastes are packaged and shipped to the burial ground.

Receiving Basin for Offsite Fuels (RBOF) and Resin Regeneration Facility (RRF)

Buildings 244- and 245-H house a versatile, multipurpose operation in which fuels from offsite reactors are received and either stored or trimmed and packaged for processing. Special SRP fuels also may be cut and sampled, or trimmed for processing. The resin regeneration area receives and regenerates ion exchange beds and filters from the reactor areas where they have been used to remove activity from the fuel and target storage basins.

Metallurgical Building

Building 235-F includes facilities for fabricating components for uranium, neptunium, and plutonium reactor elements. One process involves blending oxides of these materials with aluminum powder and pressing the mixture into slugs or billet components for extrusion. In another process, these materials are melted with aluminum in a furnace, and the resulting aluminum alloys are machined and assembled into billets for extrusion. The PuFF facility presently under construction will be used to manufacture ^{238}Pu heat source fuel forms (startup is anticipated in 1978).

Analytical Laboratory

The 772-F analytical laboratory provides radioanalytical services to the entire plant (except the Savannah River Laboratory); the bulk of the sample load comes from the two separations areas. Samples include fuel solutions, fission product waste, concentrated plutonium product solution, and plutonium powders. Also, some tritium is brought into the laboratory in reactor D_2O coolant samples. The total amount of radioactivity handled is small compared with that in the process buildings.

Atmospheric Releases

Ventilation and Filtration Systems

Ventilation air flows in series from clean areas to potentially or actually contaminated areas before discharge through filters to a stack. The air is filtered before discharge with one, two, or three stages of filtration to remove airborne particulate radioactivity. Emergency power and spare ventilation capacity are provided on air streams that normally contain particulate activity so that this air will always move through filters before discharge. Many stacks of varying heights serve the buildings of the separations areas; they provide for dispersion and dilution of any released activity before return to the ground,

and they provide some emergency natural draft in case of total power failure. Monitoring of stack exhausts to inventory releases and to detect abnormal conditions is described in Appendix E.

Canyon Building Ventilation Systems. In Building 221-F, exhaust air from the several sections and processes in the building passes through filtration and activity-removal systems before being combined into a common effluent of 225,000 cfm from a 200-ft-tall stack (Figure II-7). The main contributors to the activity in the stack effluent are the main process canyon air, the vessel vent system that collects off-gases from the process solutions, and the air from the plutonium powder-handling cabinets. The vessel vent air and the plutonium cabinet air are each filtered independently, and then combined with the canyon air for passage through deep-bed sand filters. The sand filter effluent of about 120,000 ft³/min then combines with the air from other building areas, from the dissolver off-gas, and from the acid recovery unit for exhaust from the stack.

Two small pipes are attached to the main stack and discharge at the same height; these pipes allow the ammonia-bearing off-gases from the two fuel dissolvers during decladding operations to be diverted from the main stack (after passing through an iodine absorber and filter) to prevent the formation of ammonium nitrate in the stack. Additional air is exhausted from the plutonium processing area (exclusive of the powder-handling cabinets) via separate filters and a stack 160 ft above grade mounted on top of the canyon building.

In Building 221-H, the general pattern of ventilation is similar to that in 221-F, with a few differences (Figure II-7). As in F, the exhaust air from the main canyons and the vessel vent system exhausts through a sand filter (a second sand filter is under construction), and air from the central personnel areas is filtered separately. The finishing lines for ²³⁷Np and ²³⁸Pu also are filtered separately and do not exhaust through the sand filter, unlike the 221-F system. The separate small stacks for the dissolvers are not required because there is no aluminum decladding with sodium hydroxide to generate ammonia. The combined air streams of about 220,000 ft³/min are discharged through a 200 ft stack.

Filtration System. The air systems use high-efficiency particulate air (HEPA) filters, packed glass fiber filters, and two deep-bed sand filters that serve the two main canyon buildings (221-F and H). HEPA filters are designed to remove 99.97% of all particles down to 0.3 μm in diameter. Components of the filter unit consist of a frame, filter media (glass fiber or glass asbestos), separators (aluminum, asbestos), adhesive to seal the

media to the frame, and gaskets. In manufacturing the filter unit, the filter media is pleated to increase the surface area. During pleating, a separator is inserted between each fold to separate each lap of the filter media and to allow air to flow along the channels formed by the corrugations and through the media. Continuing attention is given to maintaining adequate filtration efficiency. It generally is not feasible to measure all filter efficiencies on a continuous basis, so normally the activity releases from stacks are monitored, and when releases increase, the efficiencies are studied. Periodic tests of accessible units are made by injecting a mist of dioctylphthalate (DOP) upstream of a filter to verify that air is not bypassing the filter. In large systems where more than one filter is tested in parallel, the efficiency must be at least 99.95%.

Each separations area was originally equipped with a sand filter. A second sand filter was added for 221-F in 1975, and a second sand filter for 221-H is under construction. Performance of the sand filters is of particular importance in limiting the amounts of particulate activity releases from the separations areas. The original sand filters are 100 ft wide, 240 ft long, and 8 ft deep. Only the roof section is above ground level, and it is supported by columns that extend through the filter bed. The sides are waterproofed concrete. General construction of a filter is shown in Figure II-8. The bed is made from layers of sands and gravels in graduated sizes. The largest rocks (1-1/4-in. to 3-in. diameter) are supported on special tiles which distribute the air from 12 tunnels connecting to the main supply tunnel. Progressing from the bottom to top, the air from the canyon building passes through layers of successively smaller particle size until the primary filter medium (20 to 50-mesh sand, 3 ft deep) is reached. Coarser sand or gravel top the bed to prevent loss of sand through entrainment. Air passes through exit ports into an exhaust tunnel and then to a 200-ft stack. Sumps are provided in the supply and exhaust tunnels to collect any condensed moisture.

The new sand filters are of similar construction except that they are 300 ft long, the tile supports are replaced with stainless steel grates, and an additional 8-in.-thick layer of sand with a greater percentage of fines acts as the final filter medium. This additional sand layer improves the sand filter efficiency.

Total routine flow through a bed has ranged from 100,000 to 130,000 ft³/min depending on pressure drops allowed and operating conditions. No single number can be given for filtration efficiency, but under current routine operating conditions the release levels from the filters correspond to efficiencies of 99.8% to 99.99% for beta-gamma and 99.5% to 99.9% for alpha activity.

Tritium Facilities (232-, 234-, and 238-H). The tritium facilities are served by three 200-ft stacks and one 75-ft stack, which normally exhaust a total of 264,000 ft³ of air per minute. Tritium in this exhaust air, at very low concentrations, arises from sources such as unavoidable discards of waste gas streams, leaks, maintenance work, handling the lithium-aluminum targets, loading and unloading the extraction furnace, and packaging operations.

Receiving Basin for Offsite Fuels (244-H). The air from this building is exhausted through a HEPA filter and a short stack 53 ft above grade, and a separate vessel vent system is exhausted through a filter and 10-in.-diameter pipe, also at the 53-ft height.

Metallurgical Building (235-F). Ventilation air from the process cabinets and areas is filtered through HEPA filters and exhausted through a stack and discharged at 75 ft above grade. When the PuFF Facility installation is completed, ventilation air from PuFF and from the existing portion of 235-F will be exhausted through HEPA filters, the F-Area sand filter, and then the 200-ft stack.

Analytical Laboratory (772-F). The general ventilation air from this building is filtered through HEPA filters and exhausted via a separate 75-ft stack. Exhaust air from glove boxes is routed to the area sand filter.

Stack Releases

The activity released from the various air exhaust stacks in the two separations areas has varied widely as methods of operation have changed. Process deviations and equipment failures also have given above-normal releases. Yearly releases are tabulated in Appendix A, and offsite effects are discussed in Section III.

Canyon Buildings (221-F and 221-H). The radioactivity normally in the input air to the various filters results from entrainment of process solutions while sparging, transferring by steam jet, or evaporating. Additionally, gaseous ruthenium and iodine may be evolved in varying amounts as chemical conditions change, e.g., volatile ruthenium tetroxide may be evolved in large quantities if excess permanganate is added to give strongly oxidizing conditions when an MnO₂ precipitation is made to remove fission products from solution.

The various operations above contribute significant quantities of only two chemicals to the exhaust air from the stacks, nitrogen oxides (NO_x) and ammonia. In F Area, about 300,000 to 400,000 lb of nitrogen oxides are released per year from dissolving of irradiated fuel elements in nitric acid. In H Area, about 150,000 to 200,000 lb of nitrogen oxides are released per year. Dissolving of aluminum fuel cladding in sodium hydroxide causes the release of about 13,000 to 15,000 lb/yr of ammonia from the F-Area stack.

The old sand filter units have served well to retain activity and reduce releases except in two specific cases when failure of a support tile in the bottom air distribution system let the gravel and sand fall into the bottom supply air tunnel, forming a small hole in the bed in each case. In H Area, activity escaped over a period of five days in 1969 while the cause was being sought and before the hole was plugged; total activity released during the month of the incident was about 12 Ci of beta-gamma activity (about 70% Ru) and 0.6 Ci of alpha activity. In F Area in 1971, the H-Area experience led to quick recognition of a similar problem, and total activity released during the month of the incident was about 0.071 Ci of beta-gamma activity and 0.0003 Ci of alpha activity. The new sand filters are designed to prevent a similar type of failure.

Building 211-F. Off-gases from the acid absorber unit and from the SRL trailer unloading system contribute a measurable fraction of the radioactivity in the air normally exhausted from the main 221-F stack. Specific cases of above-normal activity have resulted when the exhaust from the denitrators has been sent directly to the stack, bypassing the acid absorber unit, to prevent pressurization of the denitration vessels. High fission-product activity in the recovered uranium also has contributed to activity in the exhaust air as well as in the recovered acid. Activity has been observed in the air from the trailer unloading station when waste from the SRL curium processing programs was transferred. This area has no apparent large potential for abnormal releases to the air.

Tritium Facilities (232-, 234-, and 238-H). Releases from the tritium process buildings can be categorized according to the controls that are imposed on the specific streams. Tritium escapes in very low concentrations in discard of discrete batches of inert gas or air, in disposal of the light hydrogen isotopes as waste from the isotopic separations, and in unavoidable release into the general ventilation system (from leaks, opening equipment, etc.).

The first category of release is individual waste batches of inert gas or air. Absorption beds are used where feasible to reduce the amount of tritium that otherwise would be lost this

way. One system in use has an oxidizing bed to convert any elemental tritium into water, followed by a zeolite bed to absorb the water; another system requires only the zeolite beds.

A second category of release is the waste from purification operations in which tritium (^3H) is separated from protium (^1H) and deuterium (^2H). The final waste fraction is analyzed for tritium before discard and is recycled if the tritium is recoverable. However, the waste hydrogen isotopes must be discarded at some point in the cycle and will contain some amount of tritium, which ultimately will be determined by separations capacity and cost.

The third category of release includes releases to the general ventilation system, so that the tritium is mixed with very large volumes of air. The large volume of dilution air precludes simple recovery because of the resulting low concentration of ^3H . Examples of the sources of these releases are leaks, opening lines for maintenance work, routine opening of equipment (such as the extraction furnace), and routine disconnections as materials are loaded into and out of the process system. Successive evacuations and flushings with inert gas are the steps taken to minimize the releases for maintenance and routine disconnections or equipment openings. Further discussion of tritium processes can be found in WASH-1269, *Tritium Control Technology*.¹¹

Unusual tritium releases occurred in May 1974 and in December 1975 due to malfunctioning equipment. These releases are described in Appendix J.

Receiving Basin for Offsite Fuels and Resin Regeneration Facility (RBOF-RRF) (244/245-H). In the receiving basin operations, casks are vented to the vessel vent system on receipt, so that any fission product gases released in transport will not constitute a source of occupational exposure to the operating personnel when the cask is opened. Underwater hoods are used to collect gaseous releases when fuel is cut and to divert these gases to the exhaust system. The continuous monitoring of the stack shows that the total amount of activity released has been small; some minor releases have resulted from solutions from resin regeneration. In 1975, releases from this facility were estimated to be 350 Ci tritium, and 48 μCi ^{131}I , predominantly from fuel cleaning operations.

Analytical Laboratory (772-F). Fission-product releases have been small (in 1975, 0.11 mCi), but alpha activity from the laboratory stack occasionally approaches that from the 221-F canyon building. Although alpha releases have been small (in 1975, 0.02 mCi) and no significant incidents have occurred, the presence of plutonium powders and some few high-level samples

do constitute a potential for release. Adherence to extensive administrative controls minimizes the potential for such unplanned releases, and the filtration system provides an additional line of defense to minimize potential consequences in event of an unlikely incident.

Metallurgical Building (235-F). Releases from this facility, which began operation in 1961, have been very low, as typified by the 1975 values of 15 μCi beta and 2.0 μCi alpha. Operation of the PuFF facility would not be expected to increase these values significantly.

Other Atmospheric Releases

Some of the tritium (as tritiated water) in the waste storage tanks is released to the atmosphere in the ventilation air as a result of both evaporation and exchange with water vapor in the air. About 4000 Ci of tritium is released per year from this source. About 5000 Ci of tritium (as tritiated water, HTO) is estimated to have evaporated from the surfaces of the seepage basins during 1975 as a result of the discharge of 14,000 Ci to these basins.

Aqueous Releases

Water Handling Systems

Main Cooling Water Systems. The two separations areas have water-handling systems that are basically the same in concept, although different in detail. Each canyon building has a single cooling water supply system fed from a cooling tower, but has two return systems: a "circulating" system which returns the water to the cooling tower for re-use, and a once-through "segregated" system which collects the effluent cooling water from those vessels most likely to introduce contamination into the water. The objectives of the dual return system are to minimize chances of contaminating the entire supply system and to reduce the amount of contaminated water to be handled if a coil leak occurs. The circulating cooling water return system serves the general area and building and is monitored continuously to detect any contamination from leaks in the cooling coils of process vessels that contain activity. If activity is found, the water is diverted to seepage basins at low levels of activity and to lined retention basins at moderate levels. The segregated cooling water is a once-through system of relatively low flow and specifically serves process vessels in which the same coil is used for both heating and cooling, by using either steam or water. This once-through water is normally

discharged to Four Mile Creek (in both F and H Area). If activity appears in the segregated system as a result of coil failure, the water may be diverted to the seepage basins or retention basins, as in the case of the circulating water. The two areas are different in that the segregated water in F Area is continuously monitored and discharged through a delaying basin that provides four hours of holdup to allow time for action if activity is detected. In H Area, the segregated water is collected and discharged on a batch basis according to the monitoring results.

Control limits for cooling water diversions are: to Four Mile Creek, less than 1.4×10^{-6} $\mu\text{Ci/ml}$ alpha* and less than 4.5×10^{-6} $\mu\text{Ci/ml}$ beta-gamma; to seepage basins, less than 4.5×10^{-5} $\mu\text{Ci/ml}$ alpha and less than 4.5×10^{-4} $\mu\text{Ci/ml}$ beta-gamma. If the seepage basin limits are exceeded, the water is sent to impermeable basins for treatment (e.g., by ion exchange).

Report DP-1349, *Control and Treatment of Radioactive Liquid Waste Effluents at the Savannah River Plant*,¹⁰ contains schematic diagrams and further descriptive material for the 200-F and -H water-handling systems. Yearly releases are tabulated in Appendix A of this statement, and the offsite effects are discussed in Section III.

Resin Regeneration Facility (245-H). In the resin regeneration facility, resin is flushed from deionizers into regeneration tanks, separated into cation and anion fractions, regenerated with nitric acid and sodium hydroxide, and loaded back into the de-ionizer units.

Liquid regeneration wastes are qualitatively monitored before disposal, and solutions with low levels of activity are collected, analyzed, and transferred to the seepage basin. Solutions with moderate levels of activity are routed to the cesium removal column in the waste tank farm and then sent to the seepage basin. At still higher levels, the wastes go through the 242-H waste evaporator, with just the condensate sent through the cesium removal column to the seepage basin, while the evaporator bottoms are stored in a waste tank. In 1975, solutions averaging 1.2×10^{-4} $\mu\text{Ci/ml}$ (exclusive of tritium) were sent directly to seepage basins. Solutions between 1.4×10^{-3} and 5×10^{-2} $\mu\text{Ci/ml}$ were sent through the cesium removal (ion exchange) columns and then to the seepage basins; those over 5×10^{-2} $\mu\text{Ci/ml}$ were sent to waste tanks for redistillation.

Releases to Plant Streams

Liquid effluents from the separations areas are directed to Four Mile Creek, to seepage basins, to lined retention basins,

* 10^{-6} μCi = 0.000001 μCi = 1 pCi.

or to waste tanks, according to the activity level. The goal always has been to minimize the activity sent directly to the creek, although peaks of activity in the creek water have resulted from individual incidents in which solution could not be diverted. Also, some activity has migrated from the seepage basins through the ground water to the creek. Data on the specific contributions from the seepage basins are available only since 1967, when continuous stream samplers and flowmeters were installed. Individual active components reach the water from different sources: 1) only small amounts of activity actually are in the direct plant effluents; 2) most of the tritium observed in Four Mile Creek upstream of the reactor area effluents has migrated from the F and H seepage basins; 3) the ^{90}Sr observed in Four Mile Creek migrates from the F and H seepage basins; and 4) the small amount of ^{137}Cs observed currently is desorbing from the stream bed. The source of the ^{137}Cs was a spill of high-level waste described in Section III.

There is no thermal effect from the separations areas effluents on plant streams. The effluents are essentially at ambient temperature where they flow into Four Mile Creek. About 15,000 lb/yr of detergents from laundry operations were discharged to Four Mile Creek prior to their discharge to the seepage basins beginning in 1976.

Releases to Seepage Basins

The 200-F and 200-H seepage basin systems are shown in Figures II-3 and II-13 and in Figures E-4 and E-5 in Appendix E. The basin systems consist of three earthen basins in F Area and four in H Area (one currently inactive). In F Area, the total capacity is 14 million gallons. The three active basins in H Area have a total capacity of 41 million gallons. These basins provide a means (by filtration, soil sorption, and radioactive decay) for reducing releases reaching streams leaving the plant. In their absence, large volume aqueous effluents containing low concentrations of radioactive materials would be released directly to plant streams or would require extensive treatment facilities. Effects of seepage basins usage are given in Section III. Alternatives to their use are discussed in Section V.

In recent years, total volumes of water disposed to the basins have ranged from 80 million to 130 million gallons each year. As water migrates from the basins, the movement of the various radioactive elements is slowed by ion exchange with the soil. The only radioactive isotopes that have moved far enough through the soil to reach a plant stream (Four Mile Creek) are tritium and ^{90}Sr . The tritium would be expected to move with the ground water, but is depleted to about one-third of its

original activity before it reaches the stream. The strontium is present because it is not held strongly by the soil under the varied chemical conditions that have occurred in the basin water. Plutonium and cesium are tightly bound by the soil and move very slowly.

In addition to the other seepage basins, an abandoned basin on the north side of F Area in the Upper Three Runs watershed was used for a short time at startup and has been used at infrequent intervals since then to dispose of nitric acid solutions containing some natural uranium. Small amounts of uranium are found where the F-Area storm drain (and any material coming from the basin) reaches Upper Three Runs.

The sources of solutions containing radioactivity that are sent to the seepage basins are shown in Figure II-9. In both areas, the condensates from the various evaporators are major contributors to the volume of water going to the seepage basins. In F Area, the largest volumes come from the nitric acid recovery unit, the waste storage system evaporator, and the evaporators that concentrate the dilute uranium nitrate solutions. In H Area, the nitric acid recovery unit, the waste storage system evaporator, and the general purpose evaporator are primary sources. Large amounts of water in this area also come from the Receiving Basin for Offsite Fuels and the tritium processing buildings. Lesser contributors in both areas are sumps and drains from many operating areas that contain only low levels of activity.

In each area there is a continuous sampler on the composite water stream flowing to the seepage basin. Samples are analyzed daily, and more-complete analyses are obtained on weekly composite samples. In 1975, discharges to the 200-F and 200-H seepage basins included 14,000 Ci of tritium, 0.2 Ci of $^{238,239}\text{Pu}$, 7.2 Ci of ^{137}Cs , 0.8 Ci of ^{90}Sr , and about 18 Ci of shorter-lived beta-gamma activities.

Typical annual discharges of chemicals to the basins include about 600,000 lb of HNO_3 , 200,000 lb of NaOH , 12,000 lb of H_3PO_4 , 1200 lb of $\text{Na}_2\text{Cr}_2\text{O}_7$, and 50 lb of mercury. Much of this is HNO_3 and NaOH from resin regeneration and H_3PO_4 and $\text{Na}_2\text{Cr}_2\text{O}_7$ from Li-Al target cleaning. Mercury has been used routinely in H-Area operations since 1959, and occasionally in F Area, to aid in dissolving aluminum alloy fuels. The sodium hydroxide that is used in large amounts in both areas also contains trace amounts of mercury as an impurity. The total amount of mercury used in fuel dissolving has been about 190,000 lb; most of this has been retained and stored in the high-level waste tanks. About 4700 lb have been discharged to the seepage basins, both as spills and flushes from the chemical feed preparation areas and in the condensates from evaporation of the high-level wastes.

Deposits in the Soil

The activity currently in the soil at the separations areas consists of the inventory in the soil under seepage basins, several leaks into the ground from waste transfer operations, fallout from stacks, general burial waste, and possible (but unconfirmed) leaks to the soil under the canyon buildings from spills inside the building. Decay of short-lived activities reduces the inventories of activities in these places to less than the sum of all input activity. An estimated 10 to 500 Ci of ^{137}Cs remains in the ground from an H-Area waste tank leak (Tank 16), and an estimated 75 to 100 Ci of ^{137}Cs remains from an overflow when a waste tank inlet riser plugged during a transfer (Tank 9). In this last case, most of the contaminated ground was excavated and transferred to the burial ground. Waste containing an estimated 3000 to 5000 Ci of ^{137}Cs escaped into the soil below grade adjacent to Tank 8 in 1961 when the tank was temporarily overfilled. However, the radioactivity has been tightly held by about 1000 ft³ of soil 14 to 26 ft below grade (18 ft above the main water table) and has not migrated (see p III-86 for additional information).

Minor amounts of activity have also been deposited in the soil infrequently from other abnormal operations. It has been standard practice to note the location and date of the occurrence with a marker and, where possible, to remove the contaminated soil to the burial ground and cover the residue in the soil with asphalt or clean soil. Most of these locations were contaminated years ago with mixed fission products, and most have decayed to low radioactivity levels. These locations are described in Appendix A.

FUEL AND TARGET FABRICATION (300 AREA)

The facilities for fabricating fuel and target elements to be irradiated in SRP reactors are located in the 300-M Area. Major products are extruded enriched uranium-aluminum alloy fuel and canned depleted uranium metal targets. The only significant radioactive release from these facilities is uranium in aqueous effluents from the canning process. The total amount released since 1955 is about 85,000 lb or 24 Ci of uranium. About 50% has settled in plant stream and pond beds adjacent to the area and has not reached the main stream leading to the river.

The 300-Area processes currently result in annual releases of about 34,000 lb of nitric acid and oxides of nitrogen to the atmosphere; about 500,000 lb of acids, bases, and salts in solution to a settling basin; and about 2,000 lb of uranium to a plant stream. The uranium will be discharged to the settling basin when the plating line rinse water is diverted in late 1976.