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EXPERIENCE WITH PROCESSING IRRADIATED FUEL AT THE SAVANNAH RIVER PLANT (1954-1976)

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MASTER



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PREPARED FOR THE U.S. ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION UNDER CONTRACT AT(07-2) 1

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Printed in the United States of America

Available from

National Technical Information Service

U.S. Department of Commerce

5285 Port Royal Road

Springfield, Virginia 22161

Price: Printed Copy \$5.25; Microfiche \$3.00

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EXPERIENCE WITH PROCESSING IRRADIATED FUEL AT THE SAVANNAH RIVER PLANT (1954-1976)

by

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Publication Date: September 1977

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ABSTRACT

The processing facilities for recovery of uranium and plutonium from irradiated fuel elements have operated since 1954 without major unplanned interruptions. The operation has comprised campaigns ranging from a few weeks to two years, with no prolonged outages except for a period of about two years when one of the two processing facilities was remodeled to increase its capacity. Over the 23-year period 1954-1976, approximately 30,000 metric tons of irradiated uranium were processed. Since 1958, in addition to recovery of uranium and weapons-grade ^{239}Pu , the plant has produced ^{238}Pu , which is used principally as a heat source. Through June 1976, a total of 320 kg of ^{238}Pu has been shipped offsite.

There have been no lost-time injuries due to radiation and no criticality accidents in these or other Savannah River Plant (SRP) facilities. However, three accidents in the separations areas have caused damage in excess of \$50,000. The most serious accident was a cooling-water contamination incident which cost about \$250,000 for cleanup. All radioactive contamination was contained within the affected buildings in these accidents.

Radiation exposures to individual workers in fuel reprocessing at SRP have averaged 0.3 to 0.7 rem per year. There have been no whole body exposures in excess of the U.S. Energy Research and Development Administration (ERDA) guide of 5 rem per year. The maximum confirmed dose to an employee to date was 3.1 rem in 1966. The average whole body exposure per year to workers for the past 12 years has been 0.5 rem or 10% of the ERDA guide. A total of 374 workers have been listed in the National Plutonium Registry as having positive plutonium uptakes, but no uptakes exceed the permissible plutonium body burden as established by the International Committee for Radiation Protection (0.04 μCi), and 88% of the listings have uptakes of less than 10% of the allowed body burden.

Releases of radioactivity to the atmosphere and to plant streams and environmental levels of radionuclides have been monitored since startup. The total offsite dose commitment (i.e., the actual dose plus the expected residual dose for a 70-year period) from SRP fuel reprocessing areas to the census population of about 670,000 persons within a 100-km radius of SRP and to 70,000 consumers of Savannah River water downstream of SRP was determined to be about 500 man-rem for the 23-year operating period. Natural background radiation to this same

population over this period is about 1,830,000 man-rem. The highest potential radiation exposure to an offsite individual was 1.2 rems to the thyroid during an accidental release of iodine-131 in 1961. The highest potential offsite whole-body dose from fuel reprocessing was calculated to be 0.47 millirem for the entire year in 1956.

Fuel irradiated in SRP reactors is stored in a water-filled basin at each reactor for a period of time to permit decay of short-lived radioactivity before shipment to the reprocessing areas. Currently that storage period is a minimum of 200 days. In addition to its fuel processing activities, SRP stores a number of special ERDA-irradiated fuels which require shear-leach dissolution or other major processes not available at SRP. These fuels, containing a total of 2500 kg of ^{235}U , are stored underwater in the RBOF facility. A number have been in storage since 1968. Storage in RBOF has been without significant incident.

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EXPERIENCE WITH PROCESSING IRRADIATED FUEL AT THE SAVANNAH RIVER PLANT (1954-1976)

INTRODUCTION

This report summarizes 23 years of radiochemical processing experience at the Savannah River Plant (SRP), including handling, storing, and processing irradiated reactor fuel and targets and in producing plutonium and other actinides. It emphasizes experience concerning productivity, personnel radiation exposure, radioactive effluent releases, and environmental effects. Almost all of this material has been published previously. A bibliography cites detailed references.

Construction of SRP and subsequent operation was undertaken for the U.S. Atomic Energy Commission in 1950 by E. I. du Pont de Nemours and Company, primarily to provide plutonium and tritium for the nation's defense program. The first irradiated fuels were processed at SRP in 1954, and Du Pont operation is continuing under the U.S. Energy Research and Development Administration.

SRP occupies an approximately circular area of 300 square miles (192,000 acres) in South Carolina, 25 miles southeast of Augusta, Georgia, and about 120 miles inland from the Atlantic coast (Figure 1). Production facilities include:

- Five heavy-water cooled and moderated production reactors, two of which are currently shut down and on standby because current production requirements are being met with three reactors.
- Storage basins for irradiated fuels.
- Two fuel reprocessing areas for recovering uranium and plutonium from the irradiated fuels.
- A tritium separations facility.
- Waste management facilities.
- A reactor fuel and target fabrication plant.
- A heavy-water production plant.

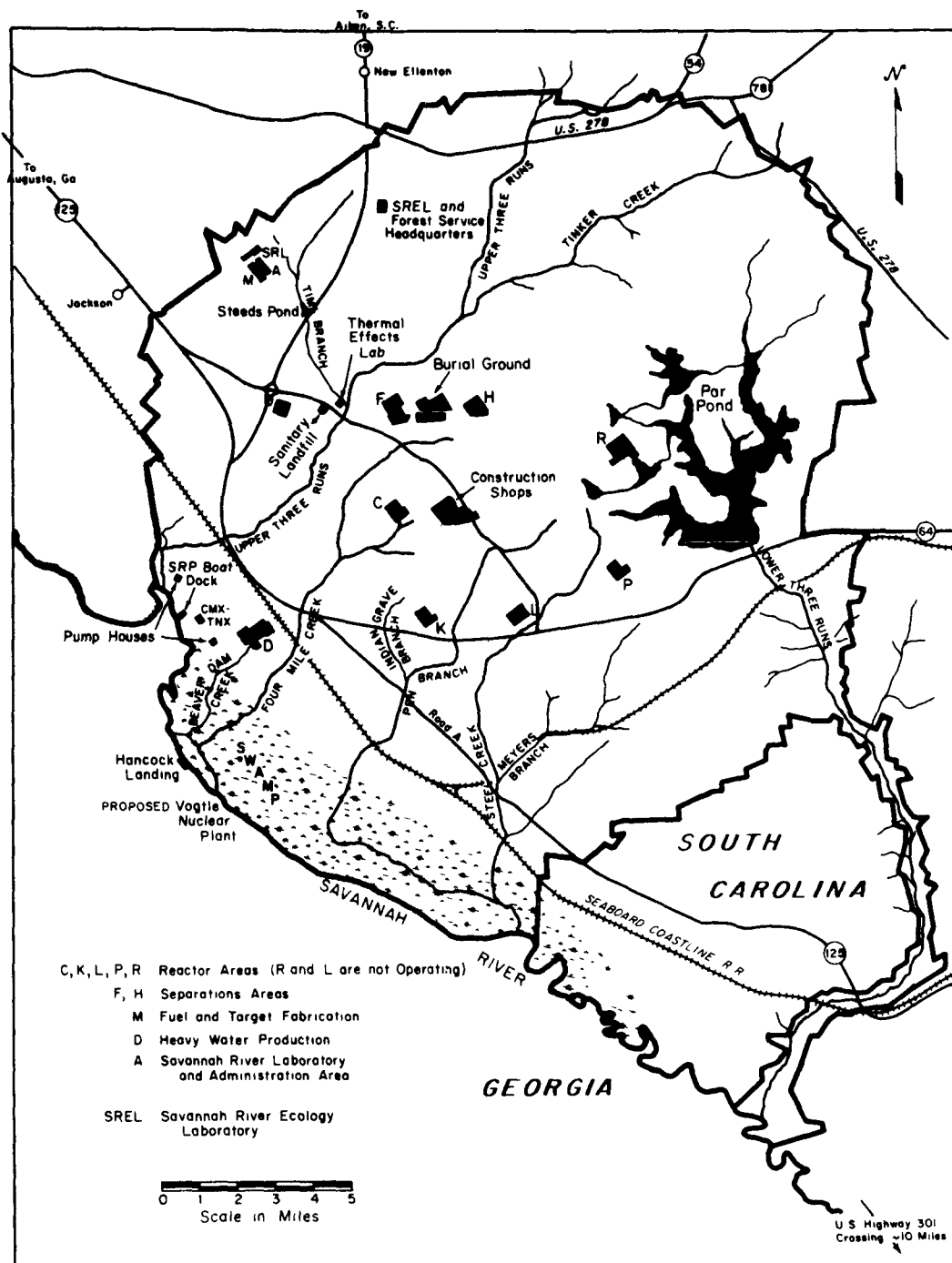


FIGURE 1. The Savannah River Plant Site

The Savannah River Ecology Laboratory operated by the University of Georgia, and the Savannah River Laboratory, which is operated by the Du Pont Company primarily for SRP technical support are also located on the plantsite. Forest and land management and related programs on the heavily forested site are the responsibility of U.S. Forest Service personnel assigned to SRP. The total working population on the site has ranged from 9000 down to the present level of about 6000; of that number, about 1360 are employed in the reprocessing areas.

The two fuel reprocessing areas, designated as F-Area and H-Area, are located near the center of the plantsite. The primary reprocessing facilities are housed in two shielded concrete canyon buildings, 221-F and 221-H, which provide for remote operation and maintenance of the processing equipment. Irradiated fuel is received at the reprocessing facilities by rail casks from the spent fuel storage basins of the SRP reactors or from the receiving basin for offsite fuels (RBOF), in which fuels from offsite reactors are either stored or prepared for processing. After chemical dissolution in nitric acid, the irradiated fuel is partitioned into separate uranium, plutonium, and waste streams by the Purex solvent extraction process. These streams are then separately processed to yield uranium oxide, plutonium metal, and an NaOH-neutralized waste solution. Although there are many significant differences among fuel forms, processing steps, and product forms, processing is basically similar to that proposed for light-water power reactor fuels.

The SRP tritium separations facilities, the production reactors, and the heavy water manufacturing facility are not within the scope of this report.

FACILITIES DESCRIPTION

Site

The Savannah River Plant (SRP) site is bounded on the southwest by the Savannah River and centered approximately 25 miles southeast of Augusta, Georgia.¹ The chemical separations areas (Figure 1) are centrally located on the plantsite.

The site has an elevation of between 90 ft and 400 ft above mean sea level, and all operating areas drain to the Savannah River. The elevation of the river at the site is approximately 84 ft above mean sea level. The plant lies on the Atlantic Coastal Plain physiographic province, and is underlain by the Tuscaloosa aquifer, which supplies well water to several operating areas.

Reprocessing Facilities

Chemical processing of irradiated reactor fuels at Savannah River is carried out in two processing areas, designated F and H on Figure 1. Each area contains a large concrete-shielded process facility or "canyon" building, designated as Building 221-F and Building 221-H, respectively. These areas also contain the receiving basin for offsite fuel (RBOF), tritium recovery and reprocessing facilities, neptunium and ^{238}Pu recovery and fabrication facilities, and waste handling and storage facilities.

Reprocessing of irradiated fuels at SRP began in 1954. Until 1957, plutonium was recovered from aluminum-clad irradiated uranium in both fuel reprocessing plants. Since 1959, all plutonium is recovered in Building 221-F; enriched uranium from aluminum-clad uranium-aluminum fuels is recovered in Building 221-H. The Purex tributyl phosphate solvent extraction process is the primary separations system in these plants. The process and a comparison of performance with design were described by Joyce in 1960.² Figure 2 summarizes the monitoring of effluents from fuel reprocessing, while essential features of Building 221-F and 221-H are shown in Figures 3 and 4 of this report, and in Figure II-7 of Reference 3.

The basic operations in the SRP separation of plutonium, uranium, and fission products from irradiated fuel elements include:

- Chemical removal of the aluminum jacket followed by dissolution of the uranium fuel.
- A head-end treatment primarily to remove insoluble matter, e.g., silica.
- A first cycle of solvent extraction for separating plutonium and uranium from fission products, followed by partition of plutonium from uranium.
- A second cycle of solvent extraction for each metal (plutonium and uranium) for further decontamination from fission products.
- A conversion process to produce plutonium metal.
- A denitration process to produce uranium oxide.

The key operations are the decontamination and partition cycles in remotely operated facilities. Plutonium and uranium are decontaminated from fission products by factors in excess of one million; plutonium is separated from uranium to the parts per billion range for plutonium in uranium, and recovery of plutonium and uranium is greater than 99.5%.

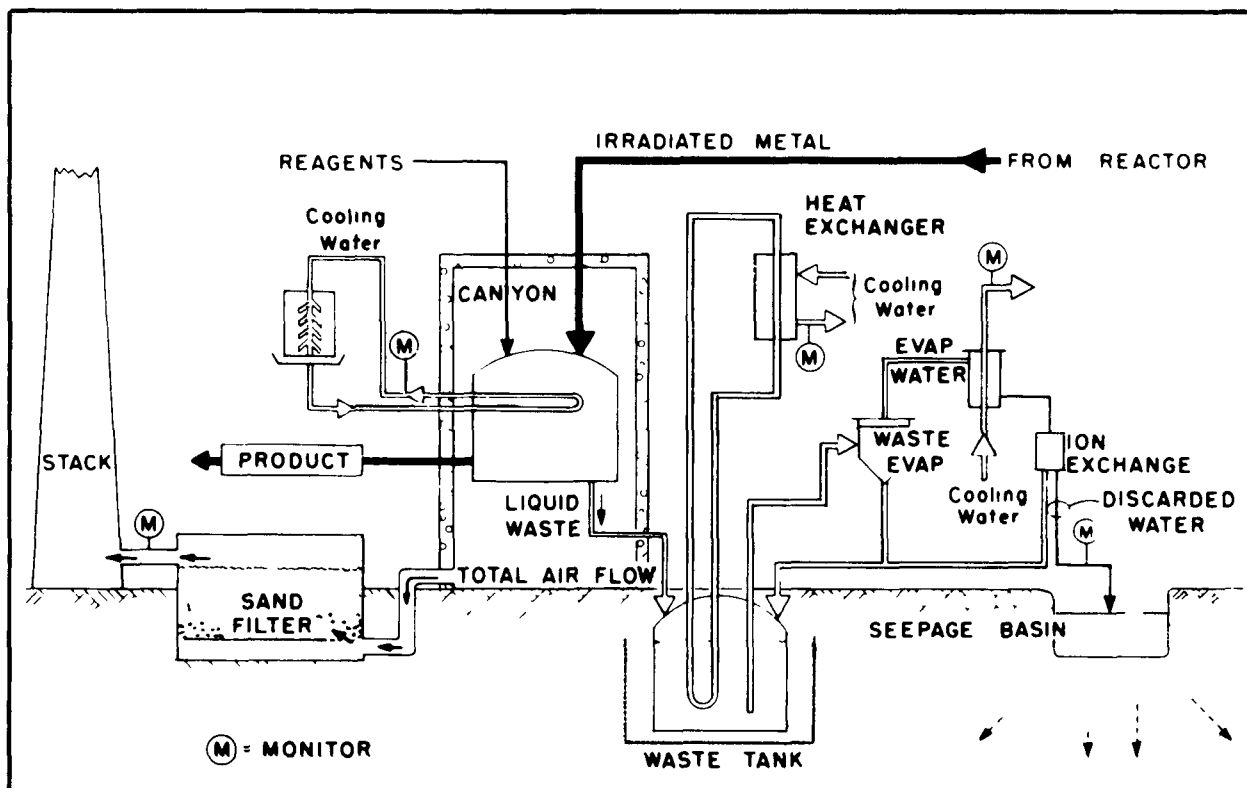


FIGURE 2. Effluent Monitors

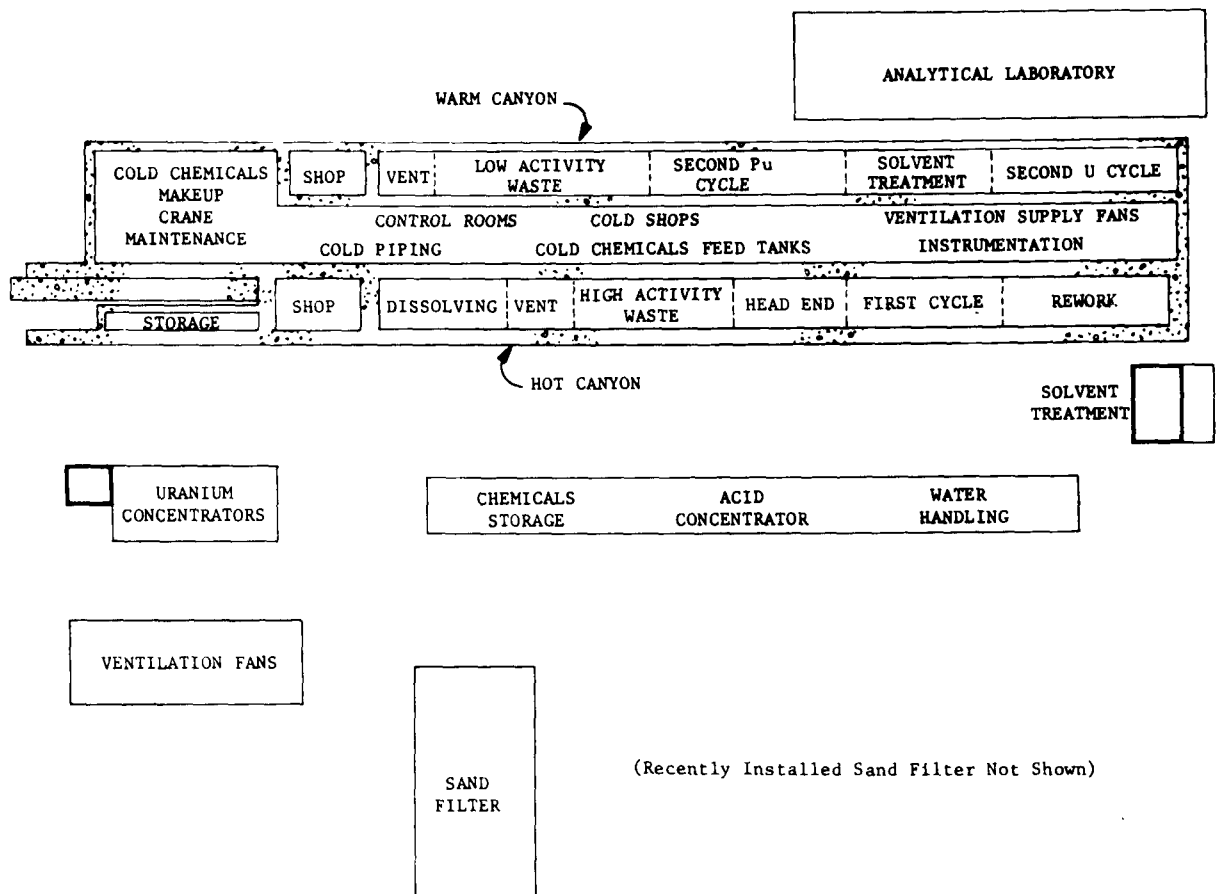


FIGURE 3. Plan View of Canyon Building and Supporting Facilities

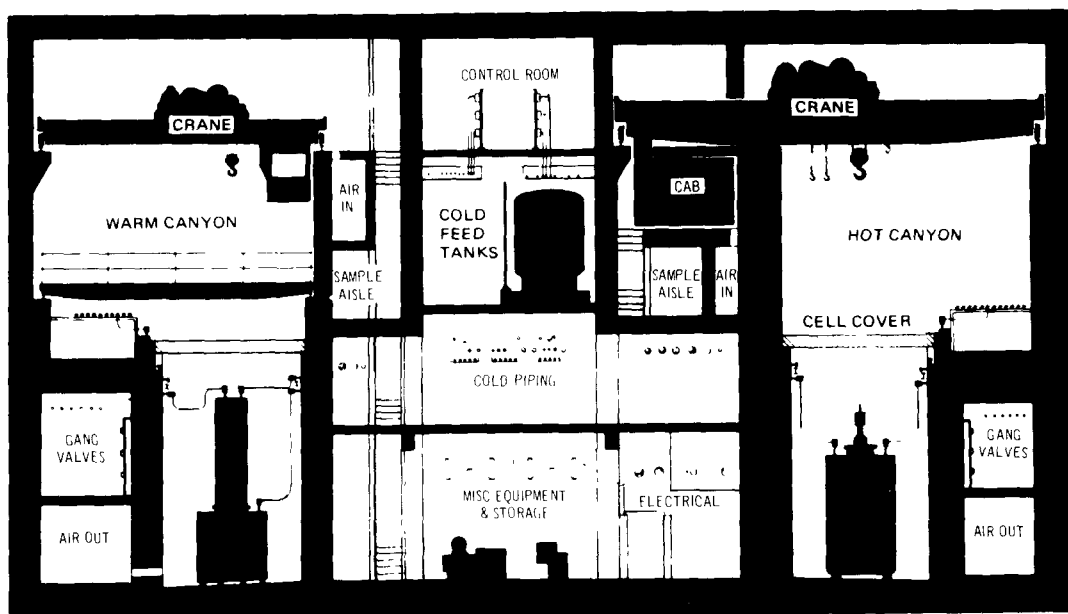


FIGURE 4. Cross Section of Canyon Building

Plutonium Facilities

Following separation from uranium in the heavily shielded canyon process, the SRP plutonium is converted to metal in an enclosed but lightly shielded refining facility. This facility, which has been described by Orth,^{4,5} incorporates a recovery operation where residues, filtrates, and scrap are processed for return to the main process line. Offsite ERDA plutonium and scrap may also be recovered in this facility. Figure 5 shows a plan view of the facility; Figure 6 is a flow diagram for ventilation.

The plutonium solution product from the Purex process goes through four major process steps to form the metal.

- Ion exchange for concentration and purification.
- Precipitation of plutonium as the trifluoride.
- Oxidation of the plutonium trifluoride to plutonium tetrafluoride.
- Reduction of the tetrafluoride to plutonium metal.

The refining operation for recovery of plutonium produces significant quantities of liquid and solid residues. Solid residues of plutonium tetrafluoride are dissolved and combined with any high loss filtrate. The fluoride is complexed with aluminum and the solution is processed through an ion exchange column. The plutonium solution eluted from the column is returned to the Purex plant for additional purification.

Plutonium metallic and oxide scrap are dissolved in small batches. These solutions are returned to the main Purex plant for separation from uranium and purification. In this way, off-specification scrap can be blended with newly made plutonium to produce acceptable product.

Receiving Basin for Offsite Fuels

To provide storage and processing capability for non-SRP-produced fuels, a receiving basin for offsite fuels (RBOF) was constructed at SRP in 1963. Design of the facility was described by Smiley.⁶

The facility (Figures 7 and 8) provides for receipt and storage of a variety of irradiated nuclear fuels in a system of water-filled basins. A building ventilation system filters and removes airborne particulate contamination before exhaust. Underwater cask unloading and fuel element handling were specified because experience with such facilities is well established and reliable; water is the primary shield for absorbing radiation.

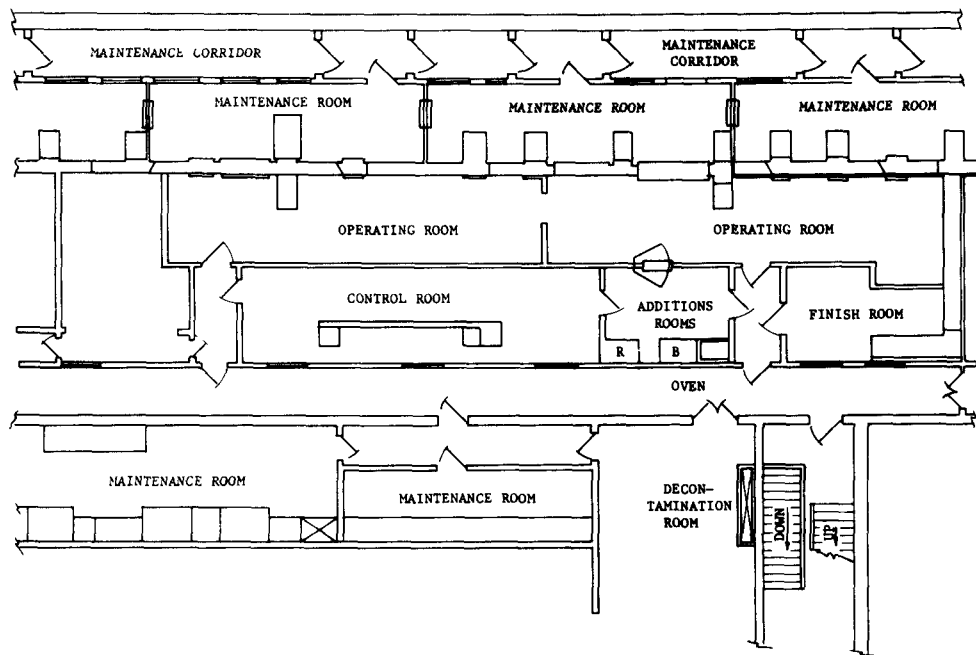


FIGURE 5. Main Floor of Plutonium Processing Facility

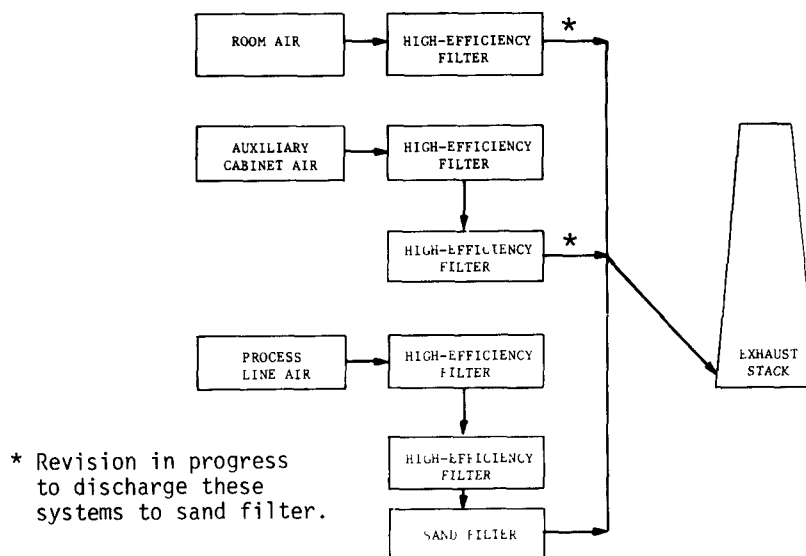


FIGURE 6. Path of Plutonium Facility Exhaust Air

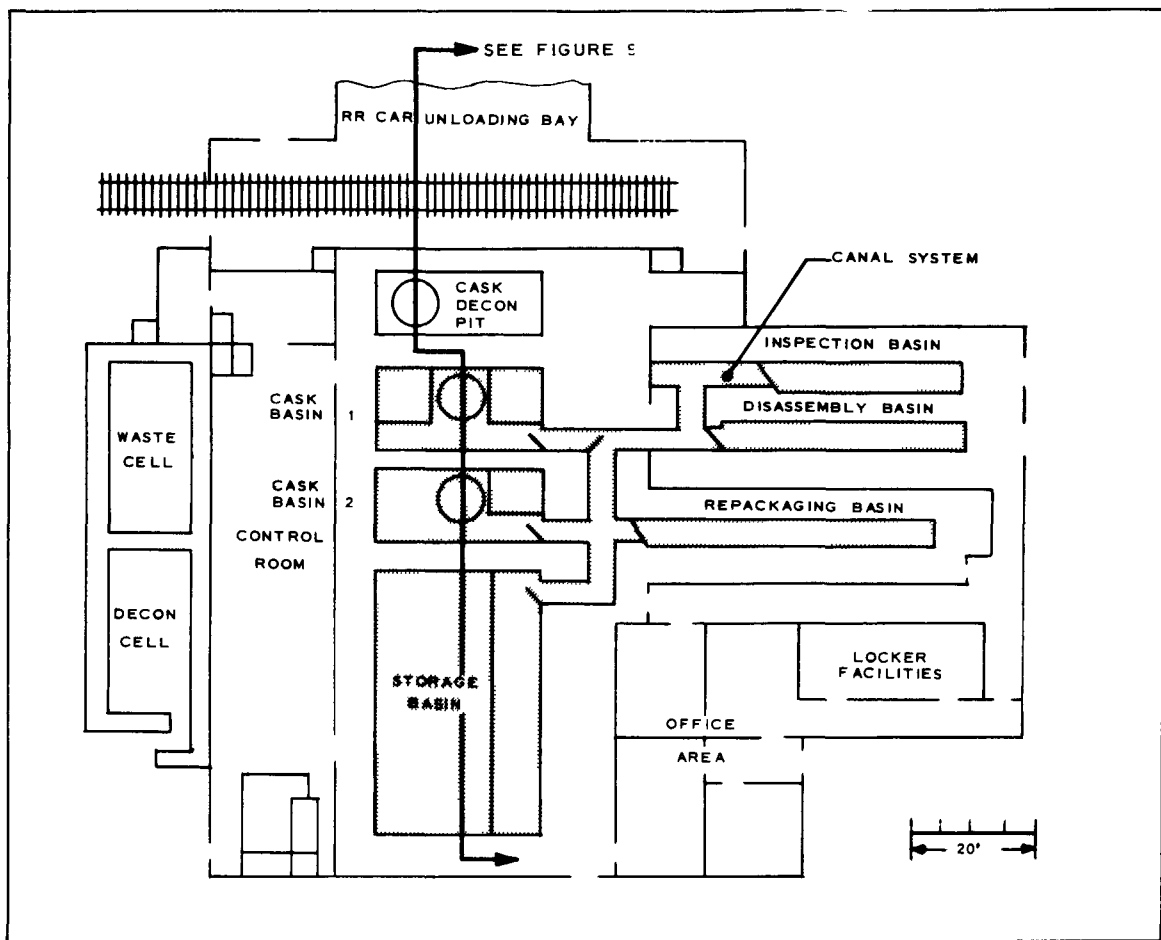


FIGURE 7. Plan View of RBOF

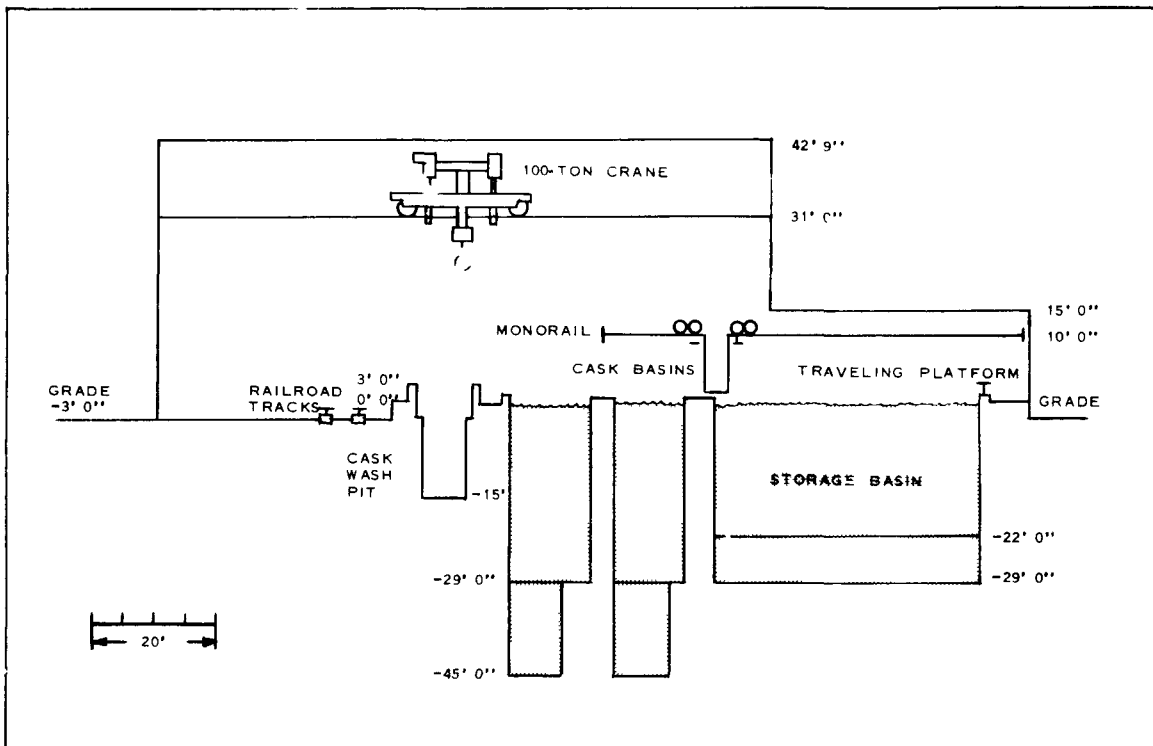


FIGURE 8. Elevation View of RBOF Basin Area

Concrete shielding is installed where required to reduce personnel radiation exposure during routine operation of the basin water purification system. Contaminated liquid waste from the purification process is discharged to waste storage tanks through imbedded or shielded pipes, valves, pumps, and tanks. Certain liquids (cooling water, evaporator condensates, etc.), which meet ERDA discharge guides for release to the uncontrolled (public) zone, are instead released to seepage basins which provide a significant holdup and decay of the minimal radioactivity.

Maintenance of most equipment is direct; water or portable shielding is used as required. Most valves in contaminated water handling service piping are located in a direct maintenance basement beneath the Control Room.

Cask and fuel handling facilities incorporate sufficient flexibility to receive and handle a variety of fuel assembly and shipping cask designs. Because fuel assemblies are stored in racks supported from the floor of the storage pools, a variety of storage arrangements is possible. Storage rack design permits use of neutron poisons (if desired) to improve storage space utilization.

MODE OF OPERATION

Administrative Controls

The principles to control radiochemical processes entail (1) eliminating or reducing hazards by process engineering and design and (2) controlling the actions of personnel. A nuclear fuel reprocessing plant must contend with the common industrial hazards and, additionally, with those hazards that arise from the radioactive materials, e.g., the spread and assimilation of radiochemicals and the potential for uncontrolled nuclear reactions. Administrative controls in fuel reprocessing have been reviewed by Egan and Mowry,⁷ while specific controls to ensure nuclear safety have been reviewed by Forstner.⁸

The Savannah River Plant organization includes two groups that share the responsibility of procedural control: the Production Department and Works Technical Department. A separate organization, the Technical Division, provides basic technical information for the design and operation of the plant, including the technical standards which specify limits within which the process must be operated. The Production Department operates the production facilities in accordance with detailed procedures which it originates. The Works Technical Department approves procedures

prepared by the Production Department and, in addition, prepares test authorizations that permit deviations from the technical standards. The systematic use of written procedures prepared by line supervision, certified as technically correct by staff technical groups and authorized by management, has been found to give good control of the actions of personnel carrying out various operations on the plant.

Specific administrative control measures are divided into four groups: primary controls, secondary controls, system audits, and procedural deviation reports. Primary controls are those exercised by the operators in the course of following operating procedures and runbooks in the conduct of a specific operation. Included under secondary controls are monitoring instruments and other techniques for reducing the potential for nuclear incidents. System audits and procedural deviation reports ensure conformity with the written procedures.

Equipment Design

Many authors⁹⁻¹¹ have described equipment for nuclear fuel reprocessing. For SRP, a remote maintenance design was selected to meet the requirements for radiation-exposure control and to ensure that production schedules could be met.¹² With this design, the highly radioactive processes are remotely controlled behind thick walls of concrete. The irradiated fuel is charged to the process, and any failed equipment is replaced by a remotely operated crane. Primary criteria were that continually occupied areas should not exceed a radiation level of 1 mR/hr and that other operating areas, to which access would be controlled, would limit radiation levels to about 6 mR/hr.

The reprocessing canyons concept at SRP consists of two parallel rows of essentially identical concrete cells. Each cell is provided with a standard pattern of services: water, steam, electrical, sample, instrument, etc., to ensure maximum flexibility. Liquid transfers between cells are made in a pipe rack in which the piping is remotely removable.

Ventilation air in each canyon building travels from regions of lowest to highest contamination, and then is discharged through filters to a 200-ft high stack.

Nuclear safety is ensured in processing plutonium or enriched uranium by (1) limiting batch size to less than a critical quantity, (2) maintaining fissile concentrations below a critical value, or (3) processing in equipment of restricted dimensions. Criticality monitors and alarms are installed where needed to

indicate abnormal conditions. In the equipment chosen for SRP canyon processing, safe concentrations are maintained so that large volumes of solutions containing more than a critical mass of plutonium can be processed rapidly and economically. After the plutonium has been separated from the uranium and fission products, the equipment for subsequent refining of plutonium is of restricted dimensions.

Most of the equipment used in SRP fuel reprocessing is of relatively conventional design. The dissolvers, evaporators, solvent washers, and fuel and run tanks are cylindrical tanks, sometimes fitted with special inserts, which are adapted for remote operation. The unique feature of the canyon vessels is that critical dimensions are held to very close tolerances so that the vessels will fit precisely into locations determined by guides in the canyon walls and will accept prefabricated piping.

For the solvent extraction system, which is the basic operation in the Purex process, the mixer-settler equipment was originally selected for use at SRP because it is adaptable to remote operation and is of a rugged, durable design. However, the long contact time in the mixer-settler causes degradation of the solvent in the primary separation from fission products, so a development effort was initiated that led to use of the centrifugal contactor for the first stage of the Purex process. This equipment has been in constant use in Building 221-F since 1966 and has met all production goals with an attendant increase in solvent life.

The equipment for refining plutonium is specially designed to provide containment and to prevent criticality. The equipment design is kept simple to permit easy cleaning and maintenance.

REPROCESSING EXPERIENCE

Missions and Goals

Building 221-F operations primarily recover ^{239}Pu from irradiated normal or depleted uranium by the Purex solvent extraction process. Building 221-H operations primarily recover ^{235}U and ^{237}Np from irradiated, highly enriched, uranium fuel elements and also process miscellaneous fuels from research, power, and military reactors to recover low-enriched to high-enriched uranium, irradiated thorium, and ^{233}U . Building 221-H has used both 30% tributyl phosphate (TBP) in the standard Purex process and lower strength TBP in the modified Purex process. Both facilities have processed irradiated plutonium-aluminum alloy fuel elements to recover residual plutonium. The requirements for successful operation of the Purex solvent extraction

process and of related lower TBP processes have been well-documented in the Geneva conferences and in international solvent extraction conferences where SRP experience has been presented.^{13,14}

Several comparisons can be made between SRP and power reactor fuel processing. Normal SRP uranium fuels contain less plutonium than irradiated uranium fuel from power reactors, which may contain up to 8 kg/T if 33,000 MWD/T (megawatt days per metric ton) burnup is reached. Long-life fission products are also more abundant in the long-irradiated power fuels; however, higher specific powers and shorter cooling times can give equivalent or higher total fission product activity in some of the fuels at SRP. Normal SRP fuels from the Purex process are aluminum-clad uranium metal; the aluminum is removed chemically and the core is dissolved in nitric acid. Standard power reactor fuels are zirconium-clad uranium oxide; the fuel rods are chopped into short sections, the uranium oxide is dissolved in nitric acid, and the zirconium hulls are discarded. The final feed solutions in both cases are chemically similar.

The goals for process performance are essentially the same for all reprocessing systems; viz., low losses and adequate chemical and radioactive purity of the products.

The uranium processing rate demonstrated in Building 221-F is as large as any projected for commercial operation. SRP plutonium throughput in solvent extraction and purification is less than projected for large commercial facilities, but parts of the SRP plutonium system have experience at about 50% of the throughput of proposed commercial systems. The differences in plutonium levels in solvent extraction between SRP and planned commercial operations have significance primarily for nuclear safety control. However, throughput rates of 50 kg/day of ^{235}U have been sustained within strict nuclear safety control requirements in the modified Purex system in Building 221-H during campaigns of several months duration.

Fuel Processing Throughput

Required production rates in Buildings 221-F and 221-H are determined by reactor operation at the SRP site; the objective has been to maintain reactor production at the level established by ERDA (formerly AEC) to meet requirements for plutonium, tritium, and other special nuclear materials. Demands on operating capacity and production rates achieved in fuel reprocessing facilities have varied through the years. Two of the original five production reactors

have been shut down, isotopic composition of SRP reactor fuel and target elements have varied, and irradiated fuel and radioactive material scrap generated at other government-operated facilities have been processed for recovery. In addition, facility and process modifications have been made to separate and purify special isotopes, such as ^{252}Cf , for nonweapons application.

Initial operation of Building 221-F was at the design rate of about 3 tons of uranium per day. Minor changes in Building 221-H based on experience in Building 221-F permitted operation of Building 221-H at about 7 tons of uranium per day. In 1957, Building 221-F was shut down for major revisions to increase capacity. Revisions were completed in 25 months, and operation resumed in March 1959. After these revisions, Building 221-F has operated at rates up to 14 tons of uranium per day. The only significant loss of Building 221-F capability to produce uranium resulted from a denitrator explosion in 1975.¹⁵ However, most of that time was gainfully used by processing a special campaign of ^{242}Pu , which did not require operation of the denitrators. Operation of Building 221-F at a high rate after the denitrators resumed operation has reduced the inventory of discharged uranium to normal in the reactor storage basins.

The ability of Building 221-F operations to process uranium at a rate sufficient to meet ERDA requirements for plutonium permitted Building 221-H to be modified to process enriched uranium, thus eliminating shipment of enriched uranium to the Idaho Chemical Reprocessing Plant. Modification of Building 221-H was relatively minor and was completed in May 1959, after a shut-down of two months for the equipment revisions. Operation of Building 221-H since that time has kept current with the reactor output of enriched uranium fuel, and Building 221-H capacity has been sufficient to permit, in addition, the variety of special campaigns mentioned previously.

Operating efficiency, or attainment, of the SRP fuel reprocessing plants has never impeded reactor output. The overall capacity of Buildings 221-F and 221-H depends on individual capacities of several unit operations. By providing for holdup between units and individual unit capacities that are generally higher than the required overall capacity, operations can continue while most equipment repairs are made. Allowance must be made, in selecting an operating rate, for expected attainment in fuel reprocessing; the value chosen by the Production Department is 80%.

Operating Experience and Unusual Incidents

Operating experience in Buildings 221-F and 221-H has indicated no major design deficiencies, but a number of unusual incidents have occurred. Accidents in all SRP facilities are listed in WASH-1192.¹⁶ One accident has occurred since that publication, the denitrator explosion in 1975.¹⁵ Accident experience in fuel reprocessing was also reported in 1964.¹⁷ A variety of important improvements have been made since initial startup:

- Aqueous streams from the solvent extraction process contain entrained solvent. This solvent presents a potential hazard in subsequent evaporation of these streams. Safety measures were provided in Building 221-F canyon evaporators. Additional equipment was installed in Building 221-H in 1956 for removal of entrained solvent, and the high-capacity equipment installed in Building 221-F also provided for better solvent removal from aqueous streams.
- Degradation of process solvent with prolonged use, due to changes caused by radiation and chemical attack, was a continuing problem in Building 221-F. The problem was solved by adopting a new diluent, a normal (i.e., straight-chain) hydrocarbon diluent that has proved very resistant to degradation. Improved facilities for solvent recovery were also an important factor in solving the problem of solvent degradation.
- An explosion in the uranium denitration process occurred in 1975. The explosion was caused by entrained solvent that reached the uranium denitrators by a previously unrecognized route. Solvent will accumulate in any unagitated vessel that receives an aqueous stream from solvent extraction; but in the normal situation, the accumulated solvent floats on the aqueous phase and can be drawn off separately. Prior to the denitrator accident, a special flush diluted the aqueous stream, and it was not recognized that the lower density of the aqueous layer permitted accumulated solvent to sink and then be transferred to the denitrators. No injuries or radiation exposure resulted from the accident, but about \$220,000 damage resulted. Procedures and equipment have been revised to avoid repetition of that accident.¹⁵
- Proper airflow is essential for controlling migration of radioactive contamination. In the canyons, air enters near the ceiling and exits at the bottom, thus minimizing contamination of the crane. Prior to Building 221-F startup, tests showed that heated equipment, e.g.,

evaporators and dissolvers, cause thermal updrafts that would readily overcome the intended downward flow of air if covers were removed from a cell containing hot equipment. Procedures required that canyon equipment be cooled before removing cell covers, but violation of this procedure in 1959 caused severe contamination of the crane. Cleanup costs were about \$130,000. Airflow in the canyons could also be disturbed by opening the large outside door at the railroad car entrance; air locks were added in 1961 at those entrances. Major improvements were also made to facilitate decontamination and maintenance of the canyon cranes.

- Exit air from Buildings 221-F and 221-H canyons passes through sand filters before discharge to atmosphere. Performance of these filters has generally been excellent, and their useful life has been much longer than anticipated when installed. Plugging of the sand filter by dust was expected to require installation of new filters within five years; however, actual failure did not occur until about 15 years later, when chemical attack on the concrete support structure permitted sand to fall out of the bed to the air inlet tunnel. The conical depression in the bed, down to the opening in the support structure, breached the filter. This failure of the Building 221-H sand filter in April 1969 permitted a release of activity to the atmosphere, particularly of ^{238}Pu , as summarized in Table 1. The bed was quickly repaired and new bed supports were installed. New filters were subsequently installed for both Buildings 221-F and 221-H and are in service.¹⁸ The original filters, modified to avoid failure of the support structure and monitored for loss of sand, are still in service, in parallel with the new filters. Radioactivity in exit air is monitored continuously.
- The total flow of cooling water in each of the canyon buildings is about 20,000 gpm; most of this water is recirculated through cooling towers (Figure 2). This water flows through coils in vessels that contain highly radioactive solutions. The original design concept for protecting the water from contamination in the event of leaks was to maintain pressure inside the coils higher than in the surrounding solution. In a few cases, operating errors or equipment failure permitted activity to reach the cooling water and exit from Building 221-F. Effects of these releases, along with others from a variety of causes, are listed in Table 2 and in ERDA-1537, Appendix A, Tables 6 and 10.³ The most serious release of activity to cooling water occurred in 1960 due to operating errors. No radioactivity was released outside of Building 221-F from this incident, but cleanup of contamination inside the building cost about \$250,000.

TABLE 1

Releases of Radioactivity to the Atmosphere, 1954-1975

Year	<i>Totals from Buildings 221-F and 221-H, Ci</i>								
	$^3\text{H}^a$ ($\times 10^4$)	^{14}C	^{85}Kr ($\times 10^5$)	$^{103,106}\text{Ru}$	^{129}I	^{131}I	^{137}Cs	^{238}Pu	^{239}Pu
1954	0.022	-	<i>b</i>	-	-	-	-	-	-
1955	0.95	35		25.1	0.21	69	1.347	-	2.65
1956	1.9	35		6.3	0.21	1580	0.239	-	0.031
1957	1.5	35		1.2	0.21	290	0.063	-	0.043
1958	1.1	56		0.6	0.21	17	0.026	-	0.021
1959	4.3	56		10.0	0.21	160	0.138	-	0.024
1960	3.8	56		8.8	0.21	7.3	0.124	-	0.071
1961	3.3	56		3.9	0.21	160	0.048	-	0.012
1962	6.1	56		2.8	0.21	16	0.037	-	0.011
1963	4.9	56		3.9	0.21	4.8	0.025	-	0.0035
1964	4.7	46		2.6	0.21	10	0.067	-	0.0056
1965	2.9	46		2.8	0.21	17	0.018	-	0.017
1966	4.1	46		4.6	0.21	32	0.048	-	0.010
1967	3.0	46		0.4	0.21	20	0.018	0.00026	0.010
1968	4.7	36		21.3	0.21	22	0.040	0.0010	0.057
1969	2.4	36		13.7	0.21	36	0.085	0.56	0.060
1970	2.9	36		1.3	0.21	34	0.042	0.021	0.0089
1971	2.5	36	6.4	5.7	0.21	27	0.009	0.021	0.0082
1972	2.4	36	6.1	7.3	0.21	2.7	0.024	0.017	0.0038
1973	2.9	36	7.7	3.21	0.21	1.8	0.003	0.022	0.0015
1974	1.9	33	5.0	0.2	0.17	1.9	0.001	0.0047	0.0029
1975	2.1	27	5.3	0.04	0.14	0.11	0.001	0.0020	0.0005
Total	64.0	880	30.5	125.7	4.3	2500	2.403	0.65	3.00

a. Calculated values.

b. ^{85}Kr not reported prior to 1971 because of security classification.

TABLE 2

Releases of Radioactivity to SRP Effluent Streams, 1954-1975

Year	<i>Totals from Buildings 221-F and 221-H, Ci</i>					
	³ H Migration from Seepage Basins	⁹⁰ Sr Migration from Seepage Basins	⁹⁰ Sr Discharge to Stream	⁹⁰ Sr Total	¹³⁷ Cs Desorbed in Stream ^a	²³⁸ Pu Discharge to Stream
1954	b					
1955	b					
1956	b					
1957	b					
1958	400					
1959	800					
1960	1,600					
1961	2,000					
1962	1,700					
1963	2,700					
1964	4,700	b				
1965	5,600	0.16		0.16		
1966	4,600	0.57		0.57		
1967	5,600	0.20		0.20	32	
1968	5,790 ^c	0.44 ^c		0.44	2.0	
1969	6,089	0.47		0.47	1.9	
1970	6,987	0.48	0.056	0.54	0.5	
1971	11,843	1.68	0.102	1.78	1.2	0.0301
1972	8,847	1.16	0.044	1.20	1.0	0.0504
1973	10,996	1.62	0.049	1.67	0.5	0.0223
1974	8,906	1.20	0.02225	1.22	1.0	
1975	8,880	1.07	0.01269	1.08	0.3	
Total	98,042	9.05	0.29	9.33 ^d	40.4	0.1028

a. Cesium reached the stream when aqueous waste entering Waste Tank 9 leaked from the inlet riser in May, 1967. Cesium sorbed on clay sediment in the stream has desorbed as shown.

b. No routine analyses were made prior to 1958 for tritium, and prior to 1965 for ⁹⁰Sr.

c. Weirs were installed in 1968 to permit more precise measurement of migration from F-Area and H-Area seepage basins.

d. Column totals inconsistent because of round-off.

- Releases of iodine-131 from fuel reprocessing in Buildings 221-F and 221-H are controlled principally by storage of the fuel before processing. Total releases from SRP operations are tabulated in ERDA-1537, Appendix A,³ and are given for fuel reprocessing facilities in Table 1. Emphasis on production at minimum cooling times of 90 days caused releases of 1580 Ci of iodine-131 in 1956 and 290 Ci in 1957. Releases since 1959, with longer cooling of fuel, have all been below 100 Ci per year, except for 1961. In 1961, irradiated fuel that had a very short cooling time was accidentally transferred to the reprocessing plant and caused release of 153 Ci of iodine-131 to the atmosphere in about 4 days. Results of environmental monitoring and estimates of potential radiation exposures offsite from this release were reported by Marter.¹⁹ Consumption of milk in the area of highest deposition could have resulted in a thyroid dose up to 1.2 rem for a child; however, the total release during 1961 was 160 Ci and the National Radiation Council limit of 1.5 rem/yr was not exceeded. Procedural and monitoring improvements were made to prevent any further dissolution of short-cooled materials.

Iodine Retention in Fuel Reprocessing

In the processing of irradiated fuel and targets, radioiodine is one of the more volatile fission products evolved during processing. SRP has experienced radioiodine evolution from processing of three types of fuel materials; uranium targets that produce weapons-grade ^{239}Pu , enriched uranium fuel, and ^{237}Np targets that produce ^{238}Pu . Only for the first of these, uranium targets processed in Building 221-F, would iodine behavior have much similarity to power reactor fuel processing. In the other two processes, mercury added to permit dissolving aluminum in nitric acid retains iodine in solution. Because the Purex process does not use mercury, iodine is more volatile from Purex solutions.

The fraction of the iodine, in material charged to a dissolver, that is not released to the atmosphere can be expressed as a retention factor ($\text{RF} = I_{\text{input}}/I_{\text{effluent}}$). The iodine retention factor is highest for neptunium processing and lowest for uranium-target processing.

Iodine release from the Purex process is controlled during dissolving at SRP by a silver nitrate reactor in the dissolver off-gas system that effectively retains any iodine volatilized from the dissolver. Early experience showed, however, that only about half of the iodine is volatilized during dissolving; the

remaining half is partially volatilized in subsequent process steps including solvent extraction, solvent recovery, and evaporation of acidic wastes. Iodine is controlled by storing the fuel for a sufficient period before processing; the minimum storage time for Purex feed material is 200 days. The iodine release from SRP has exposed the offsite population within 100 km of SRP to 64 man-rem during the 1954-1975 period, as compared to an exposure of 1.72 million man-rem from natural sources to the same population.

In the processing of uranium targets, iodine release from the Purex process was studied in 1967 by S. R. Smith²⁰ and the following information is from that study. In normal processing of two batches of uranium, measurements showed that iodine retention factors were 10 and 6, for an average of 8. In one test, the fuel to be dissolved contained 26.9 Ci iodine-131; evolution to atmosphere from all subsequent process steps was 2.7 Ci. In the second test, 21.6 Ci were charged and 3.65 Ci were released.

Much greater retention of iodine in neptunium processing was measured in plant tests in 1967 that showed a retention factor of 530.²¹ Of 1600 Ci of iodine-131 charged to the neptunium dissolver, 3 Ci was released during all process steps: dissolving, anion exchange, waste transfer, and waste recovery. As previously stated, the high retention factor is attributed to mercuric ion, approximately 0.005 molar, present during the neptunium dissolution process.

PLUTONIUM PURIFICATION AND HANDLING EXPERIENCE

Plutonium Production

The primary mission of the SRP plutonium production facilities is to produce plutonium metal. Whereas the proposed power reactor production of plutonium is as the oxide, the initial plutonium facilities in both SRP separations plants used the plutonium peroxide precipitation method, with subsequent conversion to PuF₄, and reduction to metal with calcium. A modified process was installed in Building 221-F in 1959 to produce plutonium metal by the PuF₃ precipitation route.⁵ Plutonium oxide for various special programs has also been produced in both reprocessing plants by precipitation and calcination of plutonium oxalate. Some of the materials have been fabricated into aluminum matrix fuels, both as alloy and as compacted PuO₂-aluminum powder blends.

Isotopic composition and chemical form of the plutonium determines fission properties, specific alpha activity, neutron production rate, and daughter growth rate - all of these being

important to requirements for nuclear safety, radiation protection, and contamination control. Normal SRP Purex plutonium is weapons grade that is relatively low in higher isotopes compared to reactor grade plutonium. Weapons-grade plutonium requires more restrictive nuclear safety limits than fuel-grade plutonium, which has a lower fissile content ($^{239}\text{Pu} + ^{241}\text{Pu}$) as well as more ^{240}Pu , a neutron poison. Because of high neutron production from the α, n reaction on fluorine, the plutonium fluoride present in the SRP metal production line determines the shielding requirements to limit radiation doses in the operating areas. The α, n reaction with fluorine is sufficiently higher than the α, n reaction with oxygen that the dose rate from the weapons-grade plutonium fluoride is several times that from fuel-grade plutonium oxide, the factor depending upon specific isotopic composition. As a result, material in the SRP process lines requires greater nuclear safety precautions and more radiation protection than would fuel-grade plutonium oxide.

In addition, SRP has conducted special plutonium production programs to produce higher plutonium isotopes for reactor experiments, and to produce ^{244}Cm and ^{252}Cf . These programs have resulted in processing campaigns that cover isotopic compositions ranging across the spectrum from high purity ^{239}Pu to ^{242}Pu . These programs have demonstrated satisfactory operation of the SRP facilities with materials whose properties encompass those of power reactor plutonium.

Approximately 600 kg of plutonium high in ^{240}Pu has been processed in plutonium finishing facilities since startup to provide feed for production of special fuel and target assemblies.

Approximately 70 kg of PuO_2 high in ^{242}Pu has also been produced in these facilities and fabricated into reactor targets for production of ^{252}Cf and special plutonium isotopic compositions.

Plutonium Recovery

The SRP plutonium metal production facilities include a small processing complex to recover and purify plutonium from solutions and solid residues generated by the primary process. Feeds to recovery include nitrate and fluoride solutions, slag and crucibles from the metal reduction step, miscellaneous sweepings from process cabinets, and off-standard metal and oxide. In addition, the complex has been used to recover large amounts of plutonium returned from offsite (including metal production residues from Los Alamos, Hanford, and Rocky Flats),

material from experimental reactor programs, and miscellaneous scrap. These materials have covered a wide variety of isotopic compositions, chemical and alloy forms, and impurities. Standard procedures include dissolution where possible in nitric acid, dissolution of refractory plutonium oxide in nitric acid - hydrofluoric acid mixtures, and dissolution of plutonium metal in sulfamic acid.

The plutonium may be recovered and purified by ion exchange techniques and solvent extraction, and is finally converted to metal (or oxide) in the main processing facilities. The materials processed in SRP recovery operations appear to range in isotopic composition and chemical and physical form beyond any that might be produced in commercial reprocessing operations and fuel fabrication.

^{238}Pu Processing

The specific alpha activity of commercial plutonium is substantially higher than normal SRP plutonium, but less than that of ^{238}Pu , which is produced at SRP for use as a heat source in thermoelectric generators. ^{238}Pu is separated from irradiated ^{237}Np in auxiliary facilities within the remote process area of the enriched uranium separations plant and is converted to the oxide by the plutonium oxalate process. The specific alpha activity of normal heat source ^{238}Pu (80% ^{238}Pu - 20% higher isotopes) is about 200 times that of pure ^{239}Pu and 40 times that of nominal fuel-grade plutonium. Although the $^{238}\text{PuO}_2$ production facility is relatively small in size and throughput (25 to 50 kg/yr), SRP experience has successfully confirmed that design bases and operational control methods are adequate for processing plutonium of high alpha activity.

Fuel Fabrication

As noted previously, plutonium has been fabricated into aluminum-matrix fuel elements for irradiation in SRP reactors. Most of this fuel was made by melting aluminum metal and plutonium metal together, casting and machining a tubular alloy billet, sealing the billet in an aluminum can, and extruding the assembly to produce a tube clad in aluminum on both sides with a thin Pu-Al core. Approximately 600 kg of plutonium has been successfully handled in this manner in operations involving high temperatures, molten metals, machining and associated machine scrap handling, welding, and extrusion operations. In addition, some reactor elements have been made from blended plutonium oxide powder and aluminum powder which was pressed into compacts and sealed in

aluminum cans. These SRP fabrication operations do not directly correspond to fabrication operations projected for commercial facilities and they differ in scale; however, they do confirm that plutonium can be contained and handled without excessive exposure in a variety of unit operations.

Operating Philosophy and Criteria

Plutonium is a highly toxic material, and careful consideration of the hazards involved is necessary for safe handling. The basic requirements for safe handling are (1) contamination control to prevent uptake by personnel or release to the environs; (2) radiation control to prevent overexposure of personnel to neutrons, gamma rays, and X-rays; (3) criticality control to prevent accidental accumulation of a critical mass; and (4) fire control to prevent loss of containment and release of plutonium to the environment in the event of a major process fire. Because of the high specific alpha activity of ^{238}Pu , containment requirements are more stringent for facilities handling ^{238}Pu , but the basic contamination control principles used for ^{239}Pu are applicable. The high neutron emission rate for ^{238}Pu also requires handling small batches (100 to 200 g) in shielded glove boxes or manipulator cells for effective radiation control. Criticality controls generally are not required for ^{238}Pu except in the solid form, because the minimum critical mass for ^{238}Pu in aqueous solutions is quite large.

Contamination Control

All SRP operations involving plutonium processing are performed in ventilated cabinets or glove boxes which are maintained at a negative water pressure of 0.5 to 0.8 in. with respect to the surrounding room. Operating areas are compartmentalized with air locks at entrances and exits to minimize spread of contamination and interference with other operations in the event of a release of plutonium. Where possible, operations with a higher probability of release of contamination such as maintenance work, sampling, bag-out operations, etc., are separated from those with less risk through the concept of separate operating and maintenance rooms. Where possible, process solutions are transferred by vacuum or gravity to minimize the potential for high pressure sprays in the event of piping leaks. Transfer piping between cabinets is enclosed in ducts which are in effect extensions of the process cabinets.

Personnel working in glove ports wear rubber surgical gloves as additional protection against failure of a cabinet glove, and monitor their hands each time they remove them from the glove ports. Where the potential for cutting or puncturing is high, leather gloves are worn over cabinet gloves. Use of glass in cabinets is generally not permitted. Small glass vials, one of the few exceptions, are coated with plastic before use. Respiratory protection is worn and constant surveillance by trained Health Physics personnel is provided when any operation involves significant potential for contamination release.

Cabinets or glove boxes for handling plutonium in powder form are designed for minimum leakage. Both supply and exhaust air are filtered. Exhaust air from all process cabinets receives a minimum of two stages of filtration, and the last stage of filtration is located outside the facility to minimize the potential for filter ignition in the event of a fire in the facility. The present exhaust system is being revised to allow the sand bed filter to be the final stage of filtration for the entire facility exhaust (room and cabinet) for maximum protection against release in the event of a fire.

Radiation Control

Solvent extraction operations in the canyon are controlled to minimize radiation from residual fission products in the plutonium before transfer to finishing operations. Glove port work is minimized by extending valve handles through cabinet panels and by conveyor systems and mechanical or hydraulic lifts and devices that can be operated from outside the cabinet. Where direct hand operations are required, lead-loaded gloves, tongs, and tools are used to reduce hand exposure. Plutonium inventory in cabinets is kept to a minimum. Cabinet housekeeping standards are maintained at a high level to minimize background radiation. Cabinet sumps are cleaned by periodically flushing with acid. Loose powder is swept or vacuumed up from dry cabinets. Neutron and gamma shielding are also provided at areas where radiation is high.

Criticality Control

Strict adherence to procedures is required in handling fissile materials. Nuclear safety is ensured principally through use of dimensionally favorable equipment (approximately 4-in.-wide rectangular tanks, which are safe for relatively large masses of plutonium), and by intensive accounting for the plutonium throughout the process. Finished product is transported only in special carriers which guarantee safe spacing. Plutonium storage areas provide physical barriers to ensure proper spacing for criticality control.

Operating limits and conditions are set so that more than one error is required before a safe condition can deteriorate into a nuclear hazard. Supervisory review is required at certain critical steps in the process to ensure that observed values agree with expected values. If discrepancies are detected, operations are halted until the discrepancies are resolved. Allowances for measurement errors are included in setting operating limits. To supplement material balance data, neutron counters and gamma pulse height analyzers are installed to monitor vessels which have a potential for plutonium accumulation.

Fire Control

Prevention is the primary fire control measure. Combustible materials of construction in process cabinets and process areas are kept to a minimum. Very high housekeeping standards are maintained to prevent the accumulation of materials which could become a source of fuel in the event of a fire. Cellulose materials, such as cotton rags and wipes, in process cabinets, are strictly controlled. Each cabinet where plutonium metal is handled contains a beaker of MgO sand for use as an extinguisher in the event of spontaneous ignition. To minimize the fire hazard, maximum in-process inventory of plutonium metal is limited to approximately 10 kg, and all plutonium metal except that in undumped reduction crucibles is removed from the process cabinet whenever the facility is to be unattended. In the event a fire should break out, each process room and cabinet is equipped with automatic fire detection and Halon fire suppression systems.

Plutonium Throughput

The plutonium metal finishing line has greater capacity than the remotely operated plutonium separation and purification facility from which feed solution is received. Offsite metal scrap can be dissolved at 1 to 2 kg/day for recovery in fuel reprocessing facilities.

Throughput in commercial facilities will probably be substantially higher than present SRP operations. However, nuclear criticality considerations basically determine the capacity of individual pieces of equipment and process lines. When higher throughputs are needed, equipment is replicated rather than scaled up. The result is that larger plutonium requirements can be satisfied by duplication of existing facilities and do not require an extension of technology. The principle is illustrated by the SRP final plutonium processing system, which has, for example, four cation exchange columns for plutonium concentration, two plutonium fluoride precipitators, four drying stations, and two metal reduction furnaces.

Final ^{238}Pu purification is performed by ion exchange. Throughputs up to 4 kg/mo have been attained. Normally, 20 to 30 kg of ^{238}Pu per year are produced by ^{237}Np target irradiation in SRP reactors. The remaining capacity is utilized for recovery of ^{238}Pu scrap generated offsite and reprocessed at SRP as required.

Product Shipment

The plutonium metal "buttons" produced in Building 221-F are shipped by the USERDA in shipping containers that are designed to withstand fire, impact, and puncture. Plutonium-238 oxide produced in Building 221-H is shipped in different containers which have the same design performance requirements. Both types of containers are certified or licensed by Federal agencies. No significant accidents have occurred to date with either type of shipment.

Unusual Incidents

No accidents involving major expense or radiological hazard have occurred in SRP plutonium purification and handling. Containment of radioactive particulates by filtration of exit air from process areas requires special attention during design of plutonium facilities. The requirements have been described in a previous section (Contamination Control) and their application in design of the high capacity facilities installed in Building 221-F in 1957-1959 was described by Marter.²² Operation of Buildings 221-F and 221-H with respect to release of plutonium to the atmosphere has been improved markedly since initial operation. As shown in Table 1, 2.65 Ci of ^{239}Pu was released in 1955; but improved filtration installed in December 1955 has maintained subsequent annual releases of ^{239}Pu below 0.1 Ci/yr. Although ^{238}Pu has a specific activity about 200 times that of ^{239}Pu , proper filtration of exit air has maintained the annual release of ^{238}Pu below 0.025 Ci; except in 1969, when the H-Area sand filter failure occurred, and 0.56 Ci of ^{238}Pu was released.

SPENT FUEL RECEIPT AND STORAGE

Mission and Goals

Each of the SRP reactor areas contains a fuel cooling and disassembly basin as an integral part of the reactor building. The basins can store about 6000 irradiated fuel assemblies, in addition to other reactor components such as housing components and control rod assemblies. The basins are used to receive the irradiated fuel assemblies when they are discharged from the reactor, store them underwater in critically safe arrays where

they are shielded and cooled, and provide a facility for dis-assembling the uranium fuel from its top and bottom fittings and other support components. The fuel is stored underwater until transported in casks by rail to the fuel processing areas. The reactor basins are not part of the fuel processing areas; their counterparts in the LWR facilities are the spent fuel storage basins located in or near the LWR buildings.

The receiving basin for offsite fuels (RBOF) is located in the fuel processing area. This facility is used for underwater manipulation of casks and irradiated nuclear fuels received from various offsite reactors and, occasionally, from SRP reactors. These spent fuels may be prepared for shipment to either of the SRP separation plants for processing or they may be stored in the RBOF for long periods of time. Spent fuels from offsite are delivered by truck or railcar in casks certified or licensed by Federal agencies.

The mission of the RBOF facility is similar to the fuel receiving and storage facilities of a commercial fuel reprocessing facility. However, most fuels handled at RBOF have not received as much reactor exposure as the LWR-type fuels expected at a commercial reprocessing site. Also, the rate of fuel shipments to RBOF is less than that expected at a commercial reprocessing site.

Operating Philosophy and Criteria

The administrative control procedures for the RBOF are implemented through several levels of documentation and of approval.

Included with, but not a part of, the RBOF Technical Standards is a numbered Nuclear Safety Data Sheet (NSDS) that is written and approved for every fuel element stored in the facility. Also, an NSDS is written and approved before any fuel position or status is changed after it arrived at the RBOF. Changes include such operations as cutting or repackaging. These sheets describe the fuel type (or class), composition, dimensions, amounts of fissile material, type of cladding, and secondary containment, if any. Nuclear criticality calculations set limits on the safe amount in the worst configuration so that storage limits can be specified. Data references are included. The technical standards and NSDS's provide the basis for safe operating decisions in the RBOF.

Prior SRP evaluation and ERDA-Savannah River Operations Office approval are required before any offsite reactor fuel may be shipped to the RBOF. An exact written description of each

cask and its contents must be submitted to permit review and approval of the shipment in terms of nuclear criticality, potential operating problems, and compatibility with subsequent processing, or long-term storage.

All operations are performed according to written procedures. The operating manual includes operating logsheets for work with each of the various types of casks and fuels. Prior to the arrival of a given cask and its contents, copies of the appropriate logsheets are selected and bound by field supervision in the Production Department and the Works Technical Department. Thus, only the step-by-step procedures in effect from the time of arrival of the cask to the final disposition of the fuel and the reshipment of the cask are included in the bound volume.

Operating Experience

SRP has received, stored at RBOF, and processed a wide variety of fuels. Fuel receipts at RBOF are not typical of what a commercial reprocessor would receive; RBOF receipts include many fuel types in relatively small numbers. Enriched uranium is received from reactors licensed under the Atomic Energy Act or foreign reactors fueled with material produced or enriched by the United States if it has been determined that chemical processing services are not available from commercial fuel processors in the United States at reasonable terms and charges. During the peak of such receipts in 1966 and 1967, SRP received more than 5000 fuel assemblies in about 200 casks from five countries. The design bases for RBOF were reported by Smiley,⁶ and details of operating experience by Rogers.²³ No major radiological event or significant release of activity has occurred at the RBOF since startup at the end of 1963. Neither has there been a serious industrial accident.

Onsite Transport of Fuel

Buildings 221-F and 221-H receive irradiated materials from both RBOF and SRP reactor cooling basins; several unusual incidents have occurred in the fuel receiving and handling operations in the canyons. These incidents fall into two broad categories: damage inflicted only to rail stock while fuel is contained in a cask, and loss of positive control during the transport and storage of fuel bundles or buckets.

Two cases of cask car derailment have occurred outside the canyon tunnel at the reprocessing plants. In no case did the derailments result in fuel damage or in the loss of contaminated water to the environment. The cask cars remained upright and

damage occurred only to the tracks. One cask car was derailed within the tunnel as a result of ramming a rail stop too hard. This damaged both the cask car and the rails, but again no activity escaped. Seven other cases of damage to shielding, air lock doors, or to rail stock have occurred, but the cask cars remained on the tracks.

The most common loss of positive control during canyon fuel transport and storage is that of dropping the fuel, which has occurred about once per year. In no case was the fuel significantly damaged or activity released outside the controlled facilities.

EMPLOYEE RADIATION EXPOSURE AND SAFETY EXPERIENCE

Exposure Guides

Until 1959, SRP limited the annual occupational exposure to the values recommended by the NCRP, 5 rem per year of penetrating external radiation and 10 rad per year skin dose. In 1959, a plant control guide of 3 rem per year whole body exposure was established. The guides were amended over the years in accordance with NCRP recommendations and AEC/ERDA standards.²⁵ Present radiation exposure guide values are shown in Table 3. These guides

TABLE 3

Radiation Exposure Guides for SRP

<i>Mode of Exposure</i>	<i>Dose, rem</i>	
	<i>Per Qtr</i>	<i>Per Yr</i>
Occupational Exposure		
Whole body, head and trunk, active blood forming organs, gonads, lens of eye, red bone marrow	3	3
Skin, other organs, tissues, and organ systems (except bone)	5	15
Bone and forearms	10	30
Hands and feet	25	75
	<i>Dose, rems</i>	
	<i>Plant</i>	<i>Life</i>
	<i>Emergency^a</i>	<i>Saving</i>
Emergency exposure		
Whole Body	25	100
Hands	100 ^b	200 ^c

a. Involving protection of property or personnel.

b. Includes whole body exposure.

c. In addition to whole body exposure.

are identical to the current ERDA values, with the exception of whole body dose where the ERDA value is 5 rems per year. Expectant mothers are limited to 0.5 rem during the gestation period.

It is SRP policy that radiation exposure guides are not considered desirable dose commitments; instead exposures to employees are kept as low as reasonably achievable.

Occupational Exposure Experience

A summary of the occupational radiation dose for the period 1965 through 1976 is shown in Table 4 for personnel in SRP uranium and plutonium processing facilities. This summary excludes those employees whose jobs involve no potential occupational exposure other than background.

The annual average occupational radiation dose per monitored employee ranged from 0.32 to 0.69 rem for the 12-year period. The maximum individual dose ranged from 2.5 to 3.1 rems.

Control of radiation doses to plant employees to the lowest practical levels is accomplished by operating procedures and job plans that contain detailed instructions pertinent to the control and reduction of personnel exposure. Exposure summaries are reviewed monthly by supervision to identify trends and problem areas, and operating procedures are revised where necessary. A permanent record of each employee's radiation exposure is maintained.

Assimilation of Transuranic Nuclides

Radiation workers in uranium and plutonium processing facilities with confirmed uptakes of transuranic nuclides are listed in Table 5 by percentage of body burden based on permissible dose to the critical body organ. Urine samples are collected from transuranium workers periodically for routine analysis. Additional bioassay samples are requested immediately following known or suspected exposures to radionuclides. An uptake is confirmed by two consecutive positive bioassay samples. Assimilations greater than 10% of a body burden will be detected in routine bioassay samples. Uptakes of less than 1% of a body burden will be confirmed by urinalysis if special samples are requested soon after a contamination incident. Inhaled, relatively insoluble, transuranic nuclides will be detected by routine urinalysis if the lung burden exceeds 50% of the permissible dose rate. A lung burden will also be detected by routine semiannual in-vivo chest counts of transuranium workers or by special counts following an incident.

TABLE 4

Whole Body Occupational Dose to SRP Personnel Processing Uranium and Plutonium

<i>Year</i>	<i>Dose, rem</i>		
	<i>Total Exposure</i>	<i>Average per Badged Person</i>	<i>Maximum Individual Exposures</i>
1965 ^a	916	0.61	2.8
1966	928	0.62	3.1
1967	980	0.66	3.0
1968	829	0.57	2.9
1969	994	0.69	2.9
1970 ^b	868	0.63	2.6
1971	815	0.52	2.8
1972	685	0.39	2.9
1973	742	0.46	2.7
1974	720	0.43	2.9
1975	570	0.32	2.7
1976	677	0.34	2.5

a. Occupational dose data prior to 1965 are available only on a plantwide basis.

b. Use of thermoluminescent dosimeter (TLD) crystals for measuring external ionizing radiation began in April, 1970. Film badges and direct monitoring were used previously.

TABLE 5

Confirmed Employee Uptakes of Transuranic (TRU) Nuclides
During Uranium and Plutonium Processing

<i>Maximum Permissible Body Burden (MPBB), %</i>	<i>Number of Workers</i>
<10	325
>10 to <25	34
>25 to <50	14
>50 to <100	1 ^{a,b}
>100	0
Total:	<u>374</u>

-
- a. A Maintenance mechanic accidentally punctured his right index finger with a sampler needle used for ^{238}Pu nitrate solutions, resulting in a contaminated injury on 11/29/68. Excision and chelation therapy by plant physicians removed greater than 96% of the deposited plutonium. The body burden in the employee is estimated at 0.03 $\mu\text{Ci } ^{238}\text{Pu}$ (approximately 80% MPBB). The annual bone exposure in 1969 was 27 rem. A two-year case history of this injury was reported by Jolly, et al.²⁶
- b. The processing of ^{238}Pu is not within the scope of commercial reprocessing plant activities. Had this type accident occurred with ^{239}Pu or in a commercial plant, the uptake would have been less than 10% of a MPBB.

Safety Experience

Major and submajor injury experience in the SRL fuel re-processing areas is given in Table 6 for 23.5 years of operation (5/53 to 12/76). Fourteen major injuries and one fatality have occurred. The fatality occurred during a normal railroad car switching operation. While Table 6 presents the total number of injuries that have occurred, it is more meaningful to compare injury frequencies. The SRP major injury frequency since startup, per million manhours worked, is 0.26, which may be compared with 1973 National Safety Council frequencies of 10.55 for all industries, 35.4 for the coal mining industry, and 4.25 for the general chemical industry.

No major injuries have occurred in the fuel reprocessing areas since 1973. Over 26 million manhours have been worked in F-Area and H-Area since the last major injury. If the chemical industry frequency rate had prevailed in these areas, over 100 major injuries would have occurred.

RADIOACTIVE EFFLUENT RELEASES

Limits and Guides

Releases of radioactivity to the environs as a result of SRP operations, including the reprocessing and plutonium recovery facilities, have been controlled in accordance with NCRP and ICRP recommendations and AEC/ERDA standards (e.g., ERDA Manual, Chapter 0524.²⁷)

This has been accomplished by establishing technical standards for plant releases and a system of operating guides for individual SRP facilities. The initial design criteria and subsequent operation of the reprocessing facilities provided the capability to limit releases well within the AEC/ERDA standards and permit periodic lowering of the technical standards and operating guides. The status of the SRP release guide system was reported in 1967 by Evans, Marter, and Reinig.²⁵

In 1972, the guides for radiation dose to offsite populations was reduced (to the values shown below) from values equal to AEC standards.

The annual exposure to an individual in the offplant population caused specifically by release of radioactivity from SRP shall not exceed the limits in Table 7.

These limits do not in any way modify the guides stated in the ERDA Manual, Chapter 0524,²⁷ or those recommended by the ICRP, NCRP, and FRC for control of population exposure. The recommended numerical

TABLE 6

Industrial Safety Experience in Separations Areas (5/53 to 12/76)

<i>Injury Classification</i>	<i>F-Area</i>	<i>H-Area</i>	<i>Fatalities</i>
Major ^a	9	5	1 ^b
Submajor ^c	33	23	

- a.* A major injury is defined as an on-the-job injury which results in death, loss of time, or any degree of permanent disability whatsoever which interferes with normal physiological function. If an employee is unable to return to work at an established job on the day following the injury, whether scheduled or not, the injury is classified as major.
- b.* During normal railroad car switching movements in H-Area, an employee fell under a moving train (3-11-67).
- c.* A submajor injury is a case, less severe than major, which interferes with the employee's ability to perform his regular duties because of severity of the injury or duration and frequency of treatment.

TABLE 7

Limits on Annual Exposure to an Individual
Offplant from Radioactivity Released by SRP

<i>Type of Exposure</i>	<i>Dose Limit, mrem/yr</i>
Whole body	10
Gonads	10
Bone marrow	10
Gastrointestinal tract	30
Bone	30
Thyroid	30
All other organs	30

TABLE 8

Releases of Radioactivity to Seepage Basins, 1954-1975

Year	<i>Combined F-Area and H-Area Releases, Ci^a</i>						
	³ H ^b	⁹⁰ Sr	^{103, 106} Ru	^{134, 137} Cs	¹⁴⁷ Pm	²³⁸ Pu	²³⁹ Pu
1954	c						
1955		0.67	14	0.7	1.4		0.11
1956		1.0	23	1.6	3.6		0.56
1957	6.0 × 10 ³	9.8	49	2.1	2.5		0.11
1958	5.5 × 10 ³	3.0	28	1.2	3.2		0.49
1959	1.2 × 10 ⁴	2.7	140	1.5	14		0.53
1960	1.3 × 10 ⁴	4.9	160	8.0	21		0.26
1961	2.2 × 10 ⁴	5.5	53	21	6.4		0.28
1962	3.1 × 10 ⁴	3.8	120	65	14		0.60
1963	3.5 × 10 ⁴	10.0	120	24	14		0.37
1964	2.8 × 10 ⁴	2.7	230	20	31		0.16
1965	2.9 × 10 ⁴	6.6	100	21	13		0.61
1966	3.7 × 10 ⁴	1.8	95	17	4.9		0.65
1967	2.0 × 10 ⁴	1.9	60	22	2.4	0.31	0.77
1968	3.5 × 10 ⁴	2.4	98	29	7.1	0.52	0.40
1969	2.3 × 10 ⁴	3.3	53	29	6.6	0.11	0.49
1970	3.2 × 10 ⁴	2.7	51	15	3.6	0.55	0.35
1971	1.9 × 10 ⁴	1.7	39	9.6	2.2	0.33	0.21
1972	2.2 × 10 ⁴	0.96	35	9.2	1.8	0.40	0.24
1973	3.2 × 10 ⁴	0.82	39	7.5	2.0	0.19	0.094
1974	1.6 × 10 ⁴	0.30	26	8.1	1.1	0.10	0.077
1975	1.4 × 10 ⁴	0.79	6.5	7.7	1.3	0.15	0.031
Total	4.2 × 10 ⁵	76.2	1540	320	160	2.7	7.4

a. Includes releases from Building 221-F, Building 221-H, RBOF, and waste evaporation.

b. A portion of this tritium evaporates from the seepage basins and is included in the previous table of atmospheric releases.

c. Existence of fission product tritium discovered in 1957.²⁸

limit of standard-setting organizations is 170 mrem per year for the average dose to the whole population. This limit is accepted as the basic radiation protection criterion for control of public exposure. The limits cited above indicate SRP's objective to keep offsite exposures as far below the recommended criterion as practicable.

To provide surveillance and control of emission, the releases of radioactivity at SRP are compared monthly to prorated values of annual operating guides. The operating guides are set very close to historical plant release quantities that reflect the minimum practical levels. SRP policy, as stated in the guides, is that "the plant will confine radioactivity as completely as practical rather than release it to the environment. Release guides are not to be considered desirable discharge quantities."

Because the guides are set close to actual experience, they are occasionally exceeded even on an annual basis; the guides are reviewed at least annually and are revised if production requirements or process improvements cause such a revision to be justified. More frequently, the guides are exceeded only for a short time. These occasions focus attention on problem areas so that corrective action may be taken.

Release History - Airborne, Liquid

Radioactive releases from the fuel processing operations at SRP separation areas are summarized for the period 1954-1975 in this section. Radioactive materials are released to the environment by the following pathways.

- Releases to the atmosphere by process building ventilation exhaust stacks (Table 1).
- Releases to the seepage basins (Table 8).
- Releases to the surface streams by direct discharge or indirectly by ground-water transport from the seepage basins (Table 2).

Table 2 includes both direct discharge to the principal effluent stream (Four Mile Creek) and the migration from the seepage basins (only tritium and ⁹⁰Sr have migrated to Four Mile Creek to date).

The basis for use of seepage basins at F-Area and H-Area is discussed in detail in ERDA-1537.³ Annual releases to the basins since 1954 are given in Table 8. The seepage basins retain most of the activity in the ground close to the basins. Detailed studies of the hydrology of the separations areas, completed in

H-Area and now in progress in F-Area, show that the Tuscaloosa aquifer, which underlies part of SRP, is isolated from downward percolation of water from the seepage basins (ERDA-1537, p II-152).³

ENVIRONMENTAL MONITORING

A continuous monitoring program has been maintained at SRP since 1951 (before plant startup) to determine the concentrations of radioactive materials in a 1200-square-mile area outside the plant. Included are parts of Aiken, Barnwell, and Allendale counties in South Carolina, and Richmond, Burke, and Screven counties in Georgia.

Results from the monitoring program are published in comprehensive annual reports of the Savannah River Plant's monitoring effort; the reports are distributed annually to state and federal agencies and to the public, e.g., Reference 29.

The continuing offsite monitoring program measures contributions to offplant exposures, through direct radiation, breathing air, deposited radioactivity, and radioactivity in consumed materials (water, milk, fruit, vegetables, grain, fish, fowl, etc.).

Atmospheric Monitoring

Atmospheric radioactivity is measured at 12 monitoring stations near the plant perimeter and 12 stations around a circle of about 25-mile radius from the center of the plant (Figure 9). Four additional air monitoring stations at Savannah and Macon, Georgia, and at Columbia and Greenville, South Carolina, are so distant from SRP that the effect of SRP operations is negligible; these serve as reference points for determining background activity levels. This system permits comprehensive surveillance of radioactivity and also makes it possible to differentiate between fallout and SRP releases.

Water Monitoring

The main effort of the environmental monitoring program is devoted to radioactivity; although chemicals, pesticides, and bacteria are also covered. Stream and river temperatures are also measured as are biological indicators of river health.

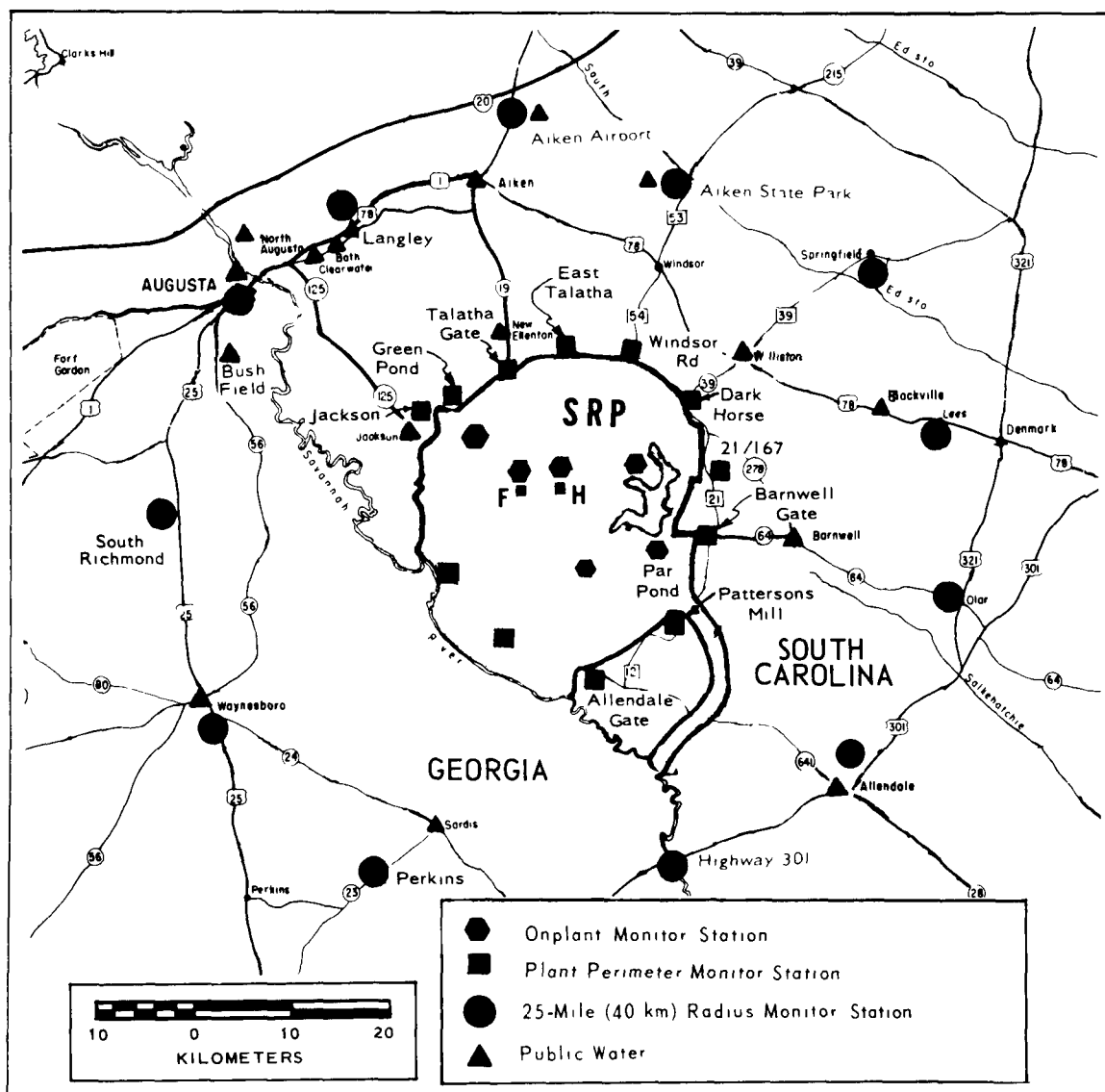


FIGURE 9. Continuous Air Monitoring Stations and Public Water Sample Locations

Radioactivity in plant effluents is measured at the point of release, at intermediate points in surface streams on the plantsite, and in the Savannah River upstream and downstream from SRP. Stream and river sample points are shown in Reference 29. The sampled water is collected weekly and is analyzed for radionuclide content.

Communities near SRP get drinking water from deep wells or surface streams. Public water supplies from 14 surrounding towns are collected twice yearly and analyzed for radionuclide content.

The Beaufort-Jasper Water Authority operates a treatment facility to furnish drinking water, partially obtained from the Savannah River, to most of Beaufort County, South Carolina. Water is supplied through a canal from the river at a location about 90 miles below SRP. A water treatment plant at Port Wentworth, Georgia, supplies water to a business-industrial complex near Savannah, Georgia. These two water supplies are analyzed monthly for tritium content. The principal radioactivity in the Savannah River that comes from SRP is tritium. However, the concentration of tritium below the plant was only 0.13% of that permitted in public drinking water in 1975. Other water-borne radioactivity from SRP contributes only about 0.1% of the permissible concentration in public drinking water.

Vegetation and Food Monitoring

The program for monitoring pasture grass, milk, produce, fish, and animals is described in ERDA-1537³ and in the annual monitoring report.²⁹

Soil Monitoring

The amount of plutonium in undisturbed soils reflects the cumulative deposition from all sources. Onplant and offplant soils have been collected and analyzed for ^{238}Pu and ^{239}Pu . Results reported by McClearen³⁰ and McLendon³¹ show a background deposition level of approximately 2.0 mCi/km^2 offsite and over most of the plant. This level is well within the range of deposition noted in the southeastern United States and indicates that offsite deposits due to SRP operations are small compared to background. Only samples taken within a 2-km radius of each of the two chemical separations areas show higher levels of plutonium. Results of that sampling are summarized in Figure 10.

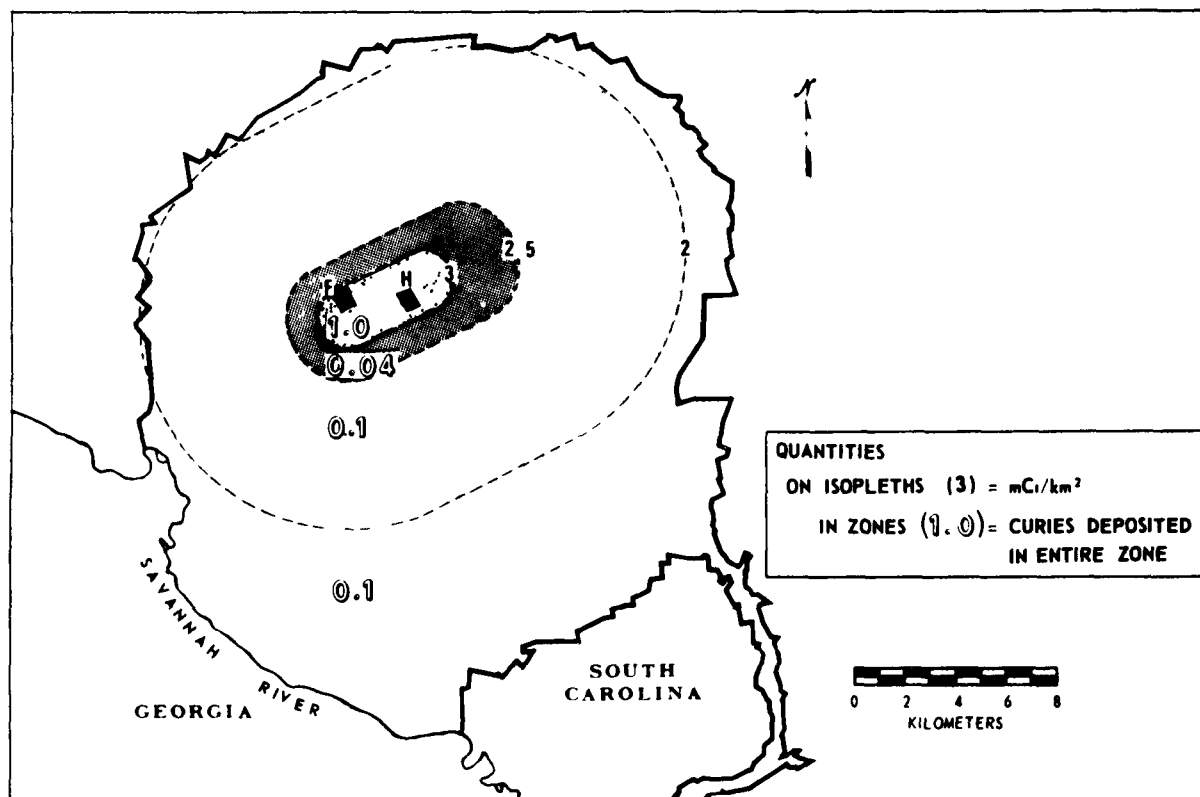


FIGURE 10. Plutonium Deposition on SRP Site

ENVIRONMENTAL EFFECTS

Radiation Dose Commitment to Surrounding Population

As discussed in the previous section, all releases of radioactivity from the separations areas are monitored, including all abnormal releases to the atmosphere. For the total amounts of activity that are released to the atmosphere from all operations at SRP, an annual calculation is made of the potential radiation dose commitment to a person at the plant perimeter and to the population within 100 km of the center of the SRP site. "Dose commitment" means the radiation dose received in a lifetime of 70 years by population groups as a result of a given release of radioactive materials to the environment; it does not include global recycling of noble gases, tritium, or carbon-14. A separate radiation dose commitment calculation is made for releases to streams that affect downstream users of Savannah River water. These calculations are described in Reference 3, p III-27. Reports of total SRP releases to the environment and calculated radiation dose commitment to the surrounding population are issued annually, e. g., Reference 29.

The following tables summarize dose commitments for releases from the fuel reprocessing operations only:

<i>Table</i>	<i>Title</i>
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9	Total Population Whole-Body Dose Commitment from Atmospheric Releases
10	Total Population Whole-Body Dose Commitment from Atmospheric Releases from the Reprocessing Areas
11	Total Population Whole-Body Dose Commitment from Liquid Releases
12	Comparative Whole-Body Radiation Dose Commitment, 1975

For atmospheric releases, the plutonium-uranium fuel processing areas contributed an offsite radiation exposure of 426 man-rem for the years 1954-1975, compared with an SRP total of 6551 man-rem or 6.4%. For liquid releases, these areas contributed 58 man-rem out of an SRP total of 531, or 11%.

The total SRP atmospheric and liquid contributions to offsite radiation dose were 0.15% and 0.19%, respectively, of the radiation dose from natural sources.

TABLE 9

Total Population Whole-Body Dose Commitment from Atmospheric Releases

<i>Period</i>	<i>Population Size^b</i>	<i>Dose Commitment, man-rem^a</i>			
		<i>Natural Sources</i>	<i>Artificial Sources</i>	<i>All SRP Sources^c</i>	<i>Fuel Processing Sources</i>
1975	668,000	78,000	71,000	115	10.8
1954-1975	668,000	1,720,000	1,560,000	6,651	426

a. From all SRP operations.

b. Based on 1970 census.

c. Dose commitments do not include the evaporation of tritium from seepage basins and waste tanks, estimated to be 6.0 man-rem for 1975.

TABLE 10

Total Population Whole-Body Dose Commitment from Atmospheric Releases from the Reprocessing Areas

Dose Commitment, man-rem ^a

Year	³ H	¹⁴ C	⁸⁵ Kr	¹⁰³ , ¹⁰⁶ Ru	¹²⁹ I	¹³¹ I	²³⁸ Pu	²³⁹ Pu	Total
1954	0.07	0	b	-	-	-	0	-	0.07
1955	3.0	4.8	-	0.003	0.4	1.5	0	24.6	34.3
1956	6.0	4.8	-	0.0007	0.4	35.0	0	0.29	46.5
1957	4.8	4.8	-	0.0002	0.4	6.5	0	0.39	16.9
1958	3.4	4.8	-	0.00007	0.4	0.4	0	0.19	9.2
1959	13.5	7.6	-	0.001	0.4	3.6	0	0.22	25.3
1960	12.1	7.6	-	0.001	0.4	0.16	0	0.65	20.9
1961	10.2	7.6	-	0.0005	0.4	3.6	0	0.11	21.9
1962	19.0	7.6	-	0.0003	0.4	0.37	0	0.10	27.5
1963	15.3	7.6	-	0.0005	0.4	0.10	0	0.03	23.4
1964	14.5	6.3	-	0.0003	0.4	0.22	0	0.05	21.5
1965	9.3	6.3	-	0.0003	0.4	0.38	0	0.15	16.5
1966	12.9	6.3	-	0.0006	0.4	0.70	0	0.09	20.4
1967	9.4	6.3	-	0.00005	0.4	0.44	0.002	0.10	16.6
1968	14.7	4.9	-	0.0025	0.4	0.49	0.008	0.05	20.5
1969	7.6	4.9	-	0.0016	0.4	0.79	4.2	0.56	18.4
1970	9.0	4.9	-	0.0001	0.4	0.76	0.16	0.08	15.3
1971	7.8	4.9	0.51	0.0007	0.4	0.59	0.16	0.08	14.4
1972	7.4	4.9	0.48	0.0008	0.4	0.06	0.13	0.03	13.4
1973	9.2	4.9	0.61	0.0004	0.4	0.04	0.17	0.01	15.3
1974	6.0	4.5	0.40	0.0002	0.32	0.04	0.036	0.03	11.3
1975	6.4	3.7	0.42	0.000005	0.27	0.003	0.015	0.005	10.8
Totals ^c	201.6	120.0	2.42	0.014	8.2	55.7	4.88	27.8	420.6

a. From fuel reprocessing operations only.

b. ⁸⁵Kr releases during 1954-1970 are classified.

c. Totals not consistent because of roundoff errors.

TABLE 11

Total Population Whole-Body Dose Commitment from Liquid Releases

Dose Commitment, man-rem^a

Total Population Whole Body Dose Commitment from Liquid Releases

<i>Period</i>	<i>Population Size^b</i>	<i>Natural Sources</i>	<i>Artificial Sources</i>	<i>All SRP Sources</i>	<i>Fuel Processing Sources</i>
1975	70,000	8,200	7,400	15.5	5.2
1957-1975	70,000 ^c	109,000	99,000	531	58

^a. From all SRP operations.^b. Based on 1970 census.^c. Assumes a constant population of 20,000 for 1957-1964, increasing to 70,000 in 1965 as a result of startup of the Beaufort-Jasper Water Treatment Plant in 1965.

TABLE 12

Comparative Whole-Body Radiation Dose Commitment, 1975

<i>Dose Commitment^a</i>	<i>Atmospheric Releases (Table 10)</i>	<i>Liquid Releases (Table 11)</i>
Total SRP contribution, ^b rem	115	15.5
Fuel processing, rem	10.8	5.2
Natural Sources, rem	78,000	8,200
Artificial sources, rem	71,000	7,400
SRP contribution, as % of natural	0.15	0.19
Fuel processing, as % of natural	0.014	0.063

^a. Whole body dose commitment to the offsite population within 100 km of SRP and in the Savannah River valley.^b. Includes fuel processing, F-Area and H-Area.

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